Westinghouse Non-Proprietary Class 3

WCAP-17721-NP-A
Revision 0

September 2015

Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology

Westinghouse

WCAP-17721-NP-A
Revision 0

Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology

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Containment and Radiological Analysis

September 2015

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Westinghouse Electric Company LLC
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August 24, 2015

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-17721-P, REVISION 0, AND WCAP-17721-NP, REVISION 0, "WESTINGHOUSE CONTAINMENT ANALYSIS METHODOLOGY - PWR [PRESSURIZED WATER REACTOR] LOCA [LOSS-OF-COOLANT ACCIDENT] MASS AND ENERGY RELEASE CALCULATION METHODOLOGY"

Dear Mr. Gresham:

By letter dated May 3, 2013 (Agencywide Documents Access and Management System Accession No. ML13133A066), Westinghouse submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0 (WCAP-17721-P/NP, Revision 0), "Westinghouse Containment Analysis Methodology - PWR LOCA Mass and Energy Release Calculation Methodology."

By letter dated July 8, 2015, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-17721-P/NP, Revision 0, was provided for your review and comments. By letter dated July 31, 2015, Westinghouse commented on the draft SE. The NRC staff's disposition of Westinghouse's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter. In addition, the NRC staff has made minor editorial changes.

The NRC staff has found that TR WCAP-17721-P/NP, Revision 0, is acceptable for referencing in licensing applications for use of the methodology described in WCAP-17721-P/NP, Revision 0, for calculating the M&E release from a large-break LOCA for a PWR provided that the limitations and condition stipulated in the Section 4.0 of the enclosed NRC final SE are met along with the proper documentation.

Our acceptance applies only to material provided in the subject TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

NOTICE: Enclosure 2 and Attachment 2 transmitted herewith contains proprietary information. When separated from Enclosure 2 and Attachment 2, this document is decontrolled.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Westinghouse and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Mirela Gavrilas, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures:
1. Final Safety Evaluation
   (Non-proprietary version)
2. Final Safety Evaluation
   (Proprietary version)

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ADAMS Accession Nos.: Pkg.: ML15221A005; Ltr.: ML15221A007; Enc. 1: ML15221A008; Enc. 2: ML15221A009; Att.1: ML15221A010; Att. 2: ML15221A011 **NRR-106**

OFFICE	NRR/DPR/ PLPB/PM	NRR/DPR/ PLPB/LA	DSS/SCVB/BC	DSS/SNPB/BC	NRR/DPR/ PLPB/BC	NRR/DPR/DD
NAME	ELenning	DHarrison	RDennig	JDean	AMendiola	MGavrilas
DATE	8/24/2015	8/21/2015	8/24/2015	8/24/2015	8/24/2015	8/24/2015

OFFICIAL RECORD

Letter to James A. Gresham from Mirela Gavrilas dated August 24, 2015

SUBJECT: FINAL SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-17721-P, REVISION 0, AND WCAP-17721-NP, REVISION 0, "WESTINGHOUSE CONTAINMENT ANALYSIS METHODOLOGY - PWR [PRESSURIZED WATER REACTOR] LOCA [LOSS-OF-COOLANT ACCIDENT] MASS AND ENERGY RELEASE CALCULATION METHODOLOGY"

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) WCAP-17721-P, REVISION 0, AND WCAP-17721-NP, REVISION 0,

"WESTINGHOUSE CONTAINMENT ANALYSIS METHODOLOGY - PWR

[PRESSURIZED WATER REACTOR] LOCA [LOSS-OF-COOLANT ACCIDENT]

MASS AND ENERGY RELEASE CALCULATION METHODOLOGY"

WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE)

PROJECT NO. 700

Enclosure 1

TABLE OF CONTENTS

1.0 INTRODUCTION

By letter dated May 3, 2013, Westinghouse submitted TR WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology" (Proprietary/Non-Proprietary) (WCAP-17721-P/NP, Revision 0) (Reference 1) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The purpose of this report was to describe the evaluation framework[1] for the mass and energy (M&E) release input calculation methodology. Because this evaluation framework is specifically focused on the scenario of a LOCA, it is referred to as an evaluation model[2], with a focus on the M&E release.

The M&E evaluation model uses the WCOBRA/TRAC (WC/T) computer code. This code has previously been approved by the NRC for large-break LOCA emergency core cooling system (ECCS) analysis (Reference 2). Much of the information provided in WCAP-17721-P/NP, Revision 0, was previously submitted in WCAP-16608, Appendix C, but was withdrawn before the completion of the review. WCAP-17721-P/NP, Revision 0, is the re-submittal of that information.

The M&E release predicted is the primary input for the calculation of the containment pressure and temperature following a LOCA. The currently approved Westinghouse M&E methodology contains two distinct methodologies, one for Westinghouse Nuclear Steam Supply System (NSSS) and one for Combustion Engineering/ASEA (Allmänna Svenska Elektriska Aktiebolaget) Brown Boveri (CE/ABB) NSSS. The methodology described in WCAP-17721-P/NP, Revision 0, is a single methodology for both NSSS types and is a further development from what is currently approved. The two main improvements in the new M&E evaluation framework described in WCAP-17721-P/NP, Revision 0, are a more realistic treatment of the heat transfer in the steam generator (SG) and the ability to model conduction limited heat transfer from thick metals in the primary and secondary systems. Taking advantage of these two improvements allow for a more mechanistic treatment of the actual M&E release following a LOCA. Because the treatment is more mechanistic, the conservative assumptions used in the previously approved methods are not needed, which ultimately results in lower calculated long term containment pressures and temperatures.

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[1] Similar to an "evaluation methodology" as defined in 10 CFR 50.46, an *evaluation framework* is herein defined as the calculational framework for evaluating behavior of a system during a scenario. It includes one or more computer programs and all other information necessary for application of the calculational framework for a specific scenario, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

[2] As defined in 10 CFR 50.46, an *evaluation model* is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

The complete list of correspondence between the NRC and Westinghouse (W in this table) is provided in Table 1 below. This includes Requests for Additional Information (RAIs), responses to RAIs, audit documentation, and any other relevant correspondence to this review.

Table 1: List of Correspondence

Sender	Document	Document Date	Reference
W	Original Submittal	April 3, 2013	(Reference 1)
NRC	Acceptance Letter	July 31, 2013	(Reference 3)
NRC	Audit Plan	May 28, 2014	(Reference 4)
W	PIRT for Large Break M&E	November 2014	(Reference 5)
W	RG 1.203 Compliance	July 3, 2014	(Reference 6)
NRC	Audit Report	August 14, 2014	(Reference 7)
NRC	RAI - Round 1 (confirmatory analysis)	September 4, 2014	(Reference 8)
W	Responses – Round 1 (confirmatory analysis)	September 25, 2014	(Reference 9)
NRC	RAI - Round 1 (SNPB)	October 20, 2014	(Reference 10)
NRC	RAI - Round 1 (SCVB)	October 20, 2014	(Reference 11)
NRC	Audit Report	October 27, 2014	(Reference 12)
W	Responses – Round 1 (partial SNPB and SCVB)	January 22, 2014	(Reference 13)
W	Responses – Round 1 (partial SNPB)	March 10, 2015	(Reference 14)
W	Responses – Round 1 (partial SNPB and SCVB)	March 16, 2015	(Reference 15)
W	Responses – Round 1 (partial SNPB)	March 26, 2015	(Reference 16)
W	Responses – Round 1 (partial SNPB)	April 22, 2015	(Reference 17)
NRC	Audit Report	May 20, 2015	(Reference 18)
W	Responses – Round 1 (partial SNPB & SCVB)	May 21, 2015	(Reference 19)
NRC	RAI - Round 2 (SNPB)	June 11, 2015	(Reference 20)
W	Responses – Round 1 (partial SNPB & SCVB)	May 26, 2015	(Reference 21)

2.0 REGULATORY EVALUATION

A licensee will use a variety of methods to evaluate the transients and accidents that could occur at its nuclear power plant. The NRC staff reviews these methods to ensure that they provide a realistic or conservative result and that they adhere to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR). Because the results of the transient and accident analysis methods are important to the safety of nuclear power plants, these methods must be maintained under a quality assurance program which meets the criteria set forth in 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Other regulations which are applicable to transient and accident analysis methods are found in the following:

- 10 CFR 50.34, "Contents of Applications; Technical Information," which provides the requirements for the Final Safety Analysis Report required for each plant which includes the analysis of transients and accidents;
- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," which provides the requirements for a LOCA analysis;
- 10 CFR Part 50 Appendix A, "General Design Criteria," which includes the principal design criteria for the facility; and
- 10 CFR Part 50 Appendix K, "ECCS Evaluation Models," which provides further requirements for a LOCA analysis.

There are three General Design Criteria (GDC) in 10 CFR Part 50 Appendix A specifically for containment analysis: GDC 16, GDC 38, and GDC 50.

GDC 16, "Containment design," requires the containment design to be essentially leak tight and that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

GDC 38, "Containment heat removal," requires that a system to remove heat from the containment must be provided. The safety function of the system shall rapidly reduce, consistent with the functioning of other systems, the containment pressure and temperature following any LOCA and it should maintain the pressure and temperature at acceptable levels.

GDC 50, "Containment design basis," as it relates to the containment and sub-compartments being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any LOCA.

Licensees perform simulations to demonstrate that these criteria have been met, and as part of their regulatory oversight the NRC staff will review these simulations. To assure the quality and uniformity of NRC staff reviews, the NRC created NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) (Reference 22) to guide the staff in performing their reviews. Regulatory guidance for containment reviews is provided in Section 6.2.1 of the SRP, "Containment Functional Design" (Reference 23).

The focus of this safety evaluation (SE) is the simulation which predicts the M&E released into containment as a result of the LOCA. The guidance for such M&E reviews is given in Section 6.2.1.3 of the SRP, "Mass and Energy Release Analysis for Postulated Loss-Of-Coolant Accidents (LOCAs)" (Reference 24). Similar guidance was also published in the American Nuclear Society's Standard ANS 56.4, "pressure and temperature transient analysis for light water reactor containments" (Reference 25).

Both SRP Chapter 6.2.1.3 and ANS 56.4 were used by Westinghouse in generating the TR and by the NRC staff in reviewing the TR. This SE specifies each criterion provided in the SRP and ANS guidance and demonstrates how the criterion has been met. The criteria for SRP Chapter 6.2.1.3 are addressed in Section 3.2 of this SE. The criteria for ANS 56.4 are addressed in Section 3.3 of this SE.

3.0 TECHNICAL EVALUATION

The TR WCAP-17721-P/NP, Revision 0, describes the evaluation model Westinghouse will use for performing M&E release simulations following a LOCA for Westinghouse NSSS Large Dry, Sub-Atmospheric, and Ice Condenser Containment Designs, and the CE/ABB NSSS Containment Designs. (The term "containment" will be a generic term to refer to each of these designs hereafter). Specifically, the TR describes how WC/T will be used to simulate the M&E release. The NRC staff's technical evaluation is focused on determining if there is reasonable assurance that WC/T, when used in the manner prescribed in the TR, will result in an appropriate prediction of the M&E released. To better organize the NRC staff's review, the technical evaluation has been separated into the following three sections:

3.1 Background Information

3.2 SRP Chapter 6.2.1.3 Criteria

3.3 ANS 56.4 Criteria

For background information, the NRC staff has provided a discussion on containment analysis in general, how the containment analysis is typically performed, how the analysis is impacted by the M&E release rates calculated, and how the containment analysis and the M&E analysis are coupled. This section concludes with a discussion of the M&E evaluation model defined in WCAP-17721-P/NP, Revision 0.

The next two sections focused on the criteria for review of an M&E evaluation model as given in SRP Chapter 6.2.1.3 and ANS 56.4. Each section provides a listing of all the criteria given in the associated guidance document and how each distinct criterion was met. Where the SRP and ANS guidance had the same criterion, the NRC staff addressed how the criterion is satisfied in one of the sections and referenced that discussion in the other section. By satisfying these criteria, the NRC staff will demonstrate that there is reasonable assurance that the calculations of the M&E release rates for a large-break LOCA will be performed in a manner that conservatively establishes the containment design response.

3.1 Background Information

Following a large-break LOCA, hot fluid from the reactor coolant system (RCS) will flow into containment. This hot fluid will increase the pressure and temperatures in the containment and this increase could challenge the containment in the following two ways:

(1) high pressures in containment could cause excessive leakage or if the containment pressure exceeds its design pressure, the containment may fail, and

(2) high pressure and temperatures could create excessively harsh conditions such that any safety equipment in containment may fail to operate.

To ensure that large-break LOCAs do not result in failure of the containment or the equipment in containment, containments are designed to remove heat and reduce the pressure, and the equipment in containment is designed to operate under harsh conditions. In order to demonstrate that the containment has been adequately designed, licensees must perform an analysis which simulates the conditions in containment following a LOCA. Westinghouse performs such analysis using codes like GOTHIC and LOTIC1. One of the most important inputs to this containment analysis is the rate at which the M&E enters the containment from the RCS. Determining this M&E release rate is crucial to the containment analysis.

3.1.1 Containment Analysis

In order to demonstrate that the containment and its related systems have been appropriately designed, the following three analyses are typically performed:

(1) Containment Peak Pressure Analysis – Performed to ensure the peak pressure in containment does not exceed its design pressure.
(2) Environment Qualification (EQ) Analysis – Performed to ensure that the safety equipment inside containment will continue to function in spite of the higher containment pressures and temperatures.
(3) Net Positive Suction Head available (NPSHa) Analysis – Performed to ensure there is adequate net positive suction head available for the pumps in the containment sump. If not enough is available, then the pumps may experience cavitation and fail.

Each of these analyses is simulating the same event, the containment conditions following a large-break LOCA, but with a different focus. For containment peak pressure analysis and EQ analysis, the peak containment pressure and temperature are the focus. For NPSHa analysis, the maximum containment sump temperature is the focus, as higher temperatures result in less NPSHa. The focus of the analysis is also known as its figure of merit[3]. To ensure that the figure of merit is calculated appropriately, certain inputs are biased during each analysis depending on the figure of merit. One example of input biasing is the [ ]

Consider a large-break LOCA occurring in the cold leg of the RCS. Fluid will flow from the RCS into containment through both sides of the break. However, one side of the break will typically have a much higher enthalpy than the other, as one side of the break will be the fluid from the core which has traveled through the broken loop SG (i.e., higher enthalpy) and the other side of the break will be the excess ECCS injection which has not travelled through the core or a SG (lower enthalpy). WC/T calculates the M&E flowing out of each side of the break and outputs those values to the containment code. [ ] may or may not be assumed depending on the analysis being performed.

For containment peak pressure and EQ analysis, the figures of merit are the peak containment pressure and temperature. To ensure these are conservatively calculated, Westinghouse assumes [ ] The result is a higher steaming rate than if [ ] which results in higher containment pressure and temperatures. For NPSHa analysis, the figure of merit is sump temperature. To ensure that the sump temperature is conservatively calculated, Westinghouse assumes [ ] The result is less steam and more hot water than [ ] which results in higher containment sump temperatures.

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[3] A figure of merit is a variable which is calculated during the analysis and is the most important (or one of the most important) variable used to draw conclusions from the analysis. The figure of merit may have a constant value (e.g., peak containment pressure), but it also may be time dependent value (e.g., wall temperature as a function of time).

3.1.2 Coupling Containment Analysis and Mass & Energy Analysis

While the M&E release is a key input for any containment analysis, the resulting pressure calculated in the containment analysis is also a key input for the M&E analysis. The containment pressure, commonly called the back pressure, will influence the rate at which the M&E will exit the RCS and enter containment. In some situations, it is possible to decouple these two analyses by using a conservative back pressure in the M&E analysis (i.e., a back pressure which results in a higher M&E release than would actually be expected). However, in WCAP-17721-P/NP, Revision 0, Westinghouse proposed two additional ways of treating this coupling between the M&E and containment computer codes in addition to assuming a conservative back pressure: coupling the codes directly and coupling the codes iteratively.

When the codes are coupled directly, WC/T and GOTHIC are run in parallel and exchange information based on their respective time steps. WC/T outputs the M&E which flows out of the break into containment to GOTHIC, and GOTHIC outputs the current containment pressure to WC/T. Because the time step can influence the calculation of M&E release, there is a limitation on the maximum time step allowable by GOTHIC and coupling with any other code besides GOTHIC would require additional NRC review.

When the codes are coupled iteratively for a dry containment application, WC/T and a containment code (GOTHIC) are run in a standalone mode using the following process:

1. A back pressure is assumed.
2. The back pressure is input into WC/T to calculate the M&E release.
3. That calculated M&E release is used in a standalone containment code to calculate a new back pressure (GOTHIC).
4. Steps 2 and 3 are repeated until the change in the back pressure is within some small margin.

WC/T will use a [ ] backpressure curve for ice condenser applications.

3.1.3 Mass & Energy Evaluation Model

The Westinghouse new M&E evaluation model, described in WCAP-17721-P/NP, Revision 0, is an update to the previously approved M&E as described in References 1-8 of WCAP-17721-P/NP, Revision 0. This new M&E evaluation model has been updated with more mechanistic treatment of the heat transfer in the SG, conduction limited heat transfer from thick metal in the RCS, and overall more mechanistic modeling of various aspects of the large-break LOCA through the use of WC/T.

In order to determine if using the M&E evaluation model described in WCAP-17721-P/NP, Revision 0, would result an acceptable prediction of the M&E release, the NRC staff reviewed the various phenomena modeled in WC/T and determined if those phenomena were modeled appropriately. These phenomena can be organized by the phase in which they occur following a large-break LOCA. Those phases are defined in Table 2.

Table 2: Phases following a Large-break LOCA

Phase	Start Time (Seconds)	Starts When	Ends	How the Phase is Simulated
Blowdown	0	When the break initiates.	When the RCS pressure and the primary reactor containment pressure are virtually equal.	WC/T
Refill	20 - 30	When the RCS pressure and the primary reactor containment pressure are virtually equal.	When the ECCS refills the reactor vessel to the bottom of the active core.	WC/T
Reflood	30 - 50	When the ECCS refills the reactor vessel to the bottom of the active core.	When the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core.	WC/T
Post-Reflood	150 - 200	When the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core.	When the temperature of the reactor coolant system and the steam generators are essentially equal.	WC/T
Long Term Steaming (or Decay Heat)	5400 (large dry) 20000 (ice)	When the temperature of the reactor coolant system and the steam generators are essentially equal.	When the RCS and steam generators are in thermal equilibrium with the surroundings.	Separate Analysis

WC/T is used to physically model the first four phases following a large-break LOCA (Blowdown, Refill, Reflood, and Post-Reflood). The Long-Term phase is not modeled directly with WC/T as the phase can last many days and modeling such a simulation in WC/T is not necessary. Instead, WC/T is used to obtain conservative estimates of certain rates and then a separate calculation is performed for this long-term phase.

Because different phenomena are important during each phase, Westinghouse used a Phenomena Identification and Ranking Table (PIRT) to determine the most important phenomena during each phase (Reference 5). Those phenomena and the phase in which they are important are listed in Table 3 along with the sub-section of this SE where they are discussed.

Table 3: Highly Ranked Phenomena for Large-break LOCA M&E

Phenomenon	Phase	Sub-Section
Break Flow	Blowdown	3.2.3.2
Core Stored Energy Release	Blowdown	3.2.1.2, 3.2.1.3
Reflood Heat Transfer	Reflood	3.2.5.3, 3.3.7
Cold Leg/Accumulator Condensation	Reflood	3.3.4.3
Downcomer Condensation	Reflood, Post-Reflood	3.3.4.3
Downcomer Stored Energy Release	Reflood, Post-Reflood	3.2.1.7
Direct Vessel Injection	Reflood, Post-Reflood	N/A[4]
Hot Leg Entrainment/De-Entrainment	Reflood, Post-Reflood, and Long Term	3.2.5.2
Steam Generator Heat Transfer	Reflood, Post-Reflood, and Long Term	3.3.7.3, 3.3.7.4, 3.3.7.5
Upper Plenum Entrainment/De-Entrainment and Condensation	Reflood, Post-Reflood, and Long Term	3.2.5.2
Decay Heat	Reflood, Post-Reflood, and Long Term	3.2.1.5
Loop Flow Split	Reflood, Post-Reflood and Long Term	3.3.4.3
Hot Leg Condensation	Post-Reflood and Long Term	3.3.4.3

Westinghouse will be using WC/T to model the above phenomena. WC/T has been previously reviewed and approved in Westinghouse's ECCS evaluation model (Reference 2). While starting with an approved computer code did significantly reduce the review effort needed, some challenges remained. The most significant challenge was in the change of the figure of merit.

For ECCS analysis, one of, if not the most important, figures of merit is the peak cladding temperature (PCT) resulting from a large-break LOCA. Thus, when WC/T was initially reviewed, the focus was upon ensuring that it would result in an appropriate prediction of PCT. The PCT is influenced by how quickly the energy in the RCS exits out the break and into containment. However, unlike M&E analysis where it is more conservative to maximize that energy release rate from the fuel, in PCT analysis it is often conservative to minimize the energy release rate from the fuel. Minimizing the energy released leaves more energy in the fuel which results in higher cladding temperatures.

By changing the purpose of the WC/T analysis from ECCS to M&E release, the figure of merit changed from PCT to M&E released, and so did the meaning of a *conservative assumption*. In PCT analysis, a conservative assumption is an assumption which results in higher PCT. In M&E analysis, a conservative assumption is an assumption which results in greater M&E release. These two conservatisms are commonly opposed as it is usually conservative in PCT

[4] Direct vessel Injection is only associated with the AP1000, which is beyond of the scope of review for this safety evaluation.

analysis to underpredict the energy release rate and usually conservative in M&E analysis to overpredict the energy release. The shift in the figure of merit from higher PCT to greater M&E release represented a major portion of the review work performed by the NRC staff.

To perform the M&E analysis, Westinghouse began with an ECCS input deck for WC/T. Because the figure of merit had changed, Westinghouse had to modify the input to ensure that the M&E release rates would be appropriately predicted. These modifications, detailed in this SE, resulted in conservative prediction of the M&E release rates which could then be used in a containment code to perform containment analysis.

3.2 SRP Chapter 6.2.1.3 Criteria

The NRC staff followed the guidance found in SRP Chapter 6.2.1.3 (Reference 24) to perform the technical evaluation. It should be noted that the SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, should any differences exist, the applicant would be requested to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

SRP Chapter 6.2.1.3 provides guidance specifically for the review of M&E release analysis from postulated LOCAs. It includes specific criteria for various aspects of the simulation including the sources of energy, modeling of the break flow, and treatment of each phase of the event.

SRP Chapter 6.2.1.3 Section II organizes the review into the seven categories and Section III suggests an additional category. These eight categories are outlined in Table 4. The specific criteria for each category are given in the appropriate sub-sections below.

Table 4: SRP Chapter 6.2.1.3 Review Categories

Section	
3.2.1	Sources of Energy
3.2.2	Break Information and Analysis
3.2.3	Blowdown Phase
3.2.4	Refill Phase
3.2.6	Post-Reflood Phase
3.2.4	Refill Phase
3.2.7	Decay Heat Phase
3.2.8	Confirmatory Analysis

3.2.1 Sources of Energy

The sources of stored and generated energy that should be considered in analyses of a LOCA include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the SG tubing and secondary water.

Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, Paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the RCS and SG metal and increasing the RCS and SG secondary mass to account for uncertainties and thermal expansion. However, the requirements of Paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

SRP Chapter 6.2.1.3 Sub-Section II.1.A directs the reviewer to consider the eight criteria from Appendix K to 10 CFR Part 50 regarding the sources of energy. Those review criteria are outlined in Table 5 and the specific criteria are given in the subsequent sub-sections.

Table 5: Sources of Energy Review Categories

Sub-Section	
3.2.1.1	Reactor Power
3.2.1.2	Initial Stored Energy
3.2.1.3	Fission Heat
3.2.1.4	Decay of Actinides
3.2.1.5	Fission Product Decay
3.2.1.6	Metal-Water Reaction Rate
3.2.1.7	Reactor Internals Heat Transfer
3.2.1.8	Pressurized Water Reactor Primary-to-Secondary Heat Transfer

3.2.1.1 Reactor Power

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.
10 CFR Part 50, Appendix K, Paragraph I.A

In the initial submittal (Reference 1), Westinghouse stated that it will use the [

] The NRC staff agrees with this assessment and acknowledges that the above criterion in Appendix K was written specifically for PCT, where the power distribution and peaking factor would have an impact.

Because Westinghouse is using an appropriate reactor power or [
] the NRC staff has determined that the reactor power has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.2 Initial Stored Energy

Initial 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). To accomplish this, the thermal conductivity of the UO2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.
10 CFR Part 50, Appendix K, Paragraph I.A.1

In the initial submittal (Reference 1), Westinghouse stated that it will use [

] However, the NRC staff is aware that the currently approved version of Westinghouse's fuel performance code, PAD4, does not account for thermal-conductivity degradation. As of the writing of this SE, Westinghouse has submitted a newer version of PAD, PAD5, which does account for thermal conductivity degradation. Additionally, Westinghouse has used an updated version of PAD4 (PAD4TCD) in a number of licensing actions. Therefore, Westinghouse is required to use a fuel performance code which does account for fuel thermal conductivity degradation (such as PAD4TCD) for this analysis to ensure the initial stored energy is appropriate. If Westinghouse chooses to use PAD4TCD, then upon approval of a new fuel thermal mechanical code which does account for thermal conductivity degradation, Westinghouse shall confirm that the initial stored energy calculated using PAD4TCD remains accurate or conservative.

Because Westinghouse is using [

] and using a conservative burnup when determining the initial fuel temperature, the NRC staff has determined that the initial stored energy has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.3 Fission Heat

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.
10 CFR Part 50, Appendix K, Paragraph I.A.2

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is using [

]

the NRC staff has determined that fission heat has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.4 Decay of Actinides

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.

10 CFR Part 50, Appendix K, Paragraph I.A.3

In the initial submittal (Reference 1), Westinghouse stated that it will use [

]

Because Westinghouse is using an appropriate decay heat standard, the NRC staff has determined that the decay of actinides has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.5 Fission Product Decay

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 (Proposed American Nuclear Society Standards—"Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors." Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971). This standard has been approved for incorporation by reference by the Director of the Federal Register. A copy of the standard is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738. 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.

10 CFR Part 50, Appendix K, Paragraph I.A.4

In the initial submittal (Reference 1), Westinghouse stated that [

] The NRC staff has allowed the use of this standard multiple times in the past for similar analysis.

Because Westinghouse is using a well-known decay heat standard, the NRC staff has determined that the fission product decay has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.6 Metal-Water Reaction Rate

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 (Baker, L., Just, L.C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, page 7, May 1962). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of the publication is available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852-2738. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

10 CFR Part 50, Appendix K, Paragraph I.A.5

In the initial submittal (Reference 1), Westinghouse stated that while WC/T has multiple metal-water models in place, it will use the Baker-Just equation.

Because Westinghouse is using the required model, the NRC staff has determined that the metal-water reactor rate has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.7 Reactor Internals Heat Transfer

Reactor Internals Heat Transfer

Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

10 CFR Part 50, Appendix K, Paragraph I.A.6

In the initial submittal (Reference 1), Westinghouse stated that [

] This mechanistic modeling of heat transfer from the reactor internals is one of the additions to the new M&E evaluation model, as the previous M&E evaluation model did not use a mechanistic but a conservative approach. Additionally, Westinghouse referenced analysis

that demonstrated that WC/T has sufficient capability to model the stored energy release in the downcomer, but did not provide this information with the TR. Therefore, this issue was formed into RAI-SNPB-1.

Westinghouse responded to RAI-SNPB-1 by discussing the impact of downcomer stored energy release on both PCT analysis and M&E analysis. It detailed how it was conservative to over-predict this rate for both scenarios. Additionally, it provided WC/T predictions of temperatures and liquid levels that show good agreement between the code and the measured parameters.

Because Westinghouse is physically modeling the heat transferred from the reactor internals, the NRC staff has determined that the reactor internals heat transfer has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.1.8 Pressurized Water Reactor Primary-to-Secondary Heat Transfer

Pressurized Water Reactor Primary-to-Secondary Heat Transfer

Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

10 CFR Part 50, Appendix K, Paragraph I.A.7

This criterion is fully addressed in Section 3.3.7 - Modeling – Heat Transfer Correlations below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.2.2 Break Information and Analysis

SRP Chapter 6.2.1.3 Sub-Section II.1.B and II.1.C.i directs the reviewer to consider three criteria regarding the break and the sub-compartment analysis. Those review criteria are outlined in Table 6 and the specific criteria are given in the subsequent sub-sections.

Table 6: Break Information and Subcompartment Analysis Review Categories

Sub-Section	
3.2.2.1	Break
3.2.2.2	Break Size
3.2.2.3	Subcompartment Analysis

3.2.2.1 Break Location and Type

Break Location and Type

Of several breaks postulated on the basis SRP Chapter 3.6.2, the break selected as the reference case for subcompartment analysis should yield the highest M&E release rates, consistent with the criteria for establishing the break location and area.

SRP Chapter 6.2.1.3, Sub-Section II.1.B.ii

In the initial submittal (Reference 1), Westinghouse described the three break types considered during an M&E evaluation model analysis.

Because Westinghouse is considering each break location in the primary loop consisting of a hot-leg break, a cold-leg break, and a pump-suction leg break, the NRC staff has determined that the reactor break location and type has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.2.2 Break Size

Break Size

Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

SRP Chapter 6.2.1.3, Sub-Section II.1.B.iii

In the initial submittal (Reference 1), Westinghouse discussed previous break sensitivities which demonstrated that the double-ended break is limiting for calculating peak containment pressures and temperatures. However, Westinghouse did not address the use of slot breaks in its new M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-2.

Westinghouse responded to RAI-SNPB-2 by providing a sensitivity study demonstrating a double ended break would be more limiting for M&E release than a slot break of the same size. Any large break will eventually release the same amount of M&E over a longer period of time, but the double ended break releases this M&E faster as each side flows into containment and [ ] Thus, the M&E is transferred to containment faster, resulting in a more limiting event.

Because Westinghouse is using a double-ended sized break which it has demonstrated results in the highest M&E release and containment pressure, the NRC staff has determined that break size has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.2.3 Subcompartment Analysis

Subcompartment 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 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.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.i

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model is using the computer code from its approved ECCS evaluation model, WC/T. [ ] Additionally, Westinghouse provided information on how it would bias the input to the M&E evaluation model which would result in a conservative M&E prediction.

Because Westinghouse is using an analytical approach which is similar to a currently approved ECCS evaluation model, the NRC staff has determined that the M&E release for subcompartment analysis has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

To avoid potential confusion, the NRC staff is not approving WC/T to perform subcompartment analysis separate from that used during a larger-break LOCA. Such considerations were beyond the NRC staff's scope of review.

3.2.3 Blowdown Phase

The blowdown phase is defined as the time from break initiation until the time when the RCS pressure and the primary reactor containment pressure are virtually equal (Reference 25). Westinghouse added that during this phase, the containment pressure (for a dry containment) has increased substantially due to the rapid M&E release. The RCS is mostly voided and the pressure is approximately equal to the containment pressure. The SG pressure is at or near the safety valve setpoint because the turbine is tripped and the main steam isolation valves (MSIVs) are closed.

SRP Chapter 6.2.1.3 Sub-Section II.1.C.ii directs the reviewer to consider four criteria regarding the blowdown phase. Those review criteria are outlined in Table 7 and the specific criteria are given in the subsequent sub-sections.

Table 7: Blowdown Phase Review Categories

Sub-Section	
3.2.3.1	Initial Mass of Water
3.2.3.2	Mass Release Rates
3.2.3.3	Heat Transfer from Primary Surfaces
3.2.3.4	Heat Transfer from Secondary Surfaces

3.2.3.1 Initial Mass of Water

Initial Mass of Water

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 1.02 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.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.ii

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is using [

] the NRC staff has determined that the initial mass of water has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.3.2 Mass Release Rates

Mass Release Rates

Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.ii

In the initial submittal (Reference 1), Westinghouse stated that it is using the same break flow model as used in the ECCS evaluation model. However, Westinghouse did not demonstrate that the break flow model used in the ECCS evaluation model would provide an adequate estimate of break flow for the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-3.

Westinghouse responded to RAI-SNPB-3 by verifying that the break flow model was consistent over its application domain and did not systematically under or over-predict the break flow, [

] Westinghouse further detailed the method by which it makes the break flow prediction conservative for M&E cases by increasing the RCS volume and skewing RCS temperature. Additionally, Westinghouse provided analysis which demonstrated that the currently approved and proposed analysis predicted very similar behavior during blowdown through reflood, only significantly deviating during post-reflood and beyond.

Because the break flow model was not systematically biased to under-predict but rather over-predict the break flow and because the results during blowdown and reflood are consistent with the currently approved methods, the NRC staff has determined that mass release rates have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.3.3 Heat Transfer from Primary Surfaces

Heat Transfer from Primary Surfaces

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.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.ii

This criterion is fully addressed in Sub-Section 3.3.7 - Modeling – Heat Transfer Correlations below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.2.3.4 Heat Transfer from Secondary Surfaces

Heat Transfer from Secondary Surfaces
Calculations of heat transfer from the secondary coolant to the steam generator 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 Chapter 6.2.1.3, Sub-Section II.1.C.ii

This criterion is fully addressed in Sub-Section 3.3.7 - Modeling – Heat Transfer Correlations below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.2.4 Refill Phase

The refill phase is defined as the time from the end of the blowdown (i.e., the primary reactor and containment pressure are virtually equal) to the time when the ECCS refills the reactor vessel to the bottom of the active core (Reference 25). Westinghouse added that at the beginning of this phase the accumulators are injecting into the cold legs, but the downcomer and lower plenum of the vessel are mostly voided. The lower plenum pressure is starting to increase. The containment pressure is constant or slowly decreasing. Further, at the end of this phase accumulator injection has filled the vessel lower plenum to the bottom of the active fuel. The SG pressure remains high.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii, directs the reviewer to consider one criterion regarding the refill phase. That review criterion is outlined in Table 8 and the specific criterion is given in the subsequent sub-section.

Table 8: Refill Review Categories

Sub-Section	
3.2.4.1	Justification for Refill

3.2.4.1 Justification for Refill

Justification for Refill
Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when the 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.
SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii

In the initial submittal (Reference 1), Westinghouse stated that the ECCS evaluation model had been validated for refill calculations by comparison with experimental data. However, Westinghouse did not describe that validation nor demonstrate that the validation applies to the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-4.

Westinghouse responded to RAI-SNPB-4 by describing how refill was validated for WC/T. This validation demonstrated that WC/T over predicts the ECCS bypass during refill. This validation was further discussed during the RAI audit conducted on april 8-9, 2015 (Reference 18) where it was demonstrated through comparisons of WC/T to test data. While conservative for PCT analysis, this over prediction could be non-conservative for M&E analysis. Westinghouse justified the use of WC/T for M&E by demonstrating the impact of the delay, at most 20 seconds, was minimal. The power released during refill is small compared with the power released during blowdown and reflood. Further, the limiting time periods for M&E release occur during blowdown (where refill is irrelevant) or in post reflood (which occurs a very long time after refill).

Because Westinghouse is mechanistically modeling the refill period using the approved ECCS models in WC/T and those approved ECCS have been demonstrated to reasonably predict the refill period for M&E analysis, the NRC staff has determined that refill phase has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.5 Reflood Phase

The reflood phase is defined as the time from the end of the refill (i.e., when the ECCS refills the reactor vessel to the bottom of the active core) to the time when the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core (Reference 25). Westinghouse added that at the beginning of this phase the fuel temperature is slowly increasing, safety injection has actuated, and water is just starting to cover the active fuel. Further, at the end of this phase safety injection has quenched the core, the collapsed liquid level in the core is stable and slowly increasing, and the fuel temperatures are dropping. A frothy two-phase mixture is exiting the vessel. The SG pressure remains high. Containment pressure could be constant or slowly increasing (depending on the design).

SRP Chapter 6.2.1.3 Sub-Section II.1.C.iii directs the reviewer to consider four criteria regarding the reflood phase. Those review criteria are outlined in Table 9 and the specific criteria are given in the subsequent sub-sections.

Table 9: Reflood Phase Review Categories

Sub-Section	
3.2.5.1	Core Flooding Rate
3.2.5.2	Liquid Entrainment
3.2.5.3	Steam Quenching
3.2.5.4	Steam Exiting the Steam Generators

3.2.5.1 Core Flooding Rate

Core Flooding Rate

Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii

In the initial submittal (Reference 1), Westinghouse stated that it is using the same code to calculate the M&E release that is used to calculate the ECCS operating conditions. Further, the ECCS evaluation model had been validated for core reflood calculations by comparison with experimental data. However, Westinghouse did not describe that validation nor demonstrate that the validation applies to the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-5.

Westinghouse responded to RAI-SNPB-5 with an analysis of quench timings from various tests that were modeled in WC/T. Quench timings are a figure of merit associated with the core flooding rate, as that rate (along with other parameters) will impact the quench timings.

Westinghouse provided validation which demonstrated that the quench timings [

]

Because WC/T is calculating the core flooding rate based on ECCS operating conditions and that calculation results in a conservative over-prediction of heat transfer from the core, the NRC staff has determined that the core flooding rate has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.5.2 Liquid Entrainment

Liquid Entrainment

Calculations of liquid entrainment, i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR full length emergency cooling heat transfer experiments. Liquid entrainment should be assumed to continue until the water level in the core is 61 cm (2 feet) from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 46 cm (18-inch) core level, a linearly increasing CRF to 0.80 at the 61 cm (24-inch) level, and a constant CRF of 0.80 until the water level is 61 cm (2 feet) from the top of the core. Above this level, a CRF of 0.05 may be used.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model calculates the liquid entrainment from the core mechanistically, as it is based on the ECCS evaluation model. Further, it stated that the ECCS evaluation model has been validated for the liquid entrainment calculations by comparison with experimental data. However, Westinghouse did not describe if the validation which was used to justify the ECCS evaluation model would also justify the M&E evaluation model. Therefore, this issue was formed into three RAIs. RAI-SNPB-6 focused on entrainment in the core. RAI-SNPB-7 focused on entrainment in the upper plenum. RAI-SNPB-8 focused on entrainment in the hot leg.

Westinghouse responded to RAI-SNPB-6 by providing WC/T simulations of multiple reflood tests. The experimental test data and WC/T predictions demonstrated that the computer code [

]

Westinghouse responded to RAI-SNPB-7 by providing additional information on the WC/T predictions of experimental data. The concern is that the upper plenum could de-entrain too much liquid which would result in less liquid to the SGs and less overall heat transfer from the SGs to containment. It should be noted that there is no test data which explicitly measures the de-entrainment in the upper plenum. Therefore, other figures of merit must be chosen.

Westinghouse chose to compare [

] While each comparison does not explicitly focus on upper plenum de-entrainment, upper plenum de-entrainment would be expected to have a large impact. The results from the three comparisons indicated that WC/T either accurately or conservatively predicts the experimental data.

Westinghouse responded to RAI-SNPB-8 by providing additional information on the sensitivity study performed in the hot leg. The study demonstrated that the cooling rate of the broken loop SG was [

]

Because Westinghouse has demonstrated that [

] the NRC staff has determined that liquid entrainment has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.5.3 Steam Quenching during Reflood

> **Steam Quenching during Reflood**
>
> *The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.*
>
> SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii

In the initial submittal (Reference 1), Westinghouse stated that the ECCS evaluation model had been validated for steam quenching by comparison with experimental data. However, Westinghouse did not describe that validation nor demonstrate that the validation applies to the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-9.

Westinghouse responded to RAI-SNPB-9 by describing an experiment that was used specifically to capture the behavior of steam quenching by injecting water into cold leg under post-accident conditions. The comparisons between the test data and WC/T demonstrate that the code does a reasonable job of predicting the bulk fluid conditions during quenching. Specifically, Westinghouse provided WC/T comparisons to test data for fluid temperatures in the cold leg downstream of the injection location.

Because WC/T is able to reasonably predict the fluid temperature profiles that are the result of quenching, the NRC staff has determined that steam quenching has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.5.4 Steam Exiting the Steam Generators

Steam Exiting the Steam Generators

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.iii

This criterion is fully addressed in Sub-Section 3.3.7.3 - Steam Generator Tubes to Reactor Coolant below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.2.6 Post-Reflood Phase

The post-reflood phase is defined as the time from the end of the reflood (i.e., the time the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core) to the time when the temperature of the RCS and the SGs are essentially equal (Reference 25). Westinghouse added that at the beginning of this phase the core is quenched. A frothy two-phase mixture is entering the SG tubes and the lower inlet section has started to quench. Further, at the end of this phase sump recirculation has started. The SG tubes have quenched and the remaining secondary-side energy is being transferred to containment. The SG fluid and metal is cooling from the tubesheet up and containment pressure is past peak and decreasing.

SRP Chapter 6.2.1.3 Sub-Section II.1.C.iv directs the reviewer to consider two criteria regarding the post-reflood phase. Those review criteria are outlined in Table 10 and the specific criteria are given in the subsequent sub-sections.

Table 10: Post-Reflood Phase Review Categories

Sub-Section	
3.2.6.1	Remaining Stored Energy
3.2.6.2	Steam Quenching during Post-Reflood

3.2.6.1 Remaining Stored Energy

Remaining Stored Energy

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.iv

In the initial submittal (Reference 1), Westinghouse discussed the treatment of the stored energy following reflood. Unlike the currently approved M&E evaluation model which releases all of the remaining heat within one hour of event initiation, the proposed M&E evaluation model mechanistically models the transfer of that heat from the primary and secondary sources to the primary fluid. This mechanistic heat transfer lasts until sump recirculation has started and the containment pressure is past its peak and is decreasing. The phase following post-reflood is the long-term steaming phase, which is described in Section 3.2.7.1, "Steam from Decay Heat."

To ensure that the mechanistic modeling was performed adequately, the important phenomena were considered and their treatment in the WC/T M&E evaluation model is described in Table 3 above. Additionally, the NRC staff performed a confirmatory analysis to ensure that the predictions of WC/T were reasonable when compared with the computer code TRACE. That analysis is detailed in Section 3.2.8, "Confirmatory Analysis," below.

Because Westinghouse is removing the primary and secondary stored energy using a mechanistic model during the post-reflood and a conservative model during the long-term steaming phase, the NRC staff has determined that remaining stored energy and its associated heat transfer has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.6.2 Steam Quenching during Post-Reflood

Steam Quenching during Post-Reflood
Steam quenching should be justified by comparison with applicable experimental data.
SRP Chapter 6.2.1.3, Sub-Section II.1.C.iv

In the initial submittal (Reference 1), Westinghouse stated that the ECCS evaluation model had been validated for steam quenching by comparison with experimental data. However, Westinghouse did not describe that validation nor demonstrate that the validation applies to the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-9.

Westinghouse responded to RAI-SNPB-9 by describing an experiment that was used specifically to capture the behavior of steam quenching by injecting water into cold leg under post-accident conditions. The comparisons between the test data and WC/T demonstrate that the code does a reasonable job of predicting the bulk fluid conditions during quenching. Specifically, Westinghouse provided WC/T comparisons to test data for fluid temperatures in the cold leg downstream of the injection location. Further, during the audit conducted on April 8-9, 2015 (Reference 18), Westinghouse provided details (also captured in RAI-SNPB-10) about how quenching was conservatively decreased to generate more steam and result in a higher peak containment pressure or conservatively increased to generate more hot water and result in a higher sump temperature, depending on the analysis being performed.

Because Westinghouse has demonstrated that the steam quenching model results in reasonable predictions of steam quenching and because Westinghouse is taking steps to conservatively bias the steam quench (either minimizing it for peak pressure or maximizing it for peak sump temperature), the NRC staff has determined that steam quenching has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.7 Decay Heat Phase

The decay heat phase (or the Long-Term Steaming Phase) is defined as the time from the end of the post-reflood phase (i.e., the time when the temperature of the RCS and the SGs are essentially equal) to the time when the remaining decay heat, and the remaining sensible heat in the RCS and secondary system (both fluid and metal) are released (Reference 25). Westinghouse added that at the beginning of this phase there is primarily a liquid release from the vessel side and a saturated steam or two-phase release from the SG side of the break.

SRP Chapter 6.2.1.3, Sub-Section II.1.C.v directs the reviewer to consider two criteria regarding the decay heat phase. Those review criteria are outlined in Table 11 and the specific criteria are given in the subsequent sub-sections.

Table 11: Decay Heat Phase Review Categories

Sub-Section	
3.2.6.1	Remaining Stored Energy
3.2.6.2	Steam Quenching during Post-Reflood

3.2.7.1 Steam from Decay Heat

Steam from Decay Heat

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 Chapter 6.2.1.3, Sub-Section II.1.C.v

In the initial submittal (Reference 1), Westinghouse stated that it used a long-term decay heat boil-off model. However, Westinghouse did not describe how the steam-water mixing was calculated in this long-term boil off model. Therefore, this issue was formed into RAI-SNPB-11.

Westinghouse responded to RAI-SNPB-11 by providing a very detailed explanation of its long-term boil off model. The long-term boil off model considers the steam from decay heat as well as the steam produced from the remaining heat from the SG secondary metal, SG secondary fluid, and RCS metal. The long-term decay heat is calculated outside of WC/T, but uses an

[

] The only exception to this process is a period of 24 hours following the peak pressure in an ice condenser. The non-mechanistic assumption described above would create unrealistic containment pressurization. [

]

[ ] Westinghouse ensured that the ramping was conservative by comparing it with a WC/T analysis. This comparison demonstrated that the ramping was somewhat mechanistic, but ultimately conservative compared to the WC/T results over the same 24 hour period.

Because Westinghouse is using a conservative method to calculate the long-term boil off, which includes a conservative method for calculating the steam from the decay heat (and other heat sources), the NRC staff has determined that steam from decay heat has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.7.2 Remaining Decay Heat

> **Remaining Decay Heat**
>
> *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 SRP Chapter 9.2.5.*
>
> SRP Chapter 6.2.1.3 Sub-Section II.1.C.v

In the initial submittal (Reference 1), Westinghouse stated that [ ] While this modeling of decay heat was determined to be appropriate, Westinghouse did not fully describe what portions of the event are modeled explicitly in the simulation, when the simulation ends, and how the remaining energy (including decay heat) is treated once the simulation ends. Therefore, this issue was formed into RAI-SNPB-12.

Westinghouse responded to RAI-SNPB-12 by providing a table which defines each phase of the event and another table which provides a summary of the energy inventory in each of the main components during each phase. By provided these definitions, the NRC staff could better understand how decay heat was being treated during the different phases of the accident.

Because the decay heat is being appropriately accounted for using a conservative decay heat model, the NRC staff has determined that the remaining decay heat has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.2.8 Confirmatory Analysis

The NRC staff performed a confirmatory analysis by comparing the results of TRACE to WC/T for an M&E analysis. This analysis was performed to demonstrate that when WC/T is used with the appropriate inputs for M&E analysis, its results are comparable to other system codes such as TRACE.

Table 12: Confirmatory Analysis Review Categories

Sub-Section
3.2.8.1 Confirmatory Analysis

3.2.8.1 Confirmatory Analysis

Confirmatory Analysis

The reviewer may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

SRP Chapter 6.2.1.3, Sub-Section III

The NRC staff performed a confirmatory analysis using the TRACE code. This code does share a lineage with WC/T (as they are both further developments of the TRAC-P code). In order to support this analysis and the comparison of the results the NRC staff needed more information; therefore, this information was requested in the eight Confirmatory RAIs (RAI-Confirmatory-1 through RAI-Confirmatory-8). Westinghouse provided the requested information.

Because Westinghouse's approved and new M&E evaluation model have very little response change to the hot-leg break, the confirmatory analysis focused on the cold-leg and pump-suction breaks. Both TRACE and WC/T predicted similar flow rates exiting both sides of the break.

The NRC staff performed a confirmatory analysis which demonstrated that TRACE and WC/T result in similar predictions of the M&E release and resulting containment pressures following a large-break LOCA. The NRC staff has concluded that this optional criterion has been satisfied.

3.3 ANS 56.4 Criteria

Along with the guidance from SRP Chapter 6.2.1.3, the NRC staff followed the guidance of ANS 56.4 (Reference 25) in performing this technical evaluation. Unlike the SRP, the guidance of ANS 56.4 is not formally endorsed by the NRC, and its use here is no such endorsement. This guidance was followed by Westinghouse. Upon reviewing the guidance, the NRC staff found the criteria a very helpful supplement to the SRP.

ANS 56.4 provides guidance in performing an acceptable analysis for determining the pressure and temperature histories in reactor containment during design basis and other events. It includes specific criteria for various aspects of the simulation including initial conditions, sources of energy, and modeling the heat transfer.

ANS 56.4 Section 3.2 organizes the review into the seven categories outlined in Table 13. The specific criteria for each category are given in the appropriate sections below.

Table 13: ANS 56.4 Review Categories

Section	
3.3.1	Energy Sources
3.3.2	Initial Conditions
3.3.3	Single Failures and Nonemergency Power
3.3.4	Modeling
3.3.5	Modeling – Break Flow
3.3.6	Modeling – Primary Containment Backpressure
3.3.6	Modeling – Heat Transfer Correlations

3.3.1 Energy Sources

ANS 56.4, Section 3.2.1, directs the reviewer to consider twelve criteria regarding sources of energy. Those review criteria are outlined in Table 14 and the specific criteria are given in the subsequent sub-sections.

Table 14: Energy Sources Review Categories

Sub-Section	
3.3.1.1	Reactor Coolant System Water and Metal
3.3.1.2	Steam Generator Secondary Water and Metal
3.3.1.3	Core Stored Energy
3.3.1.4	Fission Heat
3.3.1.5	Decay of Actinides
3.3.1.6	Fission Product Decay
3.3.1.7	Metal-Water Reaction Rate
3.3.1.8	Main Steam Lines
3.3.1.9	Main Feedwater Line
3.3.1.10	Auxiliary Feedwater System
3.3.1.11	ECCS Flow
3.3.1.12	Safety Injection Tank Nitrogen Expansion

3.3.1.1 Reactor Coolant System Water and Metal

Reactor Coolant System Water and Metal

Maximizing reactor coolant system water inventory and metal energy is conservative for RCS M&E release calculations. Since the reactor coolant system water inventory is an important parameter, the inventory shall be accurately determined. The increase in the RCS 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 RCS pressure boundary and internals metal thermally in contact with the RCS water shall be included.

ANS-56.4-1983, Sub-Section 3.2.1.1

This criterion is partially addressed in Sub-Sections 3.2.3.1, "Initial Mass of Water," and 3.2.1.7, "Reactor Internals Heat Transfer," above. In the initial submittal (Reference 1), Westinghouse also stated that [

]

Because Westinghouse is using an appropriate power level, [ ] and accounting for all appropriate stored energy in the RCS and SG metal, the NRC staff has determined that the RCS water and metal has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.2 Steam Generator Secondary Water and Metal

Steam Generator Secondary Water and Metal

Maximizing steam generator secondary water inventory and metal energy is conservative for RCS M&E release calculations. Since the steam generator secondary water inventory is an important parameter, a careful determination of the inventory shall be made. The steam generator secondary volume resulting from the pressure and temperature conditions at the initial power level defined in 3.2.2.2 shall be included. Stored energy in all steam generator secondary pressure boundary and internals metal thermally in contact with the steam generator secondary water shall be included.

ANS-56.4-1983, Sub-Section 3.2.1.2

In the initial submittal (Reference 1), Westinghouse stated that [ ]

Because Westinghouse is accounting for the metal in the SG and is using an appropriate water volume, mass, and energy, the NRC staff has determined that the SG secondary water and metal has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3 Core Stored Energy

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.

ANS-56.4-1983, Sub-Section 3.2.1.3

This criterion is fully addressed in Sub-Section 3.2.1.2 - Initial Stored Energy above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.4 Fission Heat

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.

ANS-56.4-1983, Sub-Section 3.2.1.4

This criterion is fully addressed in Sub-Section 3.2.1.3 - Fission Heat above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.5 Decay of Actinides

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-6.1-1979 for end-of-life operation time.

ANS-56.4-1983, Sub-Section 3.2.1.5

This criterion is fully addressed in Sub-Section 3.2.1.4, “Decay of Actinides,” above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.6 Fission Product Decay

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-6.1-1979 for end-of-life operation time.

ANS-56.4-1983, Sub-Section 3.2.1.6

This criterion is fully addressed in Sub-Section 3.2.1.5, “Fission Product Decay,” above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.7 Metal-Water Reaction Rate

> **Metal-Water Reaction Rate**
>
> *For both Boiling Water Reactors (BWRs) and PWRs, the amount of metal water reaction shall be calculated according to the requirements of 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors" (lo). The metal water reaction shall be assumed to occur uniformly over a period no greater than two minutes following the end of reactor vessel blowdown.*
>
> ANS-56.4-1983, Sub-Section 3.2.1.7

This criterion is fully addressed in Sub-Section 3.2.1.6, "Metal-Water Reaction Rate," above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.8 Main Steam Lines

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.1.8

In the initial submittal (Reference 1), Westinghouse stated that [

] Through the RAI process, Westinghouse determined that the steam generators [ ]

Because Westinghouse is stopping steam flow to the turbine in a conservatively short time, the NRC staff has determined that the main steam lines have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.9 Main Feedwater Line

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.1.9

In the initial submittal (Reference 1), Westinghouse stated that the main feedwater is tripped at event initiation and no feedwater flow needs to be considered as the energy released from modeling feedwater addition after the reactor trip is negligible compared with the total energy released to containment. However, Westinghouse did not provide any estimates of the amount of additional energy main feedwater flow would likely add to the secondary side. Therefore, this issue was formed into RAI-SNPB-13.

Westinghouse responded to RAI-SNPB-13 by discussing a study which examined feedwater flow's impact on the M&E event. Westinghouse quantified the impact of considering the coast down of feedwater flow on the additional energy added to the SGs and the resultant pressures in containment. While the consideration of feedwater flow did slightly increase the energy in the SGs, this increase was imperceptible on the resulting containment pressures and temperatures. This is due to the fact that the energy in the SG fluid (both the broken and intact) is not the major contributor to the energy release during any phase. Thus, a small increase in the total energies has an insignificant impact on the overall energy release to containment.

Because this additional energy would result in a very small increase in the SG fluid energies which would result in an almost imperceptible change in the peak containment pressures and temperatures, the NRC staff has determined that the main feedwater flow can be neglected. The NRC staff has concluded that this criterion has minimal impact on the M&E analysis and therefore does not need to be satisfied.

3.3.1.10 Auxiliary Feedwater System

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 flow may be conservatively assumed to be zero.

ANS-56.4-1983, Sub-Section 3.2.1.10

In the initial submittal (Reference 1), Westinghouse stated it had the capability to model the auxiliary feedwater flow and extraction steam, such that the analysis would consider the minimum flows from these systems if they were available and also use a conservatively long actuation time delay. However, the NRC staff questioned the modeling of auxiliary feedwater and extraction steam when the modeling of main feedwater was deemed to be insignificant. Therefore, this issue was formed into RAI-SNPB-14.

Westinghouse responded to RAI-SNPB-14 by clarifying its position on the auxiliary feedwater system. [ ]

Because Westinghouse is [ ] the NRC staff has determined that auxiliary feedwater system has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.11 ECCS Flow

> **ECCS Flow**
>
> *Flow from the ECCS (for example, safety injection tanks, core spray pumps, safety injection pumps) 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.*
>
> ANS-56.4-1983, Sub-Section 3.2.1.11

In the initial submittal (Reference 1), Westinghouse stated that the single failure chosen would be [

]

Because Westinghouse is considering the failure of [
] the NRC staff has determined that the ECCS flow has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.12 Safety Injection Tank Nitrogen Expansion

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.1.12

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is [ ] the NRC staff has determined that the safety injection tank nitrogen expansion has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2 Initial Conditions

ANS 56.4, Section 3.2.2, directs the reviewer to consider nine criteria regarding the initial conditions. Those review criteria are outlined in Table 14 and the specific criteria are given in the subsequent sub-sections.

Table 15: Initial Conditions Review Categories

Sub-Section	
3.3.2.1	Time of Life
3.3.2.2	Power Level
3.3.2.3	Core Inlet Temperature
3.3.2.4	Reactor Coolant System Pressure
3.3.2.5	Steam Generator Pressure
3.3.2.6	Reactor Coolant System Pressure
3.3.2.7	Steam Generator Water Level
3.3.2.8	Core Parameters
3.3.2.9	Safety Injection Tanks

3.3.2.1 Time of Life

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.

ANS-56.4-1983, Sub-Section 3.2.2.1

This criterion is fully addressed in Sub-Sections 3.2.1.2, "Initial Stored Energy," and 3.2.1.5, "Fission Product Decay," above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.2 Power Level

Power Level

The initial power level shall be at least as high as the licensed power level plus uncertainties such as instrumentation American National Standard ANSIIANS-66.4-1983 error (typically 102 percent of the licensed power level).

ANS-56.4-1983, Sub-Section 3.2.2.2

This criterion is fully addressed in Sub-Section 3.2.1.1, "Reactor Power," above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.3 Core Inlet Temperature

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 accident.
ANS-56.4-1983, Sub-Section 3.2.2.3

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is using [
] the NRC staff has determined that the core inlet temperature has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.4 Reactor Coolant System Pressure

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.
ANS-56.4-1983, Sub-Section 3.2.2.4

In the initial submittal (Reference 1), Westinghouse stated that [
]

Because Westinghouse is using [ ] the NRC staff has determined that the RCS pressure has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.5 Steam Generator Pressure

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.
ANS-56.4-1983, Sub-Section 3.2.2.5

In the initial submittal (Reference 1), Westinghouse stated that a short steady state calculation is used to establish and verify the operating conditions before the M&E simulation is performed, and that the SG pressure is adjusted to obtain the target steady state RCS average temperature. However, Westinghouse did not confirm that this resulting SG pressure would be greater than or equal to the pressure plus uncertainty in the SG. Therefore, this issue was formed into RAI-SNPB-15.

Westinghouse responded to RAI-SNPB-15 by discussing the steady state calculation which is used to obtain the SG secondary side pressure. For this calculation [ ] This assumption results in a conservatively high SG heat transfer which acts to increases the secondary side calculated pressure. Further, during the audit conducted on April 8-9, 2015 (Reference 18), Westinghouse confirmed that the reactor power used for the steady state calculation was the [ ]

Because Westinghouse is considering the [ ] and is using an assumption which results in a greater than expected heat transfer in the SG resulting in a larger secondary side pressure, the NRC staff has determined the SG pressure has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.6 Reactor Coolant System Pressure

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.2.6

In the initial submittal (Reference 1), Westinghouse stated that [ ]

Because Westinghouse is using the [ ] the NRC staff has determined that the RCS pressurizer level has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.7 Steam Generator Water Level

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.2.7

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is using the [ ] the NRC staff has determined that the RCS pressurizer level has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.8 Core Parameters

> **Core Parameters**
>
> *Initial core parameters (including physics parameters, fuel properties, gas conductivity) shall be chosen so as to maximize core stored energy.*
>
> ANS-56.4-1983, Sub-Section 3.2.2.8

In the initial submittal (Reference 1), Westinghouse stated [

] However, as stated in Sub-Section 3.2.1.2, the currently approved version of PAD4 does not adequately model thermal conductivity degradation as a function of burnup and a fuel performance code which does account for fuel thermal conductivity degradation (such as PAD4TCD) must be used.

Because Westinghouse is using fuel parameters which maximize core stored energy, the NRC staff has determined that the core parameters have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.2.9 Safety Injection Tanks

> **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.*
>
> ANS-56.4-1983, Sub-Section 3.2.2.9

In the initial submittal (Reference 1), Westinghouse stated that [

] However, Westinghouse did not indicate whether measurement uncertainties were included in the volume and temperature values. Therefore, this issue was formed into RAI-SNPB-16.

Westinghouse responded to RAI-SNPB-16 by clarifying that the initial temperature is set to the [ ] and the initial volume is set to the [ ]

Because Westinghouse had selected values which are consistent with common practice and which result in the [ ] the NRC staff has determined the safety injection tanks have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.3 Single Failures and Nonemergency Power

ANS 56.4, Section 3.2.3, directs the reviewer to consider three criteria regarding single failures and nonemergency power. Those review criteria are outlined in Table 16 and the specific criteria are given in the subsequent sub-sections.

Table 16: Single Failures and Nonemergency Power Review Categories

Sub-Section	
3.3.3.1	Single Active Failures
3.3.3.2	Single Passive Failures
3.3.3.3	Nonemergency Power

3.3.3.1 Single Active Failures

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 system (including primary containment heat removal system; see 4.2.5) required to mitigate the consequences of the event need to be considered.

Potential single active failures which affect the M&E release rates may include the failure of an emergency diesel generator, safety injection pump, core spray pump, emergency feedwater pump, or component cooling water pump.

Further guidance on single active failures is given in IEEE 379-1977, ANSIIANS51.1-1983, and ANSIIANS-52.1-1983.

ANS-56.4-1983, Sub-Section 3.2.3.1

In the initial submittal (Reference 1), Westinghouse stated that [

]

Because Westinghouse is considering the failure of [ ] for the M&E analysis both the resulting pressure and temperature in containment will be biased high and no other single failure would conceivably result in a more limiting condition, the NRC staff has determined the single active failure has been adequately modeled for containment peak pressure and temperature analyses.

During the audit (Reference 18), Westinghouse confirmed that according to NUREG-0588, EQ analysis uses the peak containment pressure calculation (which includes the single worst failure). For NPSHa, the analysis is typically performed following Regulatory Guide 1.82, but this analysis is not performed directly by Westinghouse, and is typically performed by the individual licensee.

Because Westinghouse is considering the single worst active failure, the NRC staff has determined that the single active failure has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.3.2 Single Passive Failures

Single Passive Failures

Passive failures normally need not be considered. Guidance is provided on passive failures and their consideration in ANSI/ANS-51.1-1983 and ANSI/ANS-52.1-1983.

ANS-56.4-1983, Sub-Section 3.2.3.2

In the initial submittal (Reference 1), Westinghouse stated that [
]

Because Westinghouse is not required to consider a passive failure, the NRC staff has concluded that this criterion has been satisfied.

3.3.3.3 Nonemergency Power

Nonemergency Power

The loss of nonemergency 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.

ANS-56.4-1983, Sub-Section 3.2.3.3

In the initial submittal (Reference 1), Westinghouse stated that [ ]

Because Westinghouse is [ ] the NRC staff has determined that the nonemergency power is being ignored and therefore adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4 Modeling

ANS 56.4 Section 3.2.4 directs the reviewer to consider eight criteria regarding general modeling options. Those review criteria are outlined in Table 17 and the specific criteria are given in the subsequent sub-sections.

Table 17: Modeling Review Categories

Sub-Section	
3.3.4.1	Nodalization
3.3.4.2	Thermodynamic Conditions
3.3.4.3	Flow Modeling
3.3.4.4	Pump Characteristics
3.3.4.5	Core Modeling
3.3.4.6	Modeling of Metal Walls
3.3.4.7	Modeling of Auxiliary Flows
3.3.4.8	Post-blowdown Modeling

3.3.4.1 Nodalization

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 RCS shall be modeled with sufficient detail so that the quality at the break location shall not be over predicted. Sufficient detail shall also be provided so that the modeling of core-to-coolant, metal-to coolant, and steam generator-to-coolant heat transfer (PWR only) shall predict conservatively high primary containment pressure. Phase separation in each node shall be justified by experimental data or modeled to predict conservatively high primary containment pressure. The fraction of the core flow which goes through the steam generator in the loop with the rupture and the fraction through the steam generator) in the loop(s) without the rupture shall be calculated to predict conservatively high primary containment pressure.

ANS-56.4-1983, Sub-Section 3.2.4.1

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model uses the same nodalization as the ECCS evaluation model. However, Westinghouse did not provide justification that the nodalization used for the ECCS evaluation model would be appropriate for the M&E evaluation model. Additionally, Westinghouse [

] Therefore, this issue was formed into RAI-SNPB-17.

Westinghouse responded to RAI-SNPB-17 by first stating that the information pertinent to this RAI was also contained in the RAI responses to RAI-SNPB-3 and RAI-SNPB-22. These responses were addressed in Sections 3.2.3.2 and 3.3.4.3 and used to support the conclusions that (Reference 1) the mass release rates have been adequately modeled, and (Reference 2) the loop flow split has been adequately modeled. The results of this analysis demonstrated that the calculation methodology (which includes the nodalization) was sufficient to capture flow out of the break and the loop flow split. Further, Westinghouse discussed the nodalization [

] Westinghouse provided a sensitivity analysis in which it further increased the number of nodes in the SG. This study demonstrated that further increase in the number of nodes did not change the predicted behavior in the SG.

Because Westinghouse is using a nodalization similar to that of ECCS with appropriate additions and because it has demonstrated that the predicted behavior of the SG is independent of the number of nodes being used, the NRC staff has determined that nodalization has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.2 Thermodynamic Conditions

> **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 (°F) and partial steam pressure are within one percent of that which would result from use of the 1967 ASME Steam Tables with appropriate interpolation.*
>
> ANS-56.4-1983, Sub-Section 3.2.4.2

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model uses the same steam tables as the ECCS evaluation model. However, Westinghouse did not provide any information as to what those steam tables were. Therefore, this issue was formed into RAI-SNPB-18.

Westinghouse responded to RAI-SNPB-18 by detailing which steam tables were used in the M&E evaluation model. Those tables were the 1968 and 1983 versions of the ASME Steam Tables, and the 1984 National Bureau of Standards/National Research Council (NBS/NRC) Steam Tables. Because these tables are not biased for analysis and have been previously approved, the NRC staff has determined that thermodynamic conditions have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.3 Flow Modeling

Flow Modeling
The following effects may be taken into account in the flow modeling: (1) temporal change of momentum, (2) momentum convection, (3) forces due to wall friction, (4) forces due to fluid pressure, (5) forces due to gravity, and (6) forces due to geometric head loss effects (for example, contractions, expansions, bends, and pump losses). The frictional losses in pipes and other components may be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum mass and energy release rates. If an uncertainty in a pressure loss exists, the pressure loss shall be conservatively minimized.
ANS-56.4-1983, Sub-Section 3.2.4.3

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model uses the same flow models as used in the ECCS evaluation model. However, Westinghouse did not confirm that the above phenomena are addressed in the ECCS evaluation model. Therefore, this issue was formed into RAI-SNPB-19. Westinghouse also stated the M&E evaluation model uses the same modeling for cold leg/accumulator condensation, downcomer condensation, and loop flow split as was used in the ECCS evaluation model, but did not confirm that these modeling options are appropriate for the M&E evaluation model. Therefore, these issues were formed into RAI-SNPB-20 (Cold Leg / Accumulator Condensation), RAI-SNPB-21 (Downcomer Condensation), and RAI-SNPB-22 (Loop Flow Split). Westinghouse also stated that it is ignoring any hot leg condensation as this will reduce the steam flow and thus decrease the calculated containment pressure. While the NRC staff agrees this is conservative for a containment pressure evaluation, it does not seem conservative for an NPSHa evaluation. Therefore, this issue was formed into RAI-SNPB-23.

Westinghouse responded to RAI-SNPB-19 by detailing how WC/T satisfies the recommendation of Regulatory Guide 1.157 (Reference 29). These recommendations are almost identical with the criterion defined above. During the audit (Reference 18), the NRC staff confirmed that WC/T is using equations common to subchannel analysis (as WC/T is a further refinement of COBRA-IIIC, a subchannel code). These equations do account for all six of the flow effects mentioned in the criteria. Westinghouse is modeling the friction factors (both single and two phase) using the methodology of its ECCS evaluation model, which was not tuned to provide a conservative prediction for PCT but generated to provide a best estimate of the friction factors. Further, Westinghouse is using a best estimate methodology to calculate the pressure drops, the same methodology used in its ECCS evaluation model. Westinghouse's ability to model an appropriate pressure drop is detailed in the discussion of RAI-SNPB-22 given below.

Westinghouse responded to RAI-SNPB-20 (cold leg / accumulator condensation) by describing an experiment that was used specifically to capture the behavior in the cold leg during injection of ECCS flow under post-accident conditions. The comparisons between the test data and WC/T demonstrate that the code does a good job of predicting the bulk fluid conditions during this time. Specifically, Westinghouse provided measured test data and WC/T predictions of the

fluid temperature in the cold leg downstream of the injection location. Because WC/T is able to predict the fluid temperature profiles in the cold leg downstream of ECCS injection, the NRC staff has determined that cold leg/accumulator condensation has been adequately modeled.

Westinghouse responded to RAI-SNPB-21 (Downcomer Condensation) by describing an experiment that was used specifically to demonstrate WC/T's ability to accurately calculate certain parameters which would be influenced by downcomer condensation. Westinghouse provided analysis which demonstrated good agreement between the code's prediction and the measured data for both temperature and pressure during cold leg injection. Because WC/T is able to predict the fluid temperature profiles in the cold leg downstream of ECCS injection and the pressures around the core during this time, the NRC staff has determined that downcomer condensation has been adequately modeled.

Westinghouse responded to RAI-SNPB-22 (Loop Flow Split (LFS)) by describing WC/T predictions of broken and intact loop flowrates compared to test data. WC/T is able to capture these flow rates with no perceivable bias. Similar comparisons are also provided in WCAP-12945-P-A (Reference 2) in Figures 14-2-29 and 14-2-30 for a different test. Again, WC/T is able to provide a reasonable prediction for these intact and broken loop flowrates. Because WC/T is able to predict the flow rates in the broken and intact loops compared to test data, the NRC staff has determined that the loop flow split has been adequately modeled.

Westinghouse responded to RAI-SNPB-23 (Hot leg condensation) by describing the NPSHa and EQ analysis. Westinghouse stated that NPSHa is only a concern when the operator transfers from the injection mode to cold leg recirculation mode. Because this occurs about an hour after the start of the LOCA, there is no condensation in the hot leg, as there is no source of cold fluid or metal in the hot legs to condense any steam coming from the core. Hot leg condensation does not become important until after the operator transfers from cold leg to hot leg recirculation, which occurs several hours after the start of the LOCA. This condensation would only act to reduce the amount of steam released to containment, and thus lower the containment pressure and temperature. Therefore, ignoring this condensation is conservative for EQ analysis as this would result in the need to qualify equipment at higher temperatures and pressures than would actually be seen in the event. Because hot leg condensation is not a heat transfer mechanism during the period of interest for NPSHa and ignoring it is conservative for EQ and containment pressure analysis, the NRC staff has determined that ignoring hot leg condensation is appropriate.

Because Westinghouse is mechanistically modeling the flow effects (including the estimation of pressure drops), has validated these models in the WC/T ECCS evaluation model (Reference 2), and has specifically demonstrated that this validation is applicable to the M&E evaluation model through the Cold Leg/Accumulator Condensation, Downcomer Condensation, and LFS, the NRC staff has determined that flow modeling has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.4 Pump Characteristics

> **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 test data or by comparison with a full dynamic pump model may be used.*
>
> ANS-56.4-1983, Sub-Section 3.2.4.4

In the initial submittal (Reference 1), Westinghouse stated that the M&E evaluation model uses the same pump model as used in the ECCS evaluation model. Further, Westinghouse stated that for the double-ended pump suction (DEPS) break, the flow through the pump in the broken loop reverses soon after the break and is assumed to have a locked rotor for the remainder of the transient which reduces the flow rate from the pump side of the break. However, Westinghouse did not demonstrate that the dynamic pump model used in the ECCS evaluation model is appropriate for use in the M&E evaluation model or how assuming a locked rotor results in a conservative prediction of the M&E released to containment. Therefore, this issue was formed into RAI-SNPB-24.

Westinghouse responded to RAI-SNPB-24 by confirming that, from the stand point of RCP modeling, in both ECCS and M&E analysis it is conservative to empty the RCS as soon as physically possible. Therefore, the RCP modeling was not changed from that approved in the ECCS evaluation model. Further, Westinghouse clarified its use of locked rotor following a flow reversal is not an assumption, but reflects a physical anti-reversal device that exists on the pump and prevents it from rotating in the reverse direction.

Because Westinghouse is using pump models from an approved ECCS RCP model and because the locked rotor is due to a physical design mechanism on the pumps, the NRC staff has determined that the pump characteristics have been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.5 Core Modeling

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 underpredict core-to-reactor coolant heat transfer. Initial core stored energy shall be maximized.

ANS-56.4-1983, Sub-Section 3.2.4.8

This criterion is fully addressed in Sub-Sections 3.2.1.2, "Initial Stored Energy," and 3.2.1.3, "Fission Heat," above and 3.3.7, "Modeling – Heat Transfer Correlations," below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.6 Modeling of Metal Walls

Modeling of Metal Walls

Heat transfer from metal walls to coolant shall be calculated so as not to underpredict the rate of heat transfer relative to experimental data or the solution of the one dimensional, time dependent heat conduction equation.

ANS-56.4-1983, Sub-Section 3.2.4.9

This criterion is fully addressed in Sub-Sections 3.2.1.7, "Reactor Internals Heat Transfer," above and 3.3.7, "Modeling – Heat Transfer Correlations," below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.7 Modeling of Auxiliary Flows

> **Modeling of Auxiliary Flows**
>
> *Flows from the safety injection tanks (PWR only) 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 (PWR only) 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.*
>
> ANS-56.4-1983, Sub-Section 3.2.4.10

In the initial submittal (Reference 1), Westinghouse stated that [

]

Additionally, Westinghouse discusses its modeling of the auxiliary feedwater system and extraction steam, but these topics are fully addressed in Sub-Section 3.3.1.10, "Auxiliary Feedwater System," above and are those topics are not readdressed here.

Because Westinghouse is modeling the safety injection flow rates directly and that modeling has been previously reviewed and approved for the ECCS evaluation model, the NRC staff has determined that the modeling of auxiliary flows has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.4.8 Post-blowdown Modeling

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 carryout 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.

ANS-56.4-1983, Sub-Section 3.2.4.11

This criterion is fully addressed in Sub-Sections 3.2.3.2, "Mass Release Rates," 3.2.5.1, "Core Flooding Rate," 3.2.5.2, "Liquid Entrainment," 3.2.5.3, "Steam Quenching during Reflood," 3.2.6.2, "Steam Quenching during Post-Reflood," 3.2.7.1, "Steam from Decay Heat," 3.2.7.2, "Remaining Decay Heat," above and 3.3.7, "Modeling – Heat Transfer Correlations," below. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.5 Modeling – Break Flow

ANS 56.4, Section 3.2.4, directs the reviewer to consider three criteria regarding the modeling of the break flow. Those review criteria are outlined in Table 18 and the specific criteria are given in the subsequent sub-sections.

Table 18: Modeling – Break Flow Review Categories

Sub-Section	
3.3.5.1	Break Sizes
3.3.5.2	Break Flow Model
3.3.5.3	ECCS Spillage

3.3.5.1 Break Sizes

Break Sizes

For RCS 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. Pipe break characteristics are defined in accordance with ANSI/ANS-58.2-1980. The break shall be defined by its location, type, and area.

ANS-56.4-1983, Sub-Section 3.2.4.5.1

This criterion is fully addressed in Sub-Sections 3.2.2.1, "Break Location and Type," and 3.2.2.2, "Break Size," above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.5.2 Break Flow Model

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.
ANS-56.4-1983, Sub-Section 3.2.4.5.2

This criterion is fully addressed in Sub-Section 3.2.3.2, "Mass Release Rates," above. Therefore, no further review is required. The NRC staff has concluded that this criterion has been satisfied.

3.3.5.3 ECCS Spillage

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. The assumptions shall be consistent with the primary containment pressure and temperature analysis in Section 4, Dry Primary Containment Pressure and Temperature Transient Analysis. In lieu of this, a sump temperature equal to the saturation temperature of water at the total primary containment pressure at the start of recirculation from the peak pressure analysis may be used.
ANS-56.4-1983, Sub-Section 3.2.4.5.3

In the initial submittal (Reference 1), Westinghouse stated break liquid and vapor flow rates are calculated based on the upstream RCS conditions at the break. For peak pressure calculations the vapor component is maximized by [ ] For minimum NPSHa, the liquid component is maximized by assuming [ ] which would result in higher water temperatures in the containment sump and a reduced net positive suction head available.

Because Westinghouse is correctly biasing the analysis to produce more steam for peak pressure calculations and hotter sump temperatures for NPSHa analysis, the NRC staff has determined that the ECCS spillage has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.6 Modeling – Primary Containment Backpressure

ANS 56.4, Section 3.2.4, directs the reviewer to consider two criteria regarding the modeling of the primary containment backpressure. Those review criteria are outlined in Table 19 and the specific criteria are given in the subsequent sub-sections.

Table 19: Modeling – Primary Containment Backpressure Review Categories

Sub-Section	
3.3.6.1	PWR Backpressure
3.3.6.2	Drywell Backpressure

3.3.6.1 PWR Backpressure

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 flow (steam binding) in the reactor coolant loop and, therefore, affects the rate of mass and energy release. The mass and energy release calculation shall be coupled to the primary containment pressure calculation or a conservatively high backpressure (constant or time dependent function) shall be used.

ANS-56.4-1983, Sub-Section 3.2.4.6.1

In the initial submittal (Reference 1), Westinghouse stated that the M&E release calculation performed by WC/T would be coupled with GOTHIC which would calculate the containment response. Additionally, Westinghouse stated that if the M&E release calculation would not be coupled with GOTHIC, a conservative containment back pressure would be used. However, the NRC staff was unclear on the details of this coupling and if WC/T used in standalone mode would result in a conservative analysis. Therefore, these issues were formed into RAI-SNPB-25 and RAI-SNPB-26. Further, Westinghouse was asked about the documentation of the approved GOTHIC model and to demonstrate that coupling to GOTHIC was not sensitive to the time step of either code. These issues were formed into RAI-SCVB-5 and RAI-SCVB-6.

Westinghouse responded to RAI-SNPB-25 by providing further details on how the M&E was passed from WC/T to GOTHIC. This response was clarified during the audit (Reference 18) and through a further supplement to the RAI (Reference 19). However, there was still some confusion by the NRC as to exactly how the coupling between WC/T and GOTHIC was occurring, as the statements in the TR seemed to suggest that some of the M&E release

predicted by WC/T would be ignored and not transferred to containment through GOTHIC. These statements seemed to be in direct conflict with the equations given in the TR which had no such "lost" M&E. Further, any "lost" M&E would have to be very small compared to the total M&E transfer, as verified by the various sensitivity studies which demonstrated that smaller GOTHIC time steps and biased input conditions (which should have greatly increased this "lost" M&E) had almost no impact. Upon further clarification by Westinghouse, it was discovered that a statement in the TR could be misleading, but it was confirmed that all of the M&E release predicted by WC/T would be transferred to GOTHIC in an appropriate time frame and that there was no "lost" M&E release.

The confusing statement in the TR is on page 3-44 and is as follows: [

] The NRC staff believed that the M&E release [
] would be "lost" because [
] Further clarification by Westinghouse revealed that this M&E would not be "lost" but would be accounted for in the [
]

Westinghouse responded to RAI-SNPB-26 by providing an analysis which demonstrated WC/T's sensitivity to changes in the initial back pressure compared to the back pressure calculated from the resulting M&Es out of WC/T. This analysis demonstrated that WC/T is relatively insensitive to even large changes in back pressure. Further, during the audit (Reference 18), Westinghouse confirmed that part of the process for using WC/T in standalone mode required that a conservative back pressure be used (which would depend on the figure of merit of the analysis) or an iterative process would be used until the containment back pressure input to WC/T converged with the resulting containment pressure calculated from the M&E release from WC/T. This is a limitation on the approval of the WC/T M&E evaluation model.

Westinghouse responded to RAI-SCVB-5 by clarifying which version of GOTHIC and that a number of plants have amended their licensing basis to include this version of GOTHIC.

Westinghouse responded to RAI-SCVB-6 by providing a sensitivity which demonstrates that the coupling of GOTHIC and WC/T is almost entirely insensitive to the time step difference between the two codes. The same containment pressures were calculated in situations when GOTHIC's time step was much greater and much less than the WC/T time step. While this study did demonstrate that it is appropriate to use GOTHIC up to some maximum time step, in this case [ ] seconds, it did not demonstrate that GOTHIC can be used for time steps greater than this length. Therefore, a limitation is given on the maximum time step of GOTHIC.

Because Westinghouse has demonstrated that WC/T can be coupled with GOTHIC though the explanation of how information is shared between the codes, through a sensitivity study which demonstrated that the containment pressure is insensitive to the difference in the time steps, and because Westinghouse is either mechanistically calculating the back pressure directly with GOTHIC, or is iterating on the M&E release with WC/T using an appropriate code to calculate the back pressure [ ] the NRC staff has determined that the PWR backpressure has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.6.2 Drywell Backpressure

Drywell Backpressure

For most of the initial blowdown phase, the break flow is critical, so drywell backpressure is unimportant. As the blowdown continues and the vessel pressure decreases, the break flow becomes subcritical, and drywell backpressure begins to -affect the mass and energy release rate. During the post-reflood phase, drywell backpressure is again unimportant, since the break flow rate is simply equal to the ECCS flow rate. Throughout the post-LOCA transient, the mass and energy release calculation shall be coupled to the drywell pressure calculation, or a conservatively low backpressure function shall be used.

ANS-56.4-1983, Sub-Section 3.2.4.6.2

This criterion is only applicable to boiling water reactors. As the methodology described in Westinghouse's submittal (Reference 1) is only for PWRs, this criterion does not apply. The NRC staff has concluded that this criterion does not apply.

3.3.7 Modeling – Heat Transfer Correlations

ANS 56.4, Section 3.2.4, directs the reviewer to consider six criteria regarding the modeling of heat transfer. Those review criteria are outlined in Table 20 and the specific criteria are given in the subsequent sub-sections.

Table 20: Modeling – Heat Transfer Correlations Review Categories

Sub-Section	
3.3.7.1	Core to Reactor Coolant Heat Transfer
3.3.7.2	Primary Metal to Reactor Coolant Heat Transfer
3.3.7.3	Steam Generator Tubes to Reactor Coolant
3.3.7.4	Steam Generator Coolant to Steam Generator Tubes
3.3.7.5	Steam Generator Metal to Steam Generator Coolant
3.3.7.6	Acceptable Heat Transfer Correlations

3.3.7.1 Core to Reactor Coolant Heat Transfer

> **Core to Reactor Coolant Heat Transfer**
>
> *Nucleate boiling may be assumed for heat transfer from the core to two-phase reactor coolant. Forced convection may be assumed for heat transfer from the core to single phase reactor coolant.*
>
> ANS-56.4-1983, Sub-Section 3.2.4.7.1

In the initial submittal (Reference 1), Westinghouse stated that it is using the same heat transfer correlations as used in the ECCS evaluation model. However, Westinghouse did not demonstrate that the heat transfer correlations used in the ECCS evaluation model would provide an adequate estimate of the heat transfer for the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-27.

Westinghouse responded to RAI-SNPB-27 by providing a detailed listing of all of the heat transfer correlations used in WC/T, including those which transfer the heat from the core to the reactor coolant. The correlations chosen were familiar to the NRC and would be expected to result in a reasonable estimate of the heat transfer from the core.

Because Westinghouse is mechanistically modeling the heat transfer from the core (including heat transfer to the two-phase and single phase region) using equations which have been previously validated in its ECCS evaluation model and which are commonly used for evaluating the core heat transfer, the NRC staff has determined that heat transfer from the core the reactor coolant has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.7.2 Primary Metal to Reactor Coolant Heat Transfer

> **Primary Metal to Reactor Coolant Heat Transfer**
>
> *Nucleate boiling may be assumed for heat transfer from primary metal to two phase reactor coolant. Forced convection may be assumed for heat transfer from the primary metal to single-phase reactor coolant.*
>
> ANS-56.4-1983, Sub-Section 3.2.4.7.2

In the initial submittal (Reference 1), Westinghouse stated that it is using the same heat transfer correlations as used in the ECCS evaluation model. However, Westinghouse did not demonstrate that the heat transfer correlations used in the ECCS evaluation model would provide an adequate estimate of the heat transfer for the M&E evaluation model. Therefore, this issue was formed into RAI-SNPB-27. Additionally, the NRC staff was unsure as to why heat transfer from primary and secondary metal to containment was not treated directly. Therefore, this issue was formed into RAI-SNPB-28. Additionally, the NRC staff was unsure of the definition of inactive metal and how it was treated. Therefore, this issue was formed into RAI-SNPB-29.

Westinghouse responded to RAI-SNPB-27 by providing a detailed listing of the heat transfer correlations used in the WC/T M&E analysis. The correlations chosen were familiar to the NRC and would be expected to result in a reasonable estimate of the heat transfer from the primary metal.

Westinghouse responded to RAI-SNPB-28 by stating that heat transfer from the primary metal directly to containment is not modeled. First, the primary metal is insulated, which greatly reduces the heat transfer, and second, this transfer would be very small in comparison to heat transferred through the break. Additionally, Westinghouse argues that by forcing all the heat to be transferred out of the break, produces additional steam, which is more conservative for M&E analysis. Modeling heat transfer directly from the primary metal to the containment is not a criterion. Further, the NRC staff agrees with Westinghouse's reasoning that it is more conservative to put more steam in the containment than to simply heat up the containment as the pressure limit is typically most limiting.

Westinghouse responded to RAI-SNPB-29 by first defining inactive metal as metal which is not in direct contact with water at the end of the blowdown phase of the large-break LOCA event. This metal included the vessel upper head and pressurizer metal, as well as metal in the upper regions of the SGs. As this metal is not in direct contact with the coolant, its heat is either conducted to other metal which is in direct contact with the coolant or its heat is directly transferred to the containment via natural convection and radiation. Each of these heat transfer mechanisms would be expected to be small when compared with the energy being transferred from the break (especially if there is insulation). Currently, [

] If Westinghouse chooses to model the heat transfer from the inactive metal to containment directly, further justification shall be provided which demonstrates that such heat transfer has been modeled appropriately.

Because Westinghouse is mechanistically modeling the heat transfer from the primary metal (including heat transfer to the two-phase and single phase region) using equations which have been previously validated in its ECCS evaluation model and which are commonly used for evaluating the primary metal heat transfer, because it is conservatively treating the heat transfer directly to containment, and because it is conservatively treating the inactive metal, the NRC staff has determined that heat transfer from primary surfaces has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.7.3 Steam Generator Tubes to Reactor Coolant

Steam Generator Tubes to Reactor Coolant
Nucleate boiling may be assumed for heat transfer from the steam generator tubes to two-phase reactor coolant. Forced convection may be assumed for heat transfer from the steam generator tubes to the single-phase reactor coolant.
ANS-56.4-1983, Sub-Section 3.2.4.7.3

In the initial submittal (Reference 1), Westinghouse stated that multiple changes were made to the heat transfer modeling in the SG tube heat transfer logic. These changes were made in response to comparison with FLECHT-SEASET data which demonstrated that WC/T was over-predicting the heat transfer from the secondary side to the primary coolant in the SG.

The first change was the adoption of Unal's correlation (Reference 26) which accounts for the shielding effect in regions of film boiling which are far from the vicinity of the critical heat flux (CHF) point. In this region, the steam is superheated and the liquid droplet is saturated. Therefore, the droplet will evaporate and that rate of evaporation is based on the temperature difference between the saturated droplet and the superheated steam. However, this understanding somewhat oversimplifies the physical process. The droplet will only evaporate if the steam it is in contact with is superheated. Because the entire droplet does not evaporate at once, the droplet will evaporate some of its liquid into steam and that steam surrounding the droplet will not be at the superheated temperatures, but at the saturation temperature. Thus, as the droplet evaporates some of its liquid, it generates a "saturated steam shield" which will impede further evaporation. Unal's correlation is meant to account for this shielding effect. In these cases, radiative heat transfer is generally ignored as the temperature difference between the droplet and the wall is not large enough to cause any appreciable amount of heat transfer. However, Westinghouse did not provide any documentation demonstrating any validation of Unal's correlation. Therefore, this issue was formed into RAI-SNPB-30.

Westinghouse responded to RAI-SNPB-30 by providing details on the validation of Unal's correlation. This validation demonstrated that, while Unal is mostly used inside of its application domain, some usage will require Unal to exceed its range in mass flux. During the audit (Reference 18), the correlation was examined and it was determined that the correlation's behavior at the higher mass fluxes would be consistent with expected behavior (i.e., the heat transfer to the droplets would increase as mass flux increases). However, as there is no validation for the Unal correlation at higher mass fluxes, Westinghouse clarified in a supplemental response (Reference 19) that it would only apply the Unal correlation in the range over which it has been validated. Outside of this validated range, the Lee-Ryley correlation will be used by reverting back to the "WC/T standard" SG tube heat transfer logic. The Lee-Ryley correlation has been previously reviewed and approved as part of the WC/T ECCS evaluation model and is used in other system codes (e.g., RELAP5/MOD3). Further, this correlation was used to predict the FLECHT data in the TR and produced predictions which were generally similar or more conservative than the test data. Finally, the form of the Lee-Ryley correlation is more physical than the highly empirical nature of Unal's correlation.

The second change was the adoption of Pasamehmetoglu's correlation (Reference 27) which adjusts the interfacial heat transfer for subcooled droplet flow. The temperature of subcooled droplets must increase to the saturation temperature before the droplets can evaporate. The rate at which the droplet temperature increases is therefore important in determining when the droplets reach saturation temperature. The sooner the droplet reaches saturation temperature, the more likely that droplet is to evaporate into steam. Westinghouse believed the previous method for calculating this heat transfer to the subcooled droplets was over-predicting the heat transfer, which resulted in calculating droplets reaching saturation temperatures earlier and thus evaporating quicker than was experimentally observed. Therefore, it implemented this correlation which corrected for the over-prediction of the heat transfer to the subcooled droplets.

The third change was the increase in the number of nodes in the SG's primary side tubes. This number of nodes was increased to capture the presence of an axial quench front. This change is addressed in Section 3.3.4.1 - Nodalization above.

In its analysis, Westinghouse stated that it used the Biasi CHF correlation to determine the conditions of rewet. However, the NRC staff is aware that the original reference for the Biasi correlation (Reference 28) did not demonstrate the correlations predictive capability over the entire application domain given. Therefore, this issue was formed into RAI-SNPB-31. Westinghouse responded to RAI-SNPB-31 by providing details on the application range of the Biasi correlation. To perform some of the analysis, Westinghouse will need to calculate the CHF outside of the range of the Biasi correlation and its analysis confirms that using the value the Biasi correlation calculates outside its range is non-physical and non-conservative for this use. Therefore, Westinghouse has proposed to use the correlation with certain limitations. For pressure, [

] Westinghouse demonstrated that this results in a conservative estimate of the CHF by comparison to Groeneveld look up tables (in this case, conservative is an over-prediction of CHF, as that will signal an earlier re-wet and higher heat transfers from the SG). For mass fluxes, [

] Therefore, Westinghouse proposed to use this conservative estimate of CHF. Westinghouse demonstrated that this results in a conservative estimate of the CHF by comparison to Groeneveld look up tables (in this case, conservative is an over-prediction of CHF, as that will signal an earlier re-wet and higher heat transfer from the SG).

To demonstrate these modeling changes were appropriate, Westinghouse provided validation using FLECHT-SEASET data. These experiments were used to simulate the heat release from a SG. For long term M&E analysis (i.e., cold leg and pump suction breaks), the heat release rate from the SGs is one of the most influential factors in determining the conditions in containment. If the heat release rate is under-predicted, the resulting analysis will not provide an adequate estimate of the peak containment temperatures and pressures. Therefore, it is important to ensure that the heat release rate from the SGs is a best estimate prediction or is conservatively over-predicted. While assuming that the steam leaving the SGs is superheated

would certainly result in a conservative prediction, FLECHT-SEASET results demonstrated that it is not mechanistic, as much of the steam leaving the FLECHT SG was not superheated, but becomes saturated as the event continues.

Westinghouse used the modeling options in its M&E evaluation model to model the FLECHT-SEASET experiments. From this validation exercise, it provides multiple plots which demonstrated that its M&E evaluation model was able to capture most of the important behavior from the FLECHT tests. WC/T consistently over-predicted the SG outlet vapor temperature and under-predicted the SG tube wall temperatures. Both of these are indications that the M&E evaluation model predicted a higher heat transfer from the SG than was observed in the FLECHT-SEASET experiments, which would be conservative for predicting the M&E release. However, Westinghouse consistently under-predicted the secondary heat release rate, which would result in a non-conservative prediction of M&E release. Therefore, this issue was formed into RAI-SNPB-32.

Westinghouse responded to RAI-SNPB-32 by providing a further description of the comparison to FLECHT data and by further reviewing the FLECHT data. Integrated SG heat release plots were provided in response to the RAI (Reference 16). [

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Given that the comparison of integrated heat release and instantaneous heat release rates were initially very similar, that many of differences between WC/T and FLECHT in these values can be demonstrated to be from anomalies in the experiment or reporting of the data, and that the measured SG tube temperatures were consistently underpredicted by WC/T, the NRC has found that WC/T does provide an appropriate prediction of FLECHT data for purposes of M&E release.

Because Westinghouse has appropriately limited the Biasi correlation in a manner which will produce a conservative value, is limiting the use of the Unal correlation to its appropriate range and using the Lee-Ryley correlation as an alternative, and has provided a validation in the comparison to FLECHT data which demonstrates that WC/T is appropriately modeling the heat transfer, the NRC staff has determined that heat transfer on the primary side of the SG has been appropriately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.7.4 Steam Generator Coolant to Steam Generator Tubes

Steam Generator Coolant to Steam Generator Tubes

Natural convection heat transfer may be assumed for heat transfer from the steam generator liquid to the steam generator tubes. Condensation heat transfer may be assumed for heat transfer from steam generator steam to the steam generator tubes.

ANS-56.4-1983, Sub-Section 3.2.4.7.4

In the initial submittal (Reference 1), Westinghouse did not specify the type of heat transfer used in the secondary side between the SG coolant and the SG tubes. Therefore, this issue was formed into RAI-SNPB-33.

Westinghouse responded to RAI-SNPB-33 by detailing the heat transfer that occurs in the SG. Prior to the LOCA, the SG acts as a heat sink as nucleate boiling transfers the heat from the primary side of the SG tubes to the secondary side fluid. Following a LOCA, the MSIVs close and the feedwater flow stops isolating the SG. As the primary side is depressurizing, the SG is turning from a heat sink to a heat source. As a heat source, heat is transferred from the secondary generator (shell and internal metal) to the secondary fluid via natural convection or nucleate boiling. This heat is then transferred from the secondary fluid to the SG tubes primarily through natural convection.

Because Westinghouse is explicitly modeling all modes of heat transfer (including single phase liquid natural convection, single phase liquid forced convection, nucleate boiling, critical heat flux, transition boiling, film boiling, and single phase vapor convection), the NRC staff has determined that the heat transfer between the SG coolant to the SG tubes has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.7.5 Steam Generator Metal to Steam Generator Coolant

Steam Generator Metal to Steam Generator Coolant

Natural convection heat transfer may be assumed for heat transfer from steam generator metal to steam generator liquid and steam.

ANS-56.4-1983, Sub-Section 3.2.4.7.5

In the initial submittal (Reference 1), Westinghouse did not specify the type of heat transfer used in the secondary side between the SG metal and the SG coolant. Therefore, this issue was formed into RAI-SNPB-33. Additionally, the NRC staff did not know if heat transfer from the primary and secondary metal to containment was treated directly. Therefore, this issue was formed into RAI-SNPB-28.

Westinghouse responded to RAI-SNPB-28 by stating that it does not model heat transfer from the secondary metal directly to containment. First, the secondary metal is insulated, which greatly reduces the heat transfer, and second, this transfer would be very small in comparison to heat transferred through the break. Additionally, Westinghouse argues that by forcing all the heat to be transferred out of the break, produces additional steam, which is more conservative for M&E analysis. Modeling heat transfer directly from the secondary metal to the containment is not a criterion. Further, the NRC staff agrees with Westinghouse's reasoning that it is more conservative to put more steam in the containment than to simply heat up the containment as the pressure limit is typically most limiting.

Westinghouse responded to RAI-SNPB-33 by providing additional details on the heat transfer models used on the secondary side. Westinghouse has correlations to model single phase liquid natural convection, single phase liquid forced convection, nucleate boiling, critical heat flux, film boiling, and single phase vapor convection. Westinghouse further confirmed that during the event, the primary heat transfer mode is nucleate boiling or natural convection from the shell to the secondary fluid and natural convection from the secondary fluid to the tubes.

Because Westinghouse is directly modeling the heat transfer from the shell to the secondary fluid and from the secondary fluid to the tubes using appropriate heat transfer models, the NRC staff has determined that the heat transfer between the SG metal to the SG coolant has been adequately modeled. The NRC staff has concluded that this criterion has been satisfied.

3.3.7.6 Acceptable Heat Transfer Correlations

Acceptable Heat Transfer Correlations

The following correlations are acceptable for use in the preceding regime descriptions:

(1) Single phase forced convection: Dittus and Boelter
(2) Nucleate Boiling: Jen and Lottes or Thom
(3) Natural Convection: function of Grashof and Prandtl
(4) Condensation: Film-type Condensation

ANS-56.4-1983, Sub-Section 3.2.4.7.6

In the initial submittal (Reference 1), Westinghouse stated that it is using the same heat transfer correlations as used in the ECCS evaluation model. However, Westinghouse did not provide a listing of what those heat transfer correlations were. Therefore, this issue was formed into RAI-SNPB-27.

Westinghouse responded to RAI-SNPB-27 by providing a detailed listing of the heat transfer correlations used in the WC/T M&E analysis.

Because Westinghouse is using correlations which are consistent with the previously approved WC/T ECCS evaluation model, the correlations discussed above, and those commonly used for heat transfer in nuclear reactor safety analysis, the NRC staff has determined that Westinghouse is using acceptable heat transfer correlations. The NRC staff has concluded that this criterion has been satisfied.

4.0 LIMITATIONS AND CONDITION

Based on the forgoing considerations, the NRC staff concludes that the use of the methodology described in WCAP-17721-P/NP, Revision 0, for calculating the M&E release from a large-break LOCA for a PWR is acceptable provided that the following limitations and condition are met:

Limitations

1. If Westinghouse chooses to model the heat transfer from the inactive metal to containment directly, further justification shall be provided which demonstrates that such heat transfer has been modeled appropriately.

2. WC/T may only be directly coupled with the GOTHIC computer code. For codes other than GOTHIC, WC/T must be used in a standalone mode.

3. The maximum time step of GOTHIC shall be limited to a time step which has been demonstrated to result in appropriate coupling between WC/T and GOTHIC. Currently, that time step is [ ] seconds, as per the sensitivity study performed in response to RAI-SCVB-6. If Westinghouse wishes to increase the maximum time step it will need to perform a similar sensitivity study at the new maximum time step.

4. Westinghouse shall run WC/T until the calculated peak pressure is decreasing. For large dry containments this is typical early, for ice condenser containments, this is typically after the ice bed melts, a secondary pressurization occurs, and the pressure decreasing again.

5. When used in standalone mode, WC/T shall be run with either a conservative back pressure or iteratively until the containment back pressure input to WC/T converges to the resulting containment pressure calculated from the M&E release from WC/T within an acceptable margin.

6. The use of WCAP-17721-P/NP, Revision 0, is limited to analyzing the M&E release following a large-break LOCA. If a different event is found to produce a more limiting containment condition (e.g., small break LOCA), using WC/T to simulate that event would require further NRC review.

7. The energy difference caused by any discrepancy between the material properties used in WC/T and those given in the ASME Boiler and Pressure Vessel Code shall be accounted for in the M&E analysis.

Condition

1. The NRC staff is aware that the currently approved version of Westinghouse's fuel performance code (PAD4) does not account for thermal-conductivity degradation. As of the writing of this SE, Westinghouse has submitted a newer version of PAD (PAD5) for the NRC review and approval, which does account for the thermal conductivity degradation. Additionally, Westinghouse has used an updated version of PAD4 (PAD4TCD) in a number of licensing actions. Therefore, Westinghouse is required to

use a fuel performance code which does account for fuel thermal conductivity degradation (such as PAD4TCD) for this analysis to ensure the initial stored energy is appropriate. If Westinghouse chooses to use PAD4TCD, then upon approval of a new fuel thermal mechanical code which does account for thermal conductivity degradation, Westinghouse shall confirm that the initial stored energy calculated using PAD4TCD remains accurate or conservative.

The NRC staff will require licensees referencing this TR in licensing applications to document how these limitations and condition are met.

5.0 CONCLUSION

When exercised appropriately, the NRC staff has reasonable assurance that the use of the M&E methodology, as documented in Reference 1, is acceptable in calculating the M&E release for PWRs following a large-break LOCA. The NRC staff has reviewed the methodology describing the use of WC/T, and does not intend to review the associated topical report when referenced in licensing evaluations, but only finds the methods applicable when exercised in accordance with the limitations and condition described in Section 4.0 of this SE.

If the NRC's criteria or regulations change such that its conclusions about the acceptability of the methods or analyses are invalidated, the organization referencing the TR (Reference 1) will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of these methodologies without revision of the respective documentation.

6.0 REFERENCES

1. Ofstun, R.P., Espinosa, R., Logan, C.P., Genuske, M.E., and Lukas, R.B., "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology", WCAP-17721-P/NP, Revision 0, April 2013 (ADAMS Accession No. ML13133A064).

2. WCAP-12945-P-A, "Code Quantification Document for Best Estimate LOCA Analysis" (all volumes), March 1998.

3. Holonich, J.J. (U.S. NRC) to Gresham, J.A. (W), "Acceptance for review of topical report WCAP-17721-P/NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology" (TAC No. MF1797), dated July 31, 2013 (ADAMS Accession No. ML13155A244).

4. Dean, J. (U.S. NRC) to Mendiola, A. (U.S. NRC) "WCAP-17721-P, Revision 0 and WCAP-17721-NP, Revision 0, - Audit for Comprehension Plan Final" (TAC No. MF1797), dated May 28, 2014 (ADAMS Accession No. ML14139A399 (Non-Publicly Available)).

5. Ofstun, R., Smith, L., BE-2004 Proceedings, International Meeting on Updates in Best Estimate Methods in Nuclear Installation Safety Analysis, "PIRT for Large Break LOCA Mass and Energy Release Calculations," November 2004. (ADAMS Accession No. ML14192A024).

6. Gresham, J.A. (W) to U.S. NRC "Submittal of Presentation" WCAP-17721-P/NP, Revision 0, WCOBRA/TRAC LOCA Mass and Energy Release Methodology Regulatory Guide 1.203 Compliance" (Proprietary)" (TAC No. MF1797), LTR-NRC-14-40 P-Attachment, dated July 3, 2014 (ADAMS Accession No. ML14192A025 (Non-Publicly Available)).

7. Dean, J. (U.S. NRC) to Mendiola, A. (U.S. NRC), "Audit Summary for Comprehension Plan (WCAP-17721-P/NP, Revision 0)" (TAC No. MF1797), dated August 14, 2014 (ADAMS Accession No. ML14226A848 (Non-Publicly Available)).

8. Lenning, E. (U.S. NRC) to Gresham, J.A. (W), "Request for Additional Information (RAI) Re: Westinghouse Electric Company Topical Report WCAP-17721-P/NP, Revision 0, "Westinghouse Containment Analysis Methodology - PWR (Pressurized Water Reactor) LOCA (Loss-Of-Coolant Accident) Mass and Energy Release Calculation Methodology," - Set 1 (Confirmatory Analysis) (TAC No. MF1797)" September 4, 2014 (ADAMS Accession No. ML14240A017 (Publicly Available)).

9. Gresham, J.A. (W) to U.S. NRC, "Submittal of "WCAP-17721-P/NP, Revision 0, NRC Confirmatory Study Inputs - Response to RAIs" (Non-Proprietary)" (TAC No. MF1797), LTR-NRC-14-64, dated September 25, 2014. (ADAMS Accession No. ML14272A095).

10. Lenning, K. (U.S. NRC) to Gresham, J.A. (W), "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-17721-P/NP, Revision 0, "Westinghouse Containment Analysis Methodology - PWR (Pressurized Water Reactor) LOCA (Loss-Of-Coolant Accident) Mass and Energy Release Calculation Methodology," - Set 2 (Safety and Code Review Branch) (TAC No. MF1797), October 20, 2014 (ADAMS Accession No. ML14254A251).

11. Lenning, K. (U.S. NRC) to Gresham, J.A. (W), "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-17721-P/NP, Revision 0, "Westinghouse Containment Analysis Methodology - PWR (Pressurized Water Reactor) LOCA (Loss-Of-Coolant Accident) Mass and Energy Release Calculation Methodology" - Set 3 (Containment and Ventilation Branch) (TAC No. MF1797), October 20, 2014 (ADAMS Accession No. ML14254A260).

12. Denning, R. (U.S. NRC) to Mendiola, A. (NRC), "Transmittal of input to the audit report for the October 7 and 8, 2014, audit to review calculations supporting WCAP-17721-P/NP, Revision 0, Topical Report review" (TAC No. MF1797), dated October 27, 2014 (ADAMS Accession No. ML14295A283 (Non-Publicly Available)).

13. Gresham, J.A. (W) to U.S. NRC, "Submittal of"WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch, and Set 3, Containment and Ventilation Branch - Response to Selected RAIs" (Proprietary/Non- Proprietary)." (TAC No. MF1797), LTR-NRC-15-5, dated January 22, 2014 (ADAMS Accession No. ML15027A157).

14. Gresham, J.A. (W) to U.S. NRC, "LTR-NRC-15-18 P-Attachment," WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch – Response to Selected RAIs" (TAC No. MF1797), LTR-NRC-15-18, dated March 10, 2015 (ADAMS Accession No. ML15071A027).

15. Gresham, J.A. (W) to U.S. NRC, "Submittal of "WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch, and Set 3, Containment and Ventilation Branch - Response to Selected RAIs" (Proprietary/Non-Proprietary)." (TAC No. MF1797), LTR-NRC-15-19, dated March 16, 2015 (ADAMS Accession No. ML15078A055).

16. Gresham, J.A. (W) to U.S. NRC, "Submittal of "WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch - Response to Selected RAIs" (Proprietary/Non-Proprietary) (TAC No. MF1797), LTR-NRC-15-20, dated March 26, 2015 (ADAMS Accession No. ML15092A443).

17. Gresham, J.A. (W) to U.S. NRC, "WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch - RAI 2.32 Response Supplement" (Proprietary/Non-Proprietary) (TAC No. MF1797), LTR-NRC-15-33, dated April 22, 2015 (ADAMS Accession No. ML15126A280).

18. Dean, J. (U.S. NRC) to Mendiola, A. (U.S. NRC), "Audit Summary of the Audit conducted between April 8, 2015 – April 16, 2015 – RAI audit for WCAP-17721-P/NP, Revision 0, – Westinghouse containment analysis methodology – PWR LOCA Mass and Energy Release Calculation Methodology (TAC NO. MF1797)" (TAC No. MF1797), dated May 20, 2015 (ADAMS Accession No. ML15140A158 (Non-Publicly Available)).

19. Gresham, J.A. (W) to U.S. NRC, "Submittal of WCAP-17721-P/NP, Revision 0, RAI Response Supplements" (Proprietary/Non-Proprietary) (TAC No. MF1797), LTR-NRC-15-39 dated May 21, 2015 (ADAMS Accession No. ML15147A335).

20. Lenning, E. (U.S. NRC) to Gresham, J.A. (W), "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-17721-P/NP, Revision 0, 'Westinghouse Containment Analysis Methodology - PWR (Pressurized Water Reactor) LOCA (Loss-Of-Coolant Accident) Mass and Energy Release Calculation Methodology'" (TAC No. MF1797), June 11, 2015 (ADAMS Accession No. ML15148B251).

21. Gresham, J.A. (W) to U.S. NRC, "Submittal of 'WCAP-17721-P/NP, Version 0, - Response to Draft RAI' (Non-Proprietary)." (TAC NO. MF1797), LTR-NRC-15-42 dated May 26, 2015 (ADAMS Accession No. ML15152A158).

22. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987 (Certain updated sections are available from the NRC).

23. Section 6.2.1, "Containment Functional Design (LOCAs)" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 3, USNRC, March 2007 (ADAMS Accession No. ML070220505).

24. Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant-Accidents (LOCAs)" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 3, USNRC, March 2007 (ADAMS Accession No. ML053550265).

25. ANS 56.4, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments", American Nuclear Society, December 1983.

26. Unal, C., Tuzla, K., Cokmez-Tuzla, A.F., and Chen, J.C., "Vapor Generation Rate Model for Dispersed Droplet Flow," Nuclear Engineering and Design, Vol. 125, pp 161-173, 1991.

27. Pasamehmetoglu, K.O., "Transient Direct-Contact Condensation on Liquid Droplets," Nonequilibrium Transport Phenomena, Vol. 77, PP. 47-56, ASME HTD, New York, 1987.

28. Biasi, L., Clerici, G.C., Garribba, S., Sala, R., and Tozzi, A. (1967), "Studies on Burnout: Part 3 - A New Model for Round Ducts and Uniform Heating and Its Comparison with World Data", Energia Nucleare 14: 530-536.

29. NRC, Best Estimate Calculation of Emergency Core Cooling System Performance, Regulatory Guide 1.157. Washington D.C., U.S. Nuclear Regulatory Commission, 1989.

30. ASME Boiler and Pressure Vessel Code, Section II, Part D, "Material Properties," The American Society of Mechanical Engineers, New York, New York, 2004.

7.0 LIST OF ACRONYMS

ABB	ASEA Brown Boveri
ASEA	Allmänna Svenska Elektriska Aktiebolage
ASME	American Society of Mechanical Engineers
BPVC	ASME Boiler and Pressure Vessel Code
CE	Combustion Engineering
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EQ	Environment Qualification
GDC	General Design Criteria
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
M&E	Mass and Energy
NPSHa	Net Positive Suction Head available
NRC	U. S. Nuclear Regulatory Commission
PIRT	Phenomenon Identification and Ranking Table
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
SE	Safety Evaluation
SRP	Standard Review Plan
WC/T	WCOBRA/TRAC

Attachment 1: Resolution of Comments

Principal Contributors: J. Kaizer
S. Peng

Date: August 24, 2015

A. RAI SUMMARY

An RAI categorization system was in order to improve communication between the NRC technical staff, the public, an applicant, and NRC management, during the RAI process. This system categorizes each RAI by the NRC staff's perceived *level of disagreement* with a statement in the submittal. Six levels of disagreement (1 = highest, 6 = lowest) are used and are detailed in Table 1 below.

Table 21: RAI Categories

Level of Disagreement	Description	Definition
1	Dispute (RAI-DIS)	*Dispute* RAIs are asked when the staff understands the information, but believes the information is incorrect.
2	Skepticism (RAI-SK)	*Skeptical* RAIs are asked when the staff understands the information, but is unsure if the information is correct.
3	Lacking Documentation (RAI-L)	*Lacking Documentation* RAIs are asked when the staff understands the information, but considers the supporting documentation to be inadequate.
4	Clarification Necessary (RAI-C)	*Clarification Necessary* RAIs are asked when the staff does not understand the information; therefore clarification is needed to determine the staff's level of disagreement or editorial concerns that may change the meaning of the statement.
5	Basic Information (RAI-B)	*Basic Information* RAIs are focused on obtaining some basic information not provided in the submittal.
6	Editorial Change (RAI-E)	*Editorial Change* RAIs are focused an editorial change that does not change the meaning of the statement.

Note: this categorization system was only used with the Confirmatory and SNPB RAIs.

Each RAI is formatted in the following manner:

<table>
<tr><td colspan="2">RAI Title

Exact wording of the RAI
RAI Title used in Reference (Reference containing the RAI)</td></tr>
<tr><td>Disagreement Level</td><td>RAI Level of Disagreement</td></tr>
<tr><td>SE Section</td><td>Section of the Safety Evaluation which relies on responses of this RAI.</td></tr>
<tr><td>Comment</td><td>Additional comments by the NRC staff.</td></tr>
<tr><td>Reference of Response</td><td>RAI Title used in Response to RAI (Reference containing the RAI response)</td></tr>
</table>

This section contains the evaluation of the RAI. This evaluation is not a technical evaluation of the material presented in the RAI, as that evaluation is in the body of the SE in the section defined in the table above (SE section). This evaluation is focused on if the RAI satisfied the NRC staff's request for information.

A.1 Confirmatory RAIs

SRP Chapter 6.2.1.3, Section III, states the following:

The reviewer may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

The NRC staff is performing such a confirmatory analysis and makes the following requests for additional information in support of that analysis. For the cold leg and pump suction leg break simulations, provide the following:

A.1.1 RAI-Confirmatory-1 – SI Mixing

SI Mixing	
Specify if the broken loop safety injection water goes directly to the sump pool, or mixes with the containment atmosphere and condenses vapor. If it mixes provide details on the mixing such as what percentage mixes and the basis for that value. RAI-1 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #1 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.2 RAI-Confirmatory-2 – SI Temperatures

SI Temperatures	
The initial SI (RWST) and accumulator water temperature. (2 values) RAI-2 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #2 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.3 RAI-Confirmatory-3 – RWST Volume

RWST Volume	
Specify the water volume held in the RWST at the beginning of the event. (1 value) RAI-3 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #3 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.4 RAI-Confirmatory-4 – Switchover

Switchover	
During switchover, is any period of no injection assumed? If so, how long is that period of no injection? RAI-4 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #4 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.5 RAI-Confirmatory-5 – Break Flow Rates

Break Flow Rates	
Provide plots of the liquid and vapor mass flow rates and liquid and vapor temperatures from both sides of the break as a function of time. (8 plots) RAI-5 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #5 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.6 RAI-Confirmatory-6 – SG Heat Transfer

SG Heat Transfer	
Provide a plot of the Steam Generator (SG) tube heat transfer power to the primary system for the broken and intact loops, include decay heat. (5 plots) RAI-6 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #6 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.7 RAI-Confirmatory-7 – SG Pressure

SG Pressure	
Provide plots of the SG pressure for the broken and intact loops. (4 plots) RAI-7 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #7 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.1.8 RAI-Confirmatory-8 – SG Temperatures

SG Temperatures	
Provide plots of the SG wall temperatures at 1, 4 and 10 foot elevations for the broken and intact loops as a function of time. (12 plots) RAI-8 (Reference 8)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.2.8.1
Comment	Information needed as input to the confirmatory analysis.
Reference of Response	1. RAI #8 (Reference 9)

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2 RAIs from the Nuclear Performance and Code Review Branch (SNPB)

A.2.1 RAI-SNPB-1 – Downcomer Stored Energy Release

Downcomer Stored Energy Release	
Demonstrate that method for modeling the downcomer stored energy release in WC/T is appropriate for the M&E evaluation model such that the mass and energy release is adequately predicted. 2.1 RAI-3 – Downcomer stored energy release (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.1.7
Comment	In Section 2.11 of their initial submittal (Reference 1), Westinghouse stated that the same downcomer stored energy release model was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Downcomer Stored Energy Release (Reference 13)

Westinghouse provided analysis which discussed the impact of downcomer stored energy release on M&E analysis, but also on PCT analysis. Westinghouse discussed how over-predicting the downcomer energy release is not only conservative for M&E analysis by causing excess steam and hence a higher M&E release rate, but also conservative for PCT analysis. Additionally, it provided comparisons of WCOBRA/TRAC (WC/T) predictions and measured data on various parameters that would be impacted by the downcomer stored energy release rate. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.2 RAI-SNPB-2 – Break Size

Break Size	
Westinghouse stated that the break size used for the M&E evaluation model is the double ended break. Provide information on the consideration of slot breaks. If the breaks are considered, when are they used? If the breaks are not considered, what is the justification for ignoring them? 2.2 RAI-3 – Break size (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.2.2
Comment	In table 4-1, row 9 of their initial submittal (Reference 1), Westinghouse stated that their previous M&E evaluation model used a slot break to maximize M&E release in the CE NSSS designs. For the proposed EM, they did not specify if they considered slot breaks.
Reference of Response	1. 2.2 RAI-3 – Break Size (Reference 15)

In its response, Westinghouse provided details of a sensitivity study performed with slots breaks of three different sizes. This study demonstrated that a double ended break resulted in higher containment pressures than a similar slot break. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.3 RAI-SNPB-3 – Break flow model

Break flow model	
Demonstrate that break flow model used in WC/T provides an appropriate prediction of the break flow for the M&E evaluation model such that the mass and energy release is adequately predicted. 2.3 RAI-3 – Refill (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.3.2 and 3.3.4.1
Comment	In table 4-1, row 12 of their initial submittal (Reference 1), Westinghouse stated that the same break flow model was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Break Flow (Reference 13)

In its responses, Westinghouse verified that the break flow model was consistent over its application domain and did not systematically under or over-predict the break flow, **[**

] Westinghouse further detailed the method by which it makes the break flow prediction conservative for M&E cases by **[**

] Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.4 RAI-SNPB-4 – Refill

Refill	
Describe the validation data which supports WC/T ability to model the refill phase and demonstrate that this data justifies WC/T ability to predict the RCS transient response during the refill phase for the M&E evaluation model. 2.4 RAI-3 – Refill (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.4.1
Comment	In table 4-1 row 15 of their initial submittal (Reference 1), Westinghouse stated ECCS evaluation model had been validated for refill calculations by comparison with experimental data. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. 2.4 RAI-3 – Refill (Reference 15)

Westinghouse provided a summary of the validation which supported the refill phase. This summary describes WC/T's tendency to over predict the ECC bypass which results in a short delay in the start of refill. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.5 RAI-SNPB-5 – Core Flooding

Core Flooding	
Describe the validation data which supports WC/T ability to model the core flooding rate and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model. 2.5 RAI-3 – Core flooding (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.1
Comment	In table 4-1 row 16 of their initial submittal (Reference 1), Westinghouse stated ECCS evaluation model had been validated for the core flooding rate by comparison with experimental data. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Reflood Heat Transfer (Reference 13)

In its responses, Westinghouse analyzed multiple reflood tests performed to support the reflood heat transfer modeling for WC/T. It demonstrated that WC/T consistently **[** **]** Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.6 RAI-SNPB-6 – Liquid Entrainment

Liquid entrainment	
Describe the validation data which supports WC/T ability to model liquid entrainment and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model. 2.6 RAI-3 – Liquid Entrainment (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.2
Comment	In table 4-1 row 16 of their initial submittal (Reference 1), Westinghouse stated ECCS evaluation model had been validated for liquid entrainment by comparison with experimental data. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Reflood Heat Transfer (Reference 13)

In its responses, Westinghouse analyzed multiple reflood tests performed to support the reflood heat transfer modeling for WC/T. Westinghouse demonstrated that WC/T **[**

] Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.7 RAI-SNPB-7 – Upper plenum entrainment

Upper plenum entrainment	
Demonstrate that method for modeling the upper plenum entrainment/de-entrainment and condensation in WC/T is appropriate for the M&E evaluation model such that the mass and energy release is adequately predicted. 2.7 RAI-3 – Upper plenum entrainment (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.2
Comment	In Section 2.5 of their initial submittal (Reference 1), Westinghouse stated that the same upper plenum entrainment/de-entrainment and condensation model was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Upper Plenum Entrainment/De-Entrainment and Condensation (Reference 13) 2. 2.7 RAI-3 – Upper plenum entrainment (Reference 14)

Initially, Westinghouse provided an analysis of the predicted level in the upper plenum. While this analysis did provide some justification, it was not enough for the NRC to achieve reasonable assurance. Westinghouse provided three additional analyses, each of which demonstrated that the upper plenum entrainment was appropriately or conservatively predicted. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.8 RAI-SNPB-8 – Hot leg entrainment

Hot leg entrainment	
The justification for the hot leg entrainment/de-entrainment being independent of the pressure seems to suggest that all entrainment/de-entrainment modeling is independent of the final pressure calculation as the RCS steam temperatures will match those on the secondary side within minutes after event initiation. However, this concept seems to be in contradiction with the M&E PIRT which has entrainment and de-entrainment as high ranked phenomena as well as the other changes to the M&E model to better model the heat transfer from the secondary side to the primary side in the steam generators. Provide further clarification on this topic. 2.8 RAI-3 – Hot leg entrainment (Reference 10)	
Disagreement Level	Level 2 – Skepticism
SE Section	3.2.5.2
Comment	In Section 2.7 of their initial submittal (Reference 1), Westinghouse stated that the sensitivity study performed which varied the slip in the hot leg demonstrated that the mass and energy release (i.e., peak pressure) was relatively insensitive to the hot leg entrainment/de-entrainment. This was verified through a sensitivity which varied the slip ratio in the hot leg.
Reference of Response	1. SCVB-RAI-11, Hot Leg Entrainment/De-Entrainment (Reference 13)

Westinghouse provided clarification that discussed a sensitivity study performed on the flow conditions in the hot leg. This study demonstrated that, while ranked highly in the PIRT, **[**

] Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.9 RAI-SNPB-9 – Steam quenching

Steam quenching	
Describe the validation data which supports WC/T ability to model steam quenching and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model. Both the steam quenching during reflood and post-reflood should be considered. 2.9 RAI-3 – Steam quenching (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.3 and 3.2.6.2
Comment	In table 4-1 row 18 of their initial submittal (Reference 1), Westinghouse stated ECCS evaluation model had been validated for steam quenching by comparison with experimental data. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Cold Leg/Accumulator Condensation (Reference 13)

Westinghouse provided details on an analysis performed comparing experimental data to WC/T predictions for quenching in the cold leg. The analysis provided comparisons of the measured fluid temperatures and those predicted by WC/T. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.10 RAI-SNPB-10 – EQ and NPSHa

<table>
<tr><td colspan="2">EQ and NPSHa

Provide an explanation of the methodology for EQ and NPSHa analysis. With this methodology, define the acceptance criteria which are used, how those criteria are demonstrated to be met. Provide this explanation for each of the three containment types (large dry, sub-atmospheric, and ice-condenser). Additionally, address the relevant phases of each methodology, including the post-reflood phase and the decay heat phase. Also address the determination of the single active failure for both types of analyses.

2.10 RAI-3 – EQ and NPSHa (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.2.6.2</td></tr>
<tr><td>Comment</td><td>In table 4-1 row 20 their initial submittal (Reference 1), Westinghouse stated that it would assume [ ] during the long-term containment pressure and temperature analysis for EQ and [ ] for minimum NPSHa analysis. However, Westinghouse did not provide an explanation of the methodology for EQ or NPSHa analysis, what acceptance criteria were used, and how those criteria were demonstrated to be met.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.10 RAI-3 – Equipment Qualification (EQ) and Net Positive Suction Head Analysis (Reference 14)</td></tr>
</table>

Westinghouse provided a detailed description of its process of performing EQ and NPSHa analysis from an M&E perspective. Westinghouse detailed how WC/T would be used in each case, and it is in the same manner in which it is used for predicting the peak containment pressure. The difference in each analysis is that EQ analysis will extended the long term cooling calculation past the time when pressures start to decrease (up to 30 days) and NSPHa analysis will [ ] Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.11 RAI-SNPB-11 – Long term boil-off

Long term boil-off	
Describe how the steam-water mixing is calculated in this long-term boil off calculation. 2.11 RAI-3 – Long term boil-off (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.7.1
Comment	In table 4-1 row 22 of their initial submittal (Reference 1), Westinghouse discussed the long-term phases of the event, but the definitions of each phase were not entirely clear. Additionally, some additional phases were discussed, but not defined. Also, further documentation was needed to clarify the differences between the event itself and how that event was simulated. During an audit at Westinghouse, the information requested above was discussed and the NRC staff believed the information helped to provide a clearer understanding of the event and how the event was simulated.
Reference of Response	1. 2.11 RAI-3 – Long term boil-off (Reference 14)

In its response, Westinghouse provided a discussion focusing on the large dry/sub-atmospheric and ice condenser. For the large dry/sub-atmospheric, the energy transfer rate from the SG secondary metal, SG secondary fluid, and RCS metal [ ] Further, the safety injection which is not boiled is assumed to [ ] This analysis is non-mechanistic as it does not credit [ ] and calculates a conservatively high energy transfer rate. For the ice condenser containment, Westinghouse [ ] the ice condenser analysis is being performed in the same manner as the large dry/sub-atmospheric. [ ] Westinghouse ensured that its [ ] was conservative by comparing it with a WC/T analysis. This comparison demonstrated that the [ ] was somewhat mechanistic, but ultimately conservative compared to the WC/T results over the same [ ] Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.12 RAI-SNPB-12 – Event definitions

Event definitions
Provide a table which contains the following: 1. *The phase of the event (e.g., Blowdown, Refill, Reflood)* 2. *The conditions which define the beginning of that phase.* 3. *The conditions which define the end of that phase* 4. *An approximate duration of that phase (in seconds)* 5. *An approximate starting time of that phase (in seconds – with 0 being the event initiation)* 6. *A description of how the phase is simulated (e.g., mechanistically in WC/T, conservatively using certain approximations)* *Additionally, provide a second table which contains a description of the energy sources which impact each of the phases listed in the above table:* 1. *List each major energy source. The sources of energy should include, but not be limited to: Initial stored energy in the fuel, primary water, water in the broken loop SG, water in the intact SGs, primary metal, metal in the broken loop SG, metal in the intact loop SGs, decay heat.* 2. *The approximate initial energy of that energy source at the beginning of the event (in kW).* 3. *The approximate amount of energy which is released during phase 1 (include both kW and %)* 4. *The approximate amount of energy which is released during phase 2 (include both kW and %)* *The approximate amount of energy which is released during every other phase of the event (include both kW and %)* 2.12 RAI-3 – Event definitions (Reference 10)

Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.7.2
Comment	In the initial submittal (Reference 1), Westinghouse discussed the different phases of the event, but the definitions of each phase were not entirely clear. Additionally, some additional phases were discussed, but not defined. Also, further documentation was needed to clarify the differences between the event itself and how that event was simulated. During an audit at Westinghouse, the information requested above was discussed and the NRC staff believed the information helped to provide a clearer understanding of the event and how the event was simulated.
Reference of Response	1. 2.12 RAI-3 – Event Definitions (Reference 14)

In its response, Westinghouse provided the requested table. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.13 RAI-SNPB-13 – Main feedwater

Main feedwater	
Provide an estimate of the additional energy which the inclusion of main feedwater flow would add to the secondary side of the steam generator and demonstrate that including this additional energy is negligible compared to the total energy already stored in the steam generator. 2.13 RAI-3 – Main feedwater (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.1.9
Comment	In table 4-2 row 9 of their submittal (Reference 1), Westinghouse discussed how the main feedwater flow would be ignored in the modeling of the event. Main feedwater flow is relatively hot and will increase the energy stored in the steam generators, which will also increase the mass and energy released to containment and could increase the peak containment pressure and temperature. Therefore, ANS 56.4 suggests that this flow should be considered during analysis. Westinghouse stated that it did not need to consider this flow for their analysis as the additional energy was negligible, but did not provide any quantitative analysis.
Reference of Response	1. 2.13 RAI-2 – Main feedwater (Reference 14)

In its response, Westinghouse provided results of sensitivity studies which were performed to determine the energy added if main feedwater was included. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.14 RAI-SNPB-14 – Auxiliary feedwater

Auxiliary feedwater	
Clarify the modeling of the auxiliary feedwater and extraction steam. If both of these systems are being modeled in the M&E evaluation model, justify the modeling of both of these systems when the modeling of the main feedwater has been deemed negligible. 2.14 RAI-2 – Auxiliary feedwater (Reference 10)	
Disagreement Level	Level 2 – Skepticism
SE Section	3.3.1.10
Comment	In table 4-2 row 10 of their submittal (Reference 1), Westinghouse discussed how the auxiliary feedwater flow would be modeled in the event. Auxiliary feedwater flow is relatively cool and will decrease the energy stored in the steam generators, as will extraction steam. In turn, this could decrease the calculated mass and energy released to containment which would decrease the calculated peak containment pressure and temperature. While modeling of these system can be appropriate, the NRC staff questioned the validity of modeling extraction steam and auxiliary feedwater (which would reduce the mass and energy released to containment) but ignoring main feedwater flow (which would increase the mass and energy released to containment).
Reference of Response	1. 2.14 RAI-2 – Auxiliary Feedwater (Reference 14)

In its response, Westinghouse clarified its position. Westinghouse has provided this clarification; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.15 RAI-SNPB-15 – Steady state steam generator pressure

Steady state steam generator pressure	
Justify the use of the steam generator pressure calculated from the steady state calculation. Is this initial pressure always greater than or equal to the initial measured pressure in the steam generator plus uncertainty? If not, provide justification for using a pressure below the steam generator pressure plus uncertainty. 2.15 RAI-3 – Steady state steam generator pressure (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.2.5
Comment	In table 4-2 row 17 of their initial submittal (Reference 1), Westinghouse discussed how the steam generator pressure was calculated from the steady state calculation, but did not confirm that they will ensure this calculated value would be greater than or equal to the expected value plus uncertainty.
Reference of Response	1. 2.15 RAI-3 – Steady state steam generator pressure (Reference 15)

Westinghouse responded that the steady state SG pressure reflects a SG pressure **[ ]** This assumption increases the heat transfer which occurs during the steady state initialization and results in a higher SG secondary side pressure. Further, during the audit (Reference 18), Westinghouse added that the power used during the steady state run corresponds to the **[ ]** Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.16 RAI-SNPB-16 – SI water and temperature

SI water and temperature	
Are measurement uncertainties considered for the values of the initial safety injection tank water volume and water temperature? 2.16 RAI-3 – Auxiliary feedwater (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.2.9
Comment	In table 4-2 row 21 of their initial submittal (Reference 1), Westinghouse stated that measurement uncertainties were considered in the modeling of the accumulator pressure, but did not state whether measurement uncertainties were considered in the model of the water volume and temperature in the accumulator.
Reference of Response	1. 2.16 RAI-3 – Safety Injection (SI) water volume and temperature (Reference 14)

In its response, Westinghouse provided clarification on which values are chosen for the accumulator and refueling water storage tank temperature and volumes. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.17 RAI-SNPB-17 – Nodalization

<table>
<tr><td colspan="2">Nodalization

Provide justification which demonstrates that the nodalization used in WC/T results in appropriate predictions of the break flow and flow in the broken and intact loops such that the resulting predictions of mass and energy release will result in appropriate calculations of containment temperature and pressure. Additionally, provide a sensitivity study which demonstrates that the noding sensitivity in the steam generator.

2.17 RAI-3 – Nodalization (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.3.4.1</td></tr>
<tr><td>Comment</td><td>In table 4-2 row 25 of their initial submittal (Reference 1), Westinghouse stated that the same nodalization was used for the ECCS evaluation model as was used in the M&E evaluation model. However, in Section 2.8 of their submittal, Westinghouse stated that the noding was increased to account for physical phenomena. However, there is no data which demonstrates that the solution is not sensitive to the noding chosen and a further increase in noding may be needed.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.17 RAI-3 – Nodalization (Reference 14)</td></tr>
</table>

In its response, Westinghouse referenced its response on break flow modeling and loop flow split, as well as provided the requested sensitivity study. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.18 RAI-SNPB-18 – Steam tables

Steam tables	
Which steam tables are used in the M&E evaluation model? Are those steam tables consistent with the 1967 ASME Steam Tables? 2.18 RAI-5 – Steam tables (Reference 10)	
Disagreement Level	Level 5 - Basic Information
SE Section	3.3.4.2
Comment	In table 4-2 row 26 of their initial submittal (Reference 1), Westinghouse stated that the same steam tables were used for the ECCS evaluation model as was used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. 2.18 RAI-5 - Steam tables (Reference 15)

Westinghouse provided the details on which steam tables were used and confirmed that the steam tables are not biased, but are used as best estimate. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.19 RAI-SNPB-19 – Flow modeling

<table>
<tr><td colspan="2">Flow modeling

Confirm that the following effects have been taken into account in the flow modeling used in the M&E evaluation model:
(1) temporal change of momentum,
(2) momentum convection,
(3) forces due to wall friction,
(4) forces due to fluid pressure,
(5) forces due to gravity, and
(6) forces due to geometric head loss effects (for example, contractions, expansions, bends, and pump losses).

Additionally confirm that the frictional losses in pipes and other components are calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data.

Additionally confirm that if an uncertainty in a pressure loss exists, the pressure loss shall be conservatively minimized.

2.19 RAI-3 – Flow Modeling (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.3.4.3</td></tr>
<tr><td>Comment</td><td>In table 4-2 row 27 of their initial submittal (Reference 1), Westinghouse stated that the same flow modeling was used for the ECCS evaluation model as was used in the M&E evaluation model. However, they did not provide details on that flow modeling.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.19 RAI-3 – Flow Modeling (Reference 15)</td></tr>
</table>

Westinghouse responded with a discussion of the flow modeling effects and how it is taken in account in the WC/T M&E evaluation model. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.20 RAI-SNPB-20 – Cold leg/accumulator condensation

Cold leg / accumulator condensation	
Describe the validation data which supports WC/T ability to model cold leg/accumulator condensation and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model. 2.20 RAI-3 – Cold leg/accumulator condensation (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.3
Comment	In Section 2.9 of their initial submittal (Reference 1), Westinghouse stated that the same cold leg/accumulator condensation model was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Cold Leg/Accumulator Condensation (Reference 13)

Westinghouse provided details on an analysis performed comparing experimental data to WC/T predictions for injection in the cold leg. The analysis provided comparisons of the measured fluid temperatures and those predicted by WC/T. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.21 RAI-SNPB-21 – Downcomer condensation

<table>
<tr><td colspan="2">Downcomer condensation

Describe the validation data which supports WC/T ability to model downcomer condensation and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model.

2.21 RAI-3 – Downcomer condensation (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.3.4.3</td></tr>
<tr><td>Comment</td><td>In Section 2.10 of their initial submittal (Reference 1), Westinghouse stated that the same downcomer condensation model was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.</td></tr>
<tr><td>Reference of Response</td><td>1. SCVB-RAI-11, Downcomer Condensation (Reference 13)</td></tr>
</table>

Westinghouse provided clarification by discussing a test from WCAP-12945-P-A (Reference 2). It noted that while condensation is a parameter that cannot be directly measured; its impacts would be observable through other parameters. It discussed these parameters and provided plots of both measured and predicted values. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.22 RAI-SNPB-22 – Loop flow split

Loop Flow Split	
Describe the validation data which supports WC/T ability to model the loop flow split and demonstrate that this data justifies WC/T ability to predict the RCS transient response for the M&E evaluation model. 2.22 RAI-3 – Loop flow split (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.1 and 3.3.4.3
Comment	In Section 2.13 of their initial submittal (Reference 1), Westinghouse stated that the same loop flow split modeling was used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.
Reference of Response	1. SCVB-RAI-11, Loop Flow Split (Reference 13)

Westinghouse provided clarification by discussing a test from WCAP-12945-P-A (Reference 2), in which WC/T predicted both the broken and intact loops following a LOCA. In addition to the figures discussed in the RAI response, Figures 14-2-29 and 14-2-30 from WCAP-12945-P-A also provide information on the WC/T's ability to model the loop flow split. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.23 RAI-SNPB-23 – Hot leg condensation in NPSHa and EQ

Hot leg condensation in NPSHa and EQ *Demonstrate that the assumption to ignore any hot leg condensation is also appropriate for NPSHa and EQ analysis.* 2.23 RAI-3 – Hot leg condensation in NPSHa and EQ (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.3
Comment	In Section 2.6 of their initial submittal (Reference 1), Westinghouse stated that the hot leg condensation would be ignored as this was conservative for a containment pressure as it insured the maximum amount of steam to containment. However, Westinghouse did not address how this assumption would impact the other two purposes of an M&E analysis, NPSHa and EQ analysis.
Reference of Response	1. 2.23 RAI-3 – Hot leg condensation in NPSHa and EQ (Reference 14)

In its response, Westinghouse provided further details on hot leg condensation and discussed its role in NPSHa and EQ analysis. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.24 RAI-SNPB-24 – Dynamic pump model

<table>
<tr><td colspan="2">Dynamic pump model

Demonstrate that the dynamic pump model used in WC/T provides an appropriate prediction of the pump dynamics for the M&E evaluation model such that the mass and energy release is adequately predicted. Additionally, justify the rationale for assuming the rotor remains locked following the flow reversal during blowdown in a double ended pump suction break.

2.24 RAI-3 – Dynamic pump model (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.3.4.4</td></tr>
<tr><td>Comment</td><td>In table 4-2 row 28 of their initial submittal (Reference 1), Westinghouse stated that the same dynamic pump was used for the ECCS evaluation model as was used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation. Additionally, Westinghouse did not provide justification for the assumption of a locked rotor.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.24 RAI-3 – Dynamic pump model (Reference 15)</td></tr>
</table>

Westinghouse provided further clarification on the dynamic pump model and the locking of the rotor during reverse flow. The dynamic pump model is unchanged from the ECCS evaluation model. In both cases (ECCS and M&E) it is conservative to empty the RCS as fast as physically possible. Further, Westinghouse stated its use of a locked rotor following flow reversal is justified due to a physical anti-reversal device that exists on the pump and prevents it from rotating in the reverse direction. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.25 RAI-SNPB-25 – GOTHIC time step sensitivity

GOTHIC time step sensitivity	
Provide justification that WC/T mass and energy predictions are not sensitive to all possible time steps which are able to be used in GOTHIC in the M&E evaluation model. Additionally, demonstrate that the mass and energy are conserved between codes under all possible times steps and that no time step will result in numerical instabilities. Additionally, provide clarification on how the GOTHIC and WC/T time steps interface and when information is passed from code to code. 2.25 RAI-3 – GOTHIC time step sensitivity (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.6.1
Comment	In Sections 3.3 and 3.4 of their initial submittal (Reference 1), Westinghouse described the interface between WC/T and GOTHIC, but the NRC staff was not able to understand this description. Additionally, because of this coupling, there is a possibility that the mass and energy passed between WC/T and GOTHIC is not conserved and the NRC staff wanted to ensure this was not the case.
Reference of Response	1. 2.25 RAI-3 - GOTHIC time step sensitivity (Reference 16) 2. 2.25 RAI-3 - GOTHIC time step sensitivity & SCVB-RAI-6: GOTHIC Running In Parallel With WC/T (Reference 19)

Westinghouse provided an analysis which demonstrated that using a time step of **[**0.05**]** seconds for GOTHIC did not result in sensitivities to coupling with WC/T (Reference 16). Therefore, this time step is given as the upper limit for the GOTHIC time step size in the conditions and limitations of this SE.

Westinghouse provided additional clarification on the GOTHIC and WC/T time step interface, but this clarification did resolve an NRC staff misunderstanding on how the averaging was performed. Further discussion with Westinghouse did clarify this issue and that information is captured in Section 3.3.6.1 of the SE.

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.26 RAI-SNPB-26 – WC/T coupled vs. standalone

WC/T Coupled vs. Standalone	
Provide a comparison between results from a WC/T analysis which has been coupled to GOTHIC and a WC/T analysis which is run in standalone mode. Demonstrate that the results of the WC/T run in standalone mode are conservative compared to those coupled with GOTHIC. 2.26 RAI-3 – WC/T coupled vs. standalone (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.6.1
Comment	In Section 3.4 of their initial submittal (Reference 1), Westinghouse stated that using WC/T in standalone mode was conservative compared to the more mechanistic calculation of using it coupled to GOTHIC. However, Westinghouse did not provide any supporting analysis.
Reference of Response	1. 2.26 RAI-3 – WC/T coupled vs. standalone (Reference 16)

Westinghouse provided the requested analysis which compared WC/T and GOTHIC coupled to WC/T run in standalone mode. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.27 RAI-SNPB-27 – Heat transfer correlations

<table>
<tr><td colspan="2">Heat transfer correlations

Demonstrate that the heat transfer correlations used in WC/T provide an appropriate prediction of the heat transfer for the M&E evaluation model such that the mass and energy release is adequately predicted. Both the primary and secondary side heat transfer correlations should be considered.

2.27 RAI-3 – Heat transfer correlations (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>3.3.7.1, 3.3.7.2, and 3.3.7.6</td></tr>
<tr><td>Comment</td><td>In table 4-1 row 13 and 14 and table 4-2 row 33 of their initial submittal (Reference 1), Westinghouse stated that the same heat transfer correlations were used for the ECCS evaluation model as were used in the M&E evaluation model. However, the ECCS evaluation model is focused on obtaining an adequate prediction of PCT. On the other hand, the M&E evaluation model is focused on obtaining an adequate prediction of the mass and energy release rates to obtain an adequate prediction of containment pressures and temperatures. Because the figure of merit between the two evaluation models is substantially different, what may be conservative or adequate in one evaluation model may be non-conservative or inadequate in the other. For example, the M&E release is generally decreased to generate a conservative PCT calculation. On the other hand, the M&E release rate is generally increased to generate a conservative containment pressure calculation.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.27 RAI-3 – Heat transfer correlations (Reference 16)</td></tr>
</table>

Westinghouse provided an explicit listing of the heat transfer correlations used in the WC/T M&E analysis. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.28 RAI-SNPB-28 – Heat transfer directly to containment

Heat transfer directly to containment *Is heat transfer from the primary and secondary metal to containment directly calculated and if not why is this appropriate?* 2.28 RAI-3 – Heat transfer directly to containment (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.2 and 3.3.7.5
Comment	None
Reference of Response	1. 2.28 RAI-3 – Heat transfer directly to containment (Reference 14)

In its response, Westinghouse discussed why direct heat transfer to containment was not modeled. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.29 RAI-SNPB-29 – Inactive metal

Inactive metal *Define inactive metal and discuss how it is treated.* 2.29 RAI-3 – Inactive metal (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.2
Comment	None
Reference of Response	1. 2.29 RAI-3 – Inactive metal (Reference 14)

In its response, Westinghouse provided a description the inactive metal which included its definition and how it was treated. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.30 RAI-SNPB-30 – Unal's correlation

Unal's correlation	
Provide validation for Unal's correlation over its application domain as used in the M&E evaluation model. 2.30 Unal's correlation (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.3
Comment	Unal's correlation is a highly empirical correlation fitted to a specific range of data. Therefore, validation is needed to justify the use of the correlation.
Reference of Response	1. 2.30 RAI-3 – Unal's correlation (Reference 16) 2. 2.30 RAI-3 – Unal's correlation (Reference 19)

Westinghouse provided further discussion of the Unal correlation. It summarized the range over which the correlation is being applied and which range it had been validated over. While most uses of the Unal correlation were inside its validated range, it did exceed its range for mass flux. During the audit, this was further investigated and it was determined that Unal was outside of its range. This prompted further discussion and Westinghouse submitted a supplemental response in which it clarified that it will not use the Unal correlation beyond its validated range. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.31 RAI-SNPB-31 – Biasi Range

Biasi range	
Demonstrate that the Biasi critical heat flux correlation will provide a conservative estimate of the critical heat flux (which in this case is used to determine the time when rewet occurs) for the range over which the correlation is being applied. 2.31 Biasi Range (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.3
Comment	In Section 3.1.1 of their initial submittal (Reference 1), Westinghouse stated that the condition for rewet was going to be based on the critical heat flux calculated from the Biasi correlation. However, in the original paper for the Biasi correlation (Reference 28), the correlation's predictive capability was only validated over a small range of application domain due to the current state of computational resources. Therefore, the NRC staff questioned the correlation's predictive capability over its entire application domain.
Reference of Response	1. 2.31 RAI-3 – Biasi Range (Reference 16)

Westinghouse provided further details on the use of the Biasi CHF correlation which fully explained the correlation's application domain, the limitations of that domain, and the modifications necessary for appropriate use outside of the application domain. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.32 RAI-SNPB-32 – FLECHT heat release rate

<table>
<tr><td colspan="2">FLECHT heat release rate

Provide plots of the integrated secondary heat release rate as a function of time for the FLECHT-SEASET data and for the WC/T prediction (with the proposed interfacial heat transfer and steam generator heat transfer changes) for the seven FLECHT-SEASET cases described in the topical. Provide a discussion which demonstrates that WC/T with the proposed changes provides an adequate prediction of the FLECHT data.

2.32 FLEACHT heat release rate (Reference 10)</td></tr>
<tr><td>Disagreement Level</td><td>Level 2 – Skepticism</td></tr>
<tr><td>SE Section</td><td>3.3.7.3</td></tr>
<tr><td>Comment</td><td>In Section 3.2 of their initial submittal (Reference 1), Westinghouse provided plots of the secondary heat release rate. However, those plots seemed to indicate that WC/T with the proposed modifications consistently under predicted the heat release from the steam generator. Under predicting the heat release would be non-conservative and may result in an inadequate prediction of the mass and energy release.</td></tr>
<tr><td>Reference of Response</td><td>1. 2.32 RAI-2 – FLEACHT (sic) heat release rate (Reference 16)
2. WCAP-17721-P/NP, Revision 0, NRC Set 2, Safety and Code Review Branch - RAI 2.32 Response Supplement (Proprietary) (Reference 17)</td></tr>
</table>

Westinghouse provided further discussion of the comparison with the FLECHT-SEASET data. In its initial response, Westinghouse did not provide enough details to satisfy the NRC staff's questions on the difference between the FLECHT data and the WC/T predictions. In its supplement, it further discussed the differences and provided a basis which supported the conclusion that WC/T is providing an adequate prediction of the FLECHT data. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.33 RAI-SNPB-33 – Secondary side heat transfer

Secondary side heat transfer	
Specify how the heat is treated between the secondary side metal to the secondary side coolant, and from the secondary side coolant to the steam generator tubes. 2.33 RAI-3 – Secondary side heat transfer (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.4
Comment	In the initial submittal (Reference 1), Westinghouse did not specify how this heat transfer was treated.
Reference of Response	1. 2.33 RAI-3 – Secondary side heat transfer (Reference 14)

In its response, Westinghouse provided a description of the heat transfer mechanisms and the heat transfer correlations used. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.34 RAI-SNPB-34 – Definitions for acronyms

Definitions for acronyms	
Provide the definition for the following acronyms: PCWG, DEPSG, EQ, NPSHa, DEHLG, GENF 2.34 RAI-6 – Definitions for acronyms (Reference 10)	
Disagreement Level	Level 6 – Editorial
SE Section	Various
Comment	None
Reference of Response	1. 2.34 RAI-6 – Definition for acronyms (Reference 14)

In its response, Westinghouse provided the definition of the acronyms. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.35 RAI-SNPB-35 – Clarification on quench front paragraph

Clarification on quench front paragraph	
The first full paragraph on page 3-5 does not make sense. Revise this paragraph and re-submit it. 2.35 RAI-4 – Clarification on quench front paragraph (Reference 10)	
Disagreement Level	Level 4 - Clarification Necessary
SE Section	Various
Comment	None
Reference of Response	1. 2.35 RAI-4 – Clarification on quench front paragraph (Reference 15)

Westinghouse provided a revised paragraph which clearly explains the heat transfer process in the vicinity of the quench front. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.2.36 RAI-SNPB-36 – Clarification on material properties

Clarification on material properties	
The NRC staff is aware of Westinghouse's InfoGram IG-14-1 which discusses material properties for LOCA M&E release analyses. However, the InfoGram does not explicitly discuss the material properties used in WC/T. Are the volumetric heat capacities being in the WC/T M&E analysis lower than those given in the ASME Boiler and Pressure Vessel Code apply to WC/T? If so, provide appropriate justification for the use of lower volumetric heat capacities. (Reference 20)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	N/A
Comment	Westinghouse responded to the draft RAI before the final RAI was issued. There was no difference between the draft and final RAI in content.
Reference of Response	1. WCAP-17721-P/NP, Revision 0, - Response to Draft RAI (Non-Proprietary) (Reference 21and 20)

Westinghouse provided a comparison between the material properties used in WC/T and those provided in the ASME BPVC (Reference 30). This comparison showed that the property values used in WC/T agreed reasonably well with those contained in the BPVC. However, there were two deviations. First, the specific heat of Inconel 600 and 690 used in WC/T was consistently lower than reported in BPVC by 3 percent. While any correction of this specific heat would be minimal, it would increase the energy stored in the metal. Therefore a limitation has been placed such that the approved method must account for this additional energy.

Second, the specific heat of stainless steel used in WC/T was slightly lower than BPVC at lower temperatures. The NRC staff agrees that stainless steel is a very small contributor to the total energy in the vessel (which is a thick carbon steel vessel with a thin stainless steel liner) and in the loop (which is mostly carbon steel and Inconel). Further, this discrepancy is only at lower temperatures and would therefore have a minimal impact on the mass and energy release. Based on the information provided by Westinghouse, the NRC staff has concluded that this RAI has been resolved.

A.3 RAIs from the Containment and Ventilation Branch (SCVB)

A.3.1 RAI-SCVB-1 – Methodology on modeling containment condition

Methodology on modeling containment condition	
Please describe the input requirement or disposition of the containment condition. The containment condition may include the containment pressure and temperature conditions at the break locations and safety injection flow conditions during recirculation phase. Provide information about whether a containment model like the one developed with GOTHIC as shown in WCAP-17721-P/NP, Revision 0, TR is required in order to provide the required containment condition. If a containment model is required, please prescribe such a requirement in WCAP-17721-P/NP, Revision 0. Otherwise, please specify how the containment condition shall be input and justify its conservatism for the containment response. SCVB-RAI-1, Methodology on Modeling Containment Condition (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-1, Methodology on Modeling Containment Condition (Reference 13)

Westinghouse identified the input required pertaining to the containment for the M&E analysis. It also discussed each input, where the input was obtained from, and if the input was biased to make the calculation conservative. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.2 RAI-SCVB-2 – Direct vessel injection and ADS-4 operation

Direct vessel injection and ADS-4 operation	
Provide the reason to introduce the direct vessel injection and ADS-4 operation in WCAP-17721-P/NP, Revision 0, that, as described, are applicable to the passive plant design. If WCAP-17721-P/NP, Revision 0, TR covers these two operations, then a complete description and justification of modeling with WC/T should be provided in WCAP-17721-P/NP, Revision 0. Otherwise, they should be removed from TR WCAP-17721-P/NP, Revision 0. SCVB-RAI-2, Direct Vessel Injection and ADS-4 Operation (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-3, Direct Vessel Injection and ADS-4 Operation (Reference 13)

Westinghouse discussed that while these two phenomena were included in the WCAP-17721-P/NP, Revision 0, they were only included for completeness of the PIRT and they do not apply to non-passive plant designs. Westinghouse has provided this information and because it has committed to revising the TR to make the discussion clearer about how these do not apply, the NRC staff has concluded that this RAI has been resolved.

A.3.3 RAI-SCVB-3 – Control of applicability

Control of applicability	
Provide the control measure that is put in place to prevent the analyst from applying the WCAP-17721-P/NP, Revision 0, TR methodology beyond its scope and range of applicability. Provide the details and associated basis for this control measure. SCVB-RAI-3, Control of Applicability (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-3, Control of Applicability (Reference 15)

Westinghouse responded by stating that methodology specific guidance was currently being developed to ensure the methodology is not applied beyond its scope and that guidance would be available for NRC staff review at an audit. At the audit conducted on April 8-9, 2015 (Reference 18), the NRC staff was able to review the current version of the guidance document which will be used to ensure appropriate application of the WCAP-17721-P/NP, Revision 0, methodology. The NRC staff concluded that this document was consistent with similar guidance documents and believes that it will enable appropriate application of the methodology. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.4 RAI-SCVB-4 – Break spectrum

Break spectrum	
For the Item 9 of Table 4-1, justify why the result of break spectrum sensitivity studies using the previous methodology is applicable to the proposed WCAP-17721-P/NP, Revision 0, TR methodology in which the double-ended break will still be limiting when the proposed methodology is used for break spectrum studies. SCVB-RAI-4, Break Spectrum (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-4, Break Spectrum (Reference 15)

Westinghouse stated that the answer to this RAI was supplied in its response to RAI-SNPB-2 – Break Size (Reference 15). In that response, Westinghouse provided details of a sensitivity study performed with slots breaks of three different sizes. This study demonstrated that double ended break resulted in higher containment pressures than a similar slot break. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.5 RAI-SCVB-5 – GOTHIC topical report

GOTHIC topical report	
Section 5.2 stated that an approved GOTHIC TR was used in the proposed WCAP-17721-P/NP, Revision 0, TR methodology. Please add the approved GOTHIC TR in the reference section (Section 9). In addition, specify the GOTHIC version that can be as used with WC/T. SCVB-RAI-5, GOTHIC Topical Report (Reference 11)	
Disagreement Level	N/A
SE Section	3.3.6.1
Comment	None
Reference of Response	1. SCVB-RAI-5, GOTHIC TR (Reference 13)

Westinghouse explained how GOTHIC could be used in the M&E analysis, but its use is not required. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.6 RAI-SCVB-6 – GOTHIC running in parallel with WC/T

GOTHIC running in parallel with WCOBRA/TRAC *Provide the methodology for the requirement of time steps, e.g. maintaining relative magnitude or ratio of time steps, used in both GOTHIC and WC/T when GOTHIC is running in parallel with WC/T. Provide these time steps used in both Sec. 5.3 and 6.1 cases.* SCVB-RAI-6, GOTHIC Running In Parallel With WC/T (Reference 11)	
Disagreement Level	N/A
SE Section	3.3.6.1
Comment	None
Reference of Response	1. SCVB-RAI-6, GOTHIC Running In Parallel With WC/T (Reference 13) 2. SCVB-RAI-6, GOTHIC Running In Parallel With WC/T (Reference 19)

Westinghouse provided a sensitivity analysis which demonstrated that the time step used in GOTHIC is insensitive to drastically smaller time steps. This sensitivity included a case where the GOTHIC time step was less than that used in WC/T.

Westinghouse provided additional clarification on the GOTHIC and WC/T time step interface, but this clarification did resolve an NRC staff misunderstanding on how the averaging was performed. Further discussion with Westinghouse did clarify this issue and that information is captured in Section 3.3.6.1 of the SE.

Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.7 RAI-SCVB-7 – 24-hr containment pressure

24-hr containment pressure	
Figure 6-2 shows that the containment pressure, especially for double-ended pump suction (DEPS) case, is rising after 5000 seconds. Provide 24-hour (86400 seconds) containment response for both DEPS (Section 5.3) and double-ended hot leg (DEHL) (Section 5.4) cases. Provide the analysis procedure (methodology) to assure the adequacy of containment heat removal system by demonstrating, for example, that the containment pressure will be below 50 percent of peak pressure after 24-hour into LOCA (NUREG-0800, "Standard Review Plan," Section 6.2.2, "Containment Heat Removal Systems."). SCVB-RAI-7, 24-hr Containment Pressure (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-7, GOTHIC Running In Parallel With WC/T (Reference 13)

Westinghouse provided an explanation of which break would be limiting and also provided a 24-hour simulation following that break. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.8 RAI-SCVB-8 – 24-hr integrated mass and energy release via break

24-hr integrated mass and energy release via break	
Please provide the integrated mass and energy release from break up to 24-hour (86400 seconds) for Figures 5-7 and 5-8; similarly, for DEHL case (Section 5.4). SCVB-RAI-8, 24-hr Integrated Mass and Energy Release via Break (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-8, 24-hr Integrated Mass and Energy Release via Brea (Reference 13)

Westinghouse provided the requested plots for the DEPS case and an explanation of why that case was more appropriate than the DEHL case. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.9 RAI-SCVB-9 – Conservatism of calculated containment pressure peak

Conservatism of calculated containment pressure peak	
Use the sample cases in Section 5.3 or 6.1 as examples to provide the order of magnitude of the conservatism of containment peak pressure by assuming that the conservatism is completely due to the mass and energy release calculation with WCAP-17721-P/NP, Revision 0, methodology. SCVB-RAI-9, Conservatism of Calculated Containment Pressure Peak (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-9, Conservatism of Calculated Containment Pressure Peak (Reference 13).

Westinghouse provided an analysis that estimated the magnitude of the conservatism using the input biasing described in WCAP-17721-P/NP, Revision 0 (Reference 1). This analysis demonstrated the margin available for both large dry and ice condenser containments. This margin evaluation was based on WC/T with best estimate inputs case vs WC/T with inputs used in a manner consistent with the topical. Additionally, Westinghouse commented that the margin was due to input biasing described in the TR, but it is likely that other inputs not described in the TR (i.e., parameters obtained from licensees) would also be biased and result in additional margin. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.10 RAI-SCVB-10 – Limitation of containment modeling for WCAP-17721-P/NP, Revision 0, methodology

Limitation of containment modeling for WCAP-17721-P/NP, Revision 0, methodology	
Provide the limitations on the containment model in order to employ the coupling methodology between WC/T and GOTHIC as described in WCAP-17721-P/NP, Revision 0. Note that Table 5-2 indicates that the containment is modeled as one single lumped volume. Will the multi-volume containment model not be applicable to the methodology? SCVB-RAI-10, Limitation of Containment Modeling for WCAP-17721-P/NP, Revision 0, Methodology (Reference 11)	
Disagreement Level	N/A
SE Section	Various
Comment	None
Reference of Response	1. SCVB-RAI-10, Limitation of Containment Modeling for WCAP-17721, Revision 0, Methodology (Reference 13)

Westinghouse discussed the potential limitations on the GOTHIC model and identified the only potential limitation is on the time step size due to coupling with WC/T. This issue is being addressed through “SNPB-RAI-25, GOTHIC time step sensitivity” (Section A.2.25). Westinghouse further identified that there are no known limitations on the type of containment model that could be used to employ the coupling methodology. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.11 RAI-SCVB-11 – Conformance of Regulatory Guide 1.203, "Transient and accident analysis methods"

<table>
<tr><td colspan="2">

Conformance of Regulatory Guide (RG) 1.203, "Transient and accident analysis methods"

Please justify the applicability of WCAP-12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998, to the demonstration of WCAP-17721-P/NP, Revision 0, conforming to RG 1.203, "Transient and Accident Analysis Methods." Note that all development and assessment made in WCAP-12945-P-A is focused on the figure-of-merit of peak cladding temperature (PCT) while the expecting development and assessment for WCAP-17721-P/NP, Revision 0, is known to be the figure-of-merit of the peak of containment temperature and pressure. The timing (order of 100 seconds) and thermal-hydraulic conditions (reflood phase) determining PCT are not necessarily the same as those for peak containment temperature and pressure (i.e., order of 10 and 1000 seconds for blowdown and post-reflood phase, respectively). An equivalent code qualification document for TR WCAP-17721-P/NP, Revision 0, is expected to be developed and assessed with respect to the peak containment temperature and pressure, or, at least, the corresponding mass and energy release.

Conformance of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" (Reference 11)

</td></tr>
<tr><td>Disagreement Level</td><td>Level 3 - Lacking Documentation</td></tr>
<tr><td>SE Section</td><td>Various</td></tr>
<tr><td>Comment</td><td>None</td></tr>
<tr><td>Reference of Response</td><td>1. Presentation on Regulatory Guide 1.203 Compliance (Reference 6)
2. SCVB-RAI-11, Conformance of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" (Reference 13)</td></tr>
</table>

In Westinghouse's response to this RAI, it also responds to many other RAIs, such as RAI-SNPB-1, RAI-SNPB-3, RAI-SNPB-5, RAI-SNPB-6, RAI-SNPB-7, RAI-SNPB-8, RAI-SNPB-9, RAI-SNPB-20, RAI-SNPB-21, and RAI-SNPB-22. For this response, it evaluated the high ranked phenomena for M&E analysis and discussed why that specific treatment was appropriate. Westinghouse has provided this information; therefore, the NRC staff has concluded that this RAI has been resolved.

A.3.12 RAI-SCVB-12 – Audit

<table>
<tr><td colspan="2">

Audit

Provide the following calculations for audit via electronic reading room:

1. Calculation(s) supporting Section 5.1 (LOCA Mass and Energy (M&E) Model)
2. Calculation(s) supporting Section 5.2 (Containment Model)
3. Calculation(s) supporting Section 5.3 (DEPS LOCA Benchmark Case)
4. Calculation(s) supporting Section 5.4 (DEHL LOCA Benchmark Case)
5. Calculation(s) supporting Section 6 (Sample Cases) if not included in Section 5.3 and 5.4.
6. Calculation(s) supporting Section 7.1 (LOCA M&E Model)
7. Calculation(s) supporting Section 7.2 (Containment Model)
8. Calculation(s) supporting Section 7.3 (DEPS LOCA Benchmark Case)
9. Calculation(s) supporting Section 7.4 (Sample Case) if not included in Section 7.3.

SCVB-RAI-12, Audit (Reference 11)
</td></tr>
<tr><td>Disagreement Level</td><td>Level 5 - Basic Information</td></tr>
<tr><td>SE Section</td><td>Various</td></tr>
<tr><td>Comment</td><td>None</td></tr>
<tr><td>Reference of Response</td><td>1. Audit performed - see Audit Summary (Reference 12)</td></tr>
</table>

The information presented in the calculations presented at the audit provided the staff with a better understanding of the containment M&E release model development and analysis with WC/T, GOTHIC, and LOTIC1 as described in WCAP-17721-P/NP, Revision 0. Westinghouse presented this information; therefore, the NRC staff has concluded that this RAI has been resolved.

Acknowledgements

The principal technical contributors to the WCOBRA/TRAC LOCA M&E methodology were, in alphabetical order; John Besspiata, Dr. Liping Cao, Megan Durse, Ruben Espinosa, Chris Logan, Rich Lukas, Mitch Nissley, Rick Ofstun, and Dr. Katsuhiro Ohkawa. Their efforts are greatly appreciated.

Debbie Sommer was the project manager for much of the methodology development process, and we owe her thanks for her support.

Kent Bonadio and Amy Colussy should also be thanked for their support as the managers of the groups who provided input to this methodology; Containment and Radiological Analysis and LOCA Integrated Services I.

Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology

WCAP-16608-P-A 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.

WCAP-16608, Appendix C was submitted for NRC review in July 2007; it described the PWR LOCA mass and energy (M&E) release input calculation methodology. Appendix C was withdrawn from NRC review in December 2009 after several issues were discovered with the LOCA M&E version of the WCOBRA-TRAC (WC/T) code. These issues have since been resolved and the LOCA M&E version of WC/T has been merged into the standard version of WC/T.

Westinghouse met with the NRC in August 2012 to discuss the best approach for submitting an updated version of WCAP-16608, Appendix C. After that meeting, Westinghouse decided it would be best to submit two separate topical reports instead of a single updated version of WCAP-16608 Appendix C – one for the current operating Westinghouse and CE NSSS plant designs and the other for the passive AP1000® plant design. Therefore, a new number was assigned to the topical report describing the LOCA M&E release calculation methodology for the current operating plants; it is now WCAP-17721-P.

Much of the information contained in this new topical report is the same as was previously presented in WCAP-16608, Appendix C. However, a couple of new sections have been added and a copy of the Westinghouse response to previous RAIs has been included as an appendix to the report. One of the new sections describes how the high ranked LOCA M&E phenomena identification and ranking table (PIRT) items are addressed in the WC/T model. The other new section provides benchmark comparisons and sample cases for application of the methodology to the ice condenser containment design analyses. Finally, the section that describes the WC/T code changes for the LOCA M&E calculations has been updated to explain how the M&E release output data files are generated when WC/T is running in the stand-alone mode.

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NOMENCLATURE

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

Subscripts

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

1 INTRODUCTION

The mass and energy release input data is the primary input 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 topical report describes how Westinghouse calculates the mass and energy release input for the Pressurized Water Reactor (PWR) containment models for the various analyzed Loss of Cooling Accident (LOCA) event applications.

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 Reactor Coolant System (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 Steam Generator (SG) stored energy is released. If the break is located upstream of a SG (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 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.

1.1 CURRENT WESTINGHOUSE NSSS LOCA M&E RELEASE METHODOLOGY

The current Westinghouse LOCA mass and energy (M&E) release model methodology is documented in References 1 and 2. The model uses a series of three codes to calculate the mass and energy release input for the containment analysis. SATAN-VI (Reference 3) performs the blowdown phase M&E release calculations. SATAN-VI models the RCS thermal-hydraulic response with a detailed 1-D nodal network containing one lumped loop (to represent the intact loops) and one broken loop. WREFLOOD (Reference 4) covers the reflood phase of the LOCA event. WREFLOOD also 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 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 2). A third code, EPITOME, combines the

output M&E data files from SATAN-VI and REFLOOD10325. It then adjusts the break releases that were 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 backpressure is assumed to remain at design pressure during the blowdown phase. Sensitivity studies documented in WCAP-10325-P-A (Reference 2) confirmed that this is a conservative assumption. 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 in accordance with NUREG-0800 Section 6.2.1.3 (Reference 5). 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.

    a. 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.

    b. Because 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.

    c. Because the sub-atmospheric containment design is required to be depressurized to atmospheric pressure within one hour of event initiation (Reference 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, ice condenser, and large dry containment designs.

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 standard Westinghouse NSSS containment designs and this method will continue to be used, if requested by our customers.

1.2 CURRENT WESTINGHOUSE CE/ABB NSSS LOCA M&E METHODOLOGY

The Combustion Engineering Asea Brown Boveri (CE/ABB) LOCA M&E release model methodology is documented in References 6, 7, and 8. The model uses a series of three codes to calculate the mass and energy release input for the containment analysis. CEFLASH-4A (Reference 6) performs the blowdown phase M&E release calculations. CEFLASH-4A models the RCS thermal-hydraulic response with a detailed 1-D nodal network containing the two hot legs, the two steam generators, and the four cold legs. FLOOD3 (Reference 7) describes 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 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 cooldown are 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 NSSS containment designs and this method will continue to be used, if requested by Westinghouse customers.

1.3 REASONS TO UPDATE THE LOCA M&E METHODOLOGY

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 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.

2 IMPORTANT PHENOMENA FOR MODELING LOCA M&E

A phenomena identification and ranking table (PIRT) for the Westinghouse LOCA M&E release calculation was developed and is documented in Reference 23. The LOCA M&E release PIRT is very similar to the LOCA Peak Clad Temperature (PCT) PIRT (Reference 29, Section 1.2.3) and includes many of the same phenomena.

The high ranked phenomena from the LOCA M&E release PIRT are shown in Table 2-1. The methods that are used in the WC/T LOCA M&E release model to address each of these high ranked phenomena are summarized in this section.

2.1 BREAK FLOW

Most of the initial RCS fluid mass and energy is released out of the break during the blowdown phase of the LOCA event; the larger the break size, the faster the release rate. The break flow rate is maximized by assuming a full double-ended pipe rupture. The break location is varied between the hot leg, cold leg, and pump suction leg. Most of the RCS fluid flashes into steam once it enters the containment. This causes a rapid increase in the containment pressure, resulting in what is known as the blowdown peak for the large dry, sub-atmospheric, and passive containment designs. The double-ended hot leg (DEHL) break location produces the highest steam release rate, and subsequently, the highest blowdown peak pressure.

The critical flow model is a high ranked phenomenon during the blowdown phase of the transient. Along with the break location, the critical flow model determines the system response and break flow rate during this phase of the LOCA event. The TRAC PF1 break flow model is used in WC/T. The WC/T break flow model predictions of the Marviken critical flow data are presented in Section 25-2 of Reference 10. The resulting cumulative distribution function is shown in Figure 25-2-10. The 50th percentile value of the measured/predicted break flow is about [

]^{a,c}.

The measured/predicted break flow rate is a function of the pipe length to diameter ratio (L/D); it is greater for small values of L/D (< 1), but decreases as L/D increases. The L/D for a large double-ended pipe break near the vessel is greater than 1.5. For these types of breaks, the TRAC PF1 break flow model will slightly over-predict the break flow rate.

The WC/T calculated DEHL and double-ended pump suction (DEPS) LOCA M&E releases were compared with those calculated using the previously approved LOCA M&E model (Reference 2), and were found to be very close over the blowdown period. [

]^{a,c} The SATAN-VI break flow correlations were compared with data from other test facilities and found to over-predict the data (see Section III of Reference 1).

Table 2-1 Summary of High Ranked Phenomena and Associated Data

Phenomena	Transient Phase	Source of Data
Break: Critical Flow	Blowdown	Marviken, LOFT
Fuel Rod: Stored Energy Release	Blowdown	Initial Stored Energy – Fuel Related Input Parameters Transient Aspects – LOFT, ORNL, G-1, G-2
Fuel Rod: Decay Heat	Reflood Long Term	ANS 1979 Decay Heat Standard
Core: Reflood Heat Transfer	Reflood	FLECHT, FLECHT SEASET, Semiscale, SCTF, CCTF,
Upper Plenum: Entrainment/De-entrainment, Condensation (UPI)	Reflood Long Term	CCTF, SCTF, UPTF
Hot Leg: Condensation	Long Term	UPTF
Hot Leg: Entrainment/De-entrainment	Reflood Long Term	CCTF, UPTF
Steam Generator: Heat Transfer (both primary and secondary, includes secondary stratification), Steam binding	Reflood Long Term	FLECHT SEASET, CCTF, UPTF
Cold Leg/Accumulator: Condensation	Reflood	W/EPRI 1/3 scale test, COSI
Downcomer: Condensation	Reflood	UPTF
Downcomer: Stored energy release (includes Hot Wall, saturated nucleate boiling, quenching)	Reflood	LOFT, CREARE, UPTF
Downcomer: Direct vessel injection	Reflood	UPTF, CCTF
Loop: Flow Split, losses, pump, PRHR	Reflood Long Term	UPTF, CCTF
ADS-4: Entrainment	Post-ADS-4	APEX

2.2 CORE STORED ENERGY RELEASE

Core stored energy is defined as the thermal energy in the fuel and cladding that is greater than a given reference temperature.

The core stored energy release is a high ranked phenomenon during the blowdown phase of the LOCA event. However, because the core stored energy is only a small fraction (< 5%) of the total energy released to containment during blowdown, this phenomenon is not as important for the LOCA M&E release calculation as it is for the LOCA PCT calculation.

The amount of core stored energy that is released during the blowdown phase depends on the break location. The flow through the core remains positive during the DEHL event, but can stagnate and reverse

direction during a double ended cold leg (DECL) or DEPS LOCA event. This affects the heat transfer rate and the release of stored energy from the fuel during the blowdown phase.

Most of the core stored energy is released during the blowdown phase for a DEHL LOCA event. Because the core heat transfer rate is reduced when the flow rate stagnates, some of the initial core stored energy is retained in the fuel at the end of blowdown phase for the DECL and DEPS LOCA events. This is one reason why the calculated peak containment pressure is lower at the end of the blowdown phase for the DECL and DEPS LOCA events than it is for the DEHL LOCA event (see Figure 6-1).

[

]a,c

2.3 DECAY HEAT

The decay heat is a high ranked phenomenon during the reflood and long-term phases of the LOCA event. Decay heat from residual fissions and various fission products continues to be generated after the reactor is tripped. A higher decay heat rate will produce a higher break energy release and subsequently higher containment pressure and temperature.

[

]a,c

2.4 REFLOOD HEAT TRANSFER

Core heat transfer is a high ranked phenomenon during the reflood phase of the LOCA event. Because the flow rate through the core stagnates and/or reverses during the blowdown phase of the DECL and DEPS LOCA events, some of the original core stored energy remains after blowdown in these events. The remaining core stored energy (greater than the saturation temperature), along with the energy that is generated by residual fissions and fission product decay, is removed from the fuel during the reflood phase.

Forced reflood tests from five different facilities were simulated: FEBA, FLECHT Low Flood Rate, FLECHT skewed power, FLECHT-SEASET, and the G-2 loop. The FEBA reflood experiments had a different axial power shape as well as matching tests with and without the mid-plane spacer grid. The FLECHT-SEASET tests had reliable non-equilibrium vapor temperature data, axial void fraction or pressure drop data, as well as droplet diameter, velocity data, and heater rod temperature data. The G-2 reflood experiments had prototypical spacer grid geometry.

The WC/T code contains standard correlations that cover the various modes of heat transfer during the reflood phase. These correlations have been validated using scaled test data over the expected operating range for the various LOCA events. The WC/T heat transfer models have been tested over a wide range of fluid conditions, axial power shapes, bundle power, bundle array geometry, and fuel assembly designs. The WC/T calculated results for the tests described above are presented in Section 12 of Reference 10. A comparison of WC/T model results with the reflood heat transfer composite test results is presented in Section 13-5 of Reference 10.

[

]^{a,c}

Reflood heat transfer was also present in the tests and simulations of SCTF, CCTF, and LOFT, which were described in Section 14 of Reference 10. The code calculated clad temperatures were conservatively predicted for these tests. [

]^{a,c} This results in a conservatively higher calculated containment pressure response during the reflood phase and beyond.

2.5 UPPER PLENUM ENTRAINMENT/DE-ENTRAINMENT AND CONDENSATION

Entrainment/de-entrainment processes that determine the net entrainment of liquid from the upper plenum are high ranked phenomena during the reflood and long-term phases of the DECL and DEPS LOCA events. Entrained liquid drops from the core can be carried up into the upper plenum. Some of the entrained liquid drops will de-entrain on the structures within the upper plenum and form a pool above the upper core plate. The remainder of the entrained liquid drops will be swept into the hot legs and the SG tubes.

The entrainment of fluid at the quench front in the core was examined in tests performed under three different forced reflood conditions (FLECHT-SEASET, FLECHT Low Flooding Rate, and FLECHT Top Skewed Power). The FLECHT facility was full scale in height and represented the downcomer, core, and upper plenum of a reactor vessel. [

]^{a,c}

The de-entrainment of fluid within the upper plenum initially tends to form a pool at the top of the upper core plate as it is held up by steam rising up through the core. The water from this pool would begin to drain down through the fuel assemblies after the core steaming rate decreases and the steam velocity through the upper core plate is reduced. The water would normally first start to drain down through the lower power fuel assemblies that are located on the periphery of the core because they would generate less steam than the average or higher power assemblies located near the center of the core. [

]^{a,c}

The de-entrainment of fluid within the upper plenum and amount of fluid carried over to the hot legs was examined in tests performed at the Upper Plenum Test Facility (UPTF) and the Slab Core Test Facility (SCTF). The UPTF was full scale in both height and cross section and the SCTF represented a full scale radial slice of a PWR core. [

]^{a,c}

For upper plenum injection (UPI) plants, cold water from the low pressure Emergency Core Cooling System (ECCS) pumps is injected directly into the upper plenum during the refill and reflood phases of the large LOCA event. This affects both the core reflood rate and the LOCA M&E release rate. The UPI flow will form a pool above the upper core plate and counter-current flow will affect the reflood rate. The cold water will also condense some of the steam that is generated by boiling in the core before it can enter the hot legs.

The qualification of WC/T for modeling large break LOCA events with UPI is documented in Reference 28. Modeling of steam condensation by the ECCS flow that is injected into the upper plenum is important for the LOCA M&E release calculation because this will significantly reduce the steam flow rate through the loops and out into containment. [

]^{a,c}

2.6 HOT LEG CONDENSATION

Condensation in the hot legs during hot leg recirculation is a high ranked phenomenon during the long-term phase of the DECL and DEPS LOCA events. The transfer from cold-leg to hot-leg recirculation occurs several hours after the LOCA occurs. Condensation will reduce the steam release rate and this will decrease the calculated containment pressure. Hot leg recirculation is not modeled in the WC/T LOCA M&E release calculations.

2.7 HOT LEG ENTRAINMENT/DE-ENTRAINMENT

Hot leg entrainment and de-entrainment are high ranked phenomena during the reflood and long-term phases of the DECL and DEPS LOCA events. The liquid drops that are entrained in the flow from the vessel during the reflood and long-term phases can de-entrain and form a pool at the bottom of the hot legs or steam generator inlet plenums. The core reflood rate and steam generator heat release rate are affected by the amount of liquid that is able to flow into the steam generator tubes. Evaporation of the entrained liquid by the hot steam generator tubes causes the local pressure to increase and this steam binding effect decreases the core reflood rate.

[

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A slip multiplier is included in the WC/T code for potential use in the LOCA M&E application. The slip multiplier is applied to the calculation of the relative velocity between the vapor and liquid phases and provides a method to adjust the entrainment rate in the loops. The slip multiplier may be adjusted to a value between 0.0 and 1.0; the default value is 1.0. Using a slip multiplier value less than 1.0 in the hot leg could potentially allow more liquid to be carried out of the hot legs and into the steam generators. More liquid flowing through the steam generator is conservative for the LOCA M&E calculation because this would cause the steam generator to cool down faster.

Sensitivity cases were made varying the WC/T slip multiplier input values in the hot legs of the 4-loop plant model. The slip multipliers were varied between 0.10 and 1.0 for the DEPS LOCA case. Decreasing the hot leg slip multiplier input values in the WC/T LOCA M&E release calculation affected the broken loop void fraction, resulting in a higher liquid mass flow rate to that steam generator. However, there was essentially no difference in the calculated containment pressure response because the broken loop steam generator secondary fluid temperature was essentially equilibrated with the RCS steam temperature in all cases within the first 5 minutes after event initiation. Therefore, the containment response model is not sensitive to the WC/T slip multiplier input value, which has some effect on the hot leg entrainment, so the default value is used in the LOCA M&E release calculation.

2.8 STEAM GENERATOR HEAT TRANSFER

Steam generator heat transfer is a high ranked phenomenon during the reflood and long-term phases of the DECL and DEPS LOCA events. After the primary coolant energy is released during blowdown, the next largest source of energy for potential release to the containment is the fluid energy stored within the steam generators. The SG secondary fluid temperature is higher than the RCS temperature after blowdown, so the heat transfer will be reversed during the reflood and long-term phases of the DECL and DEPS LOCA events. The SG secondary fluid transfers energy to the two-phase flow passing through the SG tubes and cools from the bottom up. The FLECHT-SEASET tests (Reference 9) show the SG secondary coolant temperature stratifies during the long-term LOCA phase.

After the SGs have been isolated and the blowdown is over, the heat transfer rate from the SG secondary fluid to the outside of the SG tubes is primarily by natural convection. Heat is transferred from the SG secondary fluid to the upside SG tubes and to the SG secondary fluid from the downside SG tubes. The [ ]^{a,c} free convection heat transfer correlation is used, and has been previously accepted, for modeling heat transfer between the SG tubes and SG secondary fluid under these conditions.

The SG secondary fluid energy that is transferred to the SG tubes is released to the primary coolant by convection and evaporation of water drops in the two-phase flow that travels through the SG tubes. The heat transfer rate on the inside surface of the SG tubes is affected by the steam flow rate and amount of entrained liquid. [ ]^{a,c} The vaporization of entrained drops in the SG tubes will increase the pressure at the entrance to the SG tubes (steam binding). This could temporarily reduce the flow through the core and could reduce the core stored energy release rate for the DECL and DEPS LOCA events during the reflood phase. [ ]^{a,c}

As described in Section 3.1, an option was added to the WC/T code to improve the heat transfer modeling inside the SG tubes for the long-term LOCA M&E calculations. The interfacial heat and mass transfer and the [ ]^{a,c} nucleate boiling correlations in WC/T were modified to more accurately calculate the two-phase flow heat and mass transfer within the SG tubes. [ ]^{a,c}

WC/T model comparisons with the FLECHT-SEASET test data are presented in Section 3.2. [ ]^{a,c}

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2.9 COLD LEG/ACCUMULATOR CONDENSATION

Condensation in the cold legs, due to accumulator and safety injection, is a high ranked phenomenon during the reflood phase of the DECL and DEPS LOCA events. Condensation in the cold legs reduces the amount of steam that is released to containment through a break in the cold leg or pump suction leg, and thus reduces the containment pressure. [

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The cold leg break location could be modeled either upstream or downstream of the injection line; [ ]^{a,c}. If it is modeled downstream of the injection location (near the vessel), WC/T would calculate the condensation of some of the steam coming from the core and steam generator side of the break before releasing the mixture to containment. If it is modeled upstream of the injection location (near the pump discharge), most of the steam condensation would be calculated outside WC/T by the containment code. The containment response during blowdown should be about the same for both cold leg break locations. However, after blowdown, the containment pressure response for the case with the cold leg break located near the pump discharge would look like a pump suction break.

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2.10 DOWNCOMER CONDENSATION

Condensation in the downcomer, due to accumulator and safety injection, is a high ranked phenomenon during the reflood phase of the DECL or DEPS LOCA event. Condensation in the downcomer reduces the amount of steam released to containment through a break in the cold leg or pump suction leg, and thus reduces the containment pressure. [

]^{a,c} This is conservative for the WC/T LOCA M&E release calculation because it yields a higher calculated post-blowdown containment pressure response.

2.11 DOWNCOMER STORED ENERGY RELEASE

The release of stored energy in the vessel downcomer is a high ranked phenomenon during the reflood phase of the DECL and DEPS LOCA events. The reason behind this high ranking is that the release of the vessel downcomer stored energy could cause the water that is refilling the downcomer to boil. This would reduce the effective downcomer driving head and entrain water out of the vessel through the broken loop cold leg, which would slow the core reflood rate and core energy release rate. The reflood rate is very important for the LOCA peak clad temperature calculation, but not as important for the LOCA M&E release calculation because the core energy release rate is much less than the sum of the RCS metal stored energy and steam generator energy release rates during this time period.

Test 25, from the UPTF, was used to investigate downcomer boiling and entrainment during the reflood phase, among other things. The test was divided into two phases, each with several parts. Phase A of the test started with the downcomer wall temperature approximately 144°F greater than the saturation

temperature. The steam flow rate was varied between 30 and 15 kg/s per loop while the simulated ECC flow was maintained at 80 kg/s per loop.

Comparisons between WC/T predictions and measurements from UPTF Test 25 are presented in Sections 14-4-11 and 15-2-4 of Reference 10. [

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2.12 DIRECT VESSEL INJECTION

The phenomena associated with direct vessel injection are applicable to the passive plant design. The LOCA M&E release calculation methodology for the passive plant design is described in a separate document that is not part of this topical report.

2.13 LOOP FLOW SPLIT

The loop flow split is a high ranked phenomenon during the reflood and long term phases of the DECL and DEPS LOCA events. The two-phase flow that exits the core must pass through a steam generator before it can exit a break in the pump suction or cold leg. The amount of flow to each loop affects the SG heat release and condensation rate in that loop, which affects the transient mass and energy release from the break to containment.

A flow resistance is modeled for each component in the WC/T LOCA M&E model. The resistance input values are adjusted to produce the correct pressure drop in each component at the specified steady state initial conditions.

After the LOCA occurs, the amount of flow that passes through each loop is primarily affected by the flow path resistance to the break location. Most of the two-phase flow coming out of the upper plenum will pass through the broken loop steam generator because this path offers the least resistance to a break in the pump suction or cold leg. The location of the cold leg break could affect the long-term containment pressure and temperature response because some of the broken loop steam flow can be condensed by safety injection if the break is located near the vessel instead of near the pump discharge. The rest of the two-phase flow coming out of the upper plenum will flow through the intact loop steam generators and continue on to the intact loop cold legs where most (if not all) of the steam can be condensed by safety injection.

The flow path resistance is significantly affected by loop seal plugging. Loop seal plugging occurs as a result of the eventual filling of the vessel; therefore, it is not practical to adjust model input values to try to prevent loop seal plugging from occurring. As the downcomer fills and the overall liquid levels in the system reach the elevation of the nozzles, water from the safety injection system is able to flow into the intact loop pump suction legs. The loop seal plugging phenomenon is predicted to occur late in the WC/T LOCA M&E calculation (after 3000 seconds).

Loop seal plugging has no effect on the calculated peak containment pressure and temperature for a large dry containment because the peak typically occurs during the initial blowdown phase of the DEHL LOCA

event. However, loop seal plugging can reduce the stored energy release rates from the intact loop steam generators during the long term decay heat removal phase of the transient. When a loop seal becomes plugged, the steam and entrained liquid droplets that exit the upper plenum can no longer vent through the affected loop. This prevents reverse heat transfer from the secondary fluid of the affected steam generator and reduces the rate of energy transfer to containment, which affects the calculated long term containment pressure and temperature response.

The containment peak pressure for plants with the ice condenser containment design is dependent on the capability of the spray system to condense the steam that is released to containment just after most of the ice melts. Typically, it takes several hours to melt most of the ice that is contained within the ice bays. An earlier time of ice melt yields a higher peak pressure. Therefore, loop seal plugging could affect the calculated peak containment pressure and temperature for the ice condenser containment design; however, as mentioned earlier, it is not practical to adjust model input values to try to prevent loop seal plugging from occurring.

The CE plant design has two cold legs and one hot leg in each reactor coolant loop. The diameter of the hot leg is much larger than the cold leg or pump suction leg.

The containment peak pressure for the CE plant design also occurs during the initial blowdown phase of the DEHL LOCA event. However, a second peak in the containment pressure is possible because the steam generators in the CE plant design are much larger than the typical Westinghouse plant design. Consequently, the amount of energy that is released from the broken loop steam generator to containment after the end of blowdown for a DECL or DEPS LOCA event is much larger. The second peak occurs during, or just after, the reflood phase, so the loop flow split, and its effect on the containment pressure response, is not affected by loop seal plugging. The magnitude of the second peak is determined by using the output from the DECL and DEPS LOCA M&E release cases in the corresponding containment response calculation.

2.14 ADS-4 OPERATION

The phenomena associated with Stage 4 of the Automatic Depressurization System (ADS-4) operation are applicable to the passive plant design. The LOCA M&E release calculation methodology for the passive plant design is described in a separate document that is not a port of this topical report.

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Figure 2-1 ECCS Temperature Sensitivity Break Flow Rate Comparison

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Figure 2-2 ECCS Temperature Sensitivity Break Energy Comparison

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Figure 2-3 ECCS Temperature Sensitivity Containment Pressure Comparison

3 WCOBRA/TRAC CODE UPDATES FOR LOCA M&E

3.1 OVERVIEW OF WC/T CODE MODIFICATIONS

The approved PWR ECCS evaluation model (Reference 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 LOCA mass and energy release input data for the containment response calculations.

The heat transfer model in the WC/T ECCS evaluation model has been shown to over-predict the reverse SG heat transfer when compared with experimental data. Although this is conservative, a more realistic SG heat transfer option has been added to WC/T to improve the M&E release calculation.

The WC/T ECCS evaluation model does not consider the metal energy of the SG inlet and outlet plenums, the steam separators and driers, and the secondary side shell. Modeling of this metal energy has been included as an option in WC/T for the M&E release calculation.

The containment response calculation can be performed in parallel with the LOCA M&E release calculation using the GOTHIC code (References 24 through 26). WC/T has been modified to allow it to run in parallel with GOTHIC.

3.1.1 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 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 the significant temperature stratification seen in the experiments (see Figures 4 through 11 in Reference 11).

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Model Bases – Saturated Droplet Flow

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]a,c According to several authors (References 12, 13, and 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 a "shielding" effect.

Webb and Chen (Reference 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 off of the heated wall in the vicinity of the critical heat flux (CHF) point.
- A far-field 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 – Saturated Droplet Flow

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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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Model as Coded – Subcooled 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. The dispersed two-phase flow above the quench front provided enough heat transfer and precursory wall cooling so that the quench front advanced up the tubes with time. The abrupt drop in the temperature at a 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. [

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According to Collier (Reference 21, Section 4.5.2), the heat transfer processes in the vicinity of the quench front are characterized by two main heat transfer regimes separated by the liquid film dryout location. Downstream of the quench front, and before the dry saturated vapor region, there is a flow pattern characterized by a liquid deficient region, where dispersed flow heat transfer takes place, and an interfacial heat transfer correlation which can deal with nonequilibrium superheated steam conditions needs to be applied. Upstream of the quench front 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 energy is transferred by forced convection in the film to the liquid-vapor core interface, where evaporation takes place. This 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 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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Model as Coded – Quench Front

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Code Validation

See Section 3.2.

3.1.2 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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Model as Coded

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3.2 COMPARISON TO FLECHT-SEASET STEAM GENERATOR TEST RESULTS

The FLECHT-SEASET Steam Generator Separate Effect tests (Reference 9) were conducted in 1982. The test facility consisted of a full height U-tube steam generator, boiler, accumulator, and containment tank. The steam generator tube height and dimensions in the test facility are typical of Westinghouse series 51 steam generators. A total of 32 of 33 U-tubes were used. The boiler and accumulator supplied steam and water to a mixing chamber, which generated 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 seven FLECHT-SEASET test cases that were selected for comparison are listed in Table 3-1. 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 Reference 9 test data shows that increasing the flow rate (23402) or reducing the quality (22314, 21806, and 21909) causes the steam generator secondary side to cool faster.

The WC/T simulation model noding structure that was used to represent the FLECHT-SEASET tests is shown in Figure 3-1. [

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

Figures 3-2 through 3-43 show a comparison of FLECHT-SEASET test data with results calculated using the standard WC/T ECCS evaluation model options (curves identified as WC/T Standard) and the LOCA M&E model options (curves identified as WC/T w/Interfacial HTX and SG HTX). The WC/T LOCA M&E model options typically show a marked improvement in the calculation of the steam generator outlet temperature and the calculation of the quench front. All cases under-predict the timing of the quench front, which is conservative for the LOCA M&E release calculations because it over-predicts the energy removal rate from the SG secondary side.

Figures 3-44 through 3-50 compare the SG secondary side temperatures. The values calculated using the WC/T LOCA M&E model options are typically lower than the test data.

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Table 3-1 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 Temperature (K)	406	410	388	406	406	407	406
Initial Void Fraction	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Initial Secondary 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 3-1 WC/T Simulation Model Noding Structure for SG FLECHT-SEASET Tests

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Figure 3-2 FLECHT-SEASET Test 22701 SG Outlet Temperature

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Figure 3-3 FLECHT-SEASET Test R22701 SG Tube Wall – 1 ft

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Figure 3-4 FLECHT-SEASET Test R22701 SG Tube Wall – 4 ft

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Figure 3-5 FLECHT-SEASET Test R22701 SG Tube Wall – 10 ft

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Figure 3-6 FLECHT-SEASET Test R22701 SG Heat Release Rate

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Figure 3-7 FLECHT-SEASET Test R22701 SG Exit Quality

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Figure 3-8 FLECHT-SEASET Test R23402 SG Outlet Temperature

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Figure 3-9 FLECHT-SEASET Test R23402 SG Tube Wall – 1 ft

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Figure 3-10 FLECHT-SEASET Test R23402 SG Tube Wall – 4 ft

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Figure 3-11 FLECHT-SEASET Test R23402 SG Tube Wall – 10 ft

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Figure 3-12 FLECHT-SEASET Test R23402 SG Heat Release Rate

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Figure 3-13 FLECHT-SEASET Test R23402 SG Exit Quality

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Figure 3-14 FLECHT-SEASET Test R22503 SG Outlet Temperature

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Figure 3-15 FLECHT-SEASET Test R22503 SG Tube Wall – 1 ft

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Figure 3-16 FLECHT-SEASET Test R22503 SG Tube Wall – 4 ft

a,c

Figure 3-17 FLECHT-SEASET Test R22503 SG Tube Wall – 10 ft

a,c

Figure 3-18 FLECHT-SEASET Test R22503 SG Heat Release Rate

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Figure 3-19 FLECHT-SEASET Test R22503 SG Exit Quality

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Figure 3-20 FLECHT-SEASET Test R22314 SG Outlet Temperature

a,c

Figure 3-21 FLECHT-SEASET Test R22314 SG Tube Wall – 1 ft

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Figure 3-22 FLECHT-SEASET Test R22314 SG Tube Wall – 4 ft

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Figure 3-23 FLECHT-SEASET Test R22314 SG Tube Wall – 10 ft

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Figure 3-24 FLECHT-SEASET Test R22314 SG Heat Release Rate

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Figure 3-25 FLECHT-SEASET Test R22314 SG Exit Quality

a,c

Figure 3-26 FLECHT-SEASET Test R21806 SG Outlet Temperature

a,c

Figure 3-27 FLECHT-SEASET Test R21806 SG Tube Wall – 1 ft

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Figure 3-28 FLECHT-SEASET Test R21806 SG Tube Wall – 4 ft

a,c

Figure 3-29 FLECHT-SEASET Test R21806 SG Tube Wall – 10 ft

a,c

Figure 3-30 FLECHT-SEASET Test R21806 SG Heat Release Rate

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Figure 3-31 FLECHT-SEASET Test R21806 SG Exit Quality

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Figure 3-32 FLECHT-SEASET Test R21909 SG Outlet Temperature

a,c

Figure 3-33 FLECHT-SEASET Test R21909 SG Tube Wall – 1 ft

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Figure 3-34 FLECHT-SEASET Test R21909 SG Tube Wall – 4 ft

a,c

Figure 3-35 FLECHT-SEASET Test R21909 SG Tube Wall – 10 ft

a,c

Figure 3-36 FLECHT-SEASET Test R21909 SG Heat Release Rate

a,c

Figure 3-37 FLECHT-SEASET Test R21909 SG Exit Quality

a,c

Figure 3-38 FLECHT-SEASET Test R22920 SG Outlet Temperature

a,c

Figure 3-39 FLECHT-SEASET Test R22920 SG Tube Wall – 1 ft

a,c

Figure 3-40 FLECHT-SEASET Test R22920 SG Tube Wall – 4 ft

a,c

Figure 3-41 FLECHT-SEASET Test R22920 SG Tube Wall – 10 ft

a,c

Figure 3-42 FLECHT-SEASET Test R22920 SG Heat Release Rate

a,c

Figure 3-43 FLECHT-SEASET Test R22920 SG Exit Quality

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Figure 3-44 R22701 Steam Generator Secondary Fluid Temperatures

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Figure 3-45 R23402 Steam Generator Secondary Fluid Temperatures

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Figure 3-46 R22503 Steam Generator Secondary Fluid Temperatures

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Figure 3-47 R22314 Steam Generator Secondary Fluid Temperatures

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Figure 3-48 R21806 Steam Generator Secondary Fluid Temperatures

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Figure 3-49 R21909 Steam Generator Secondary Fluid Temperatures

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Figure 3-50 R22920 Steam Generator Secondary Fluid Temperatures

3.3 WC/T LOCA M&E RELEASE OUTPUT DATA

The WC/T LOCA M&E model output data provides input for a containment response model. The output data can be calculated by running the WC/T LOCA M&E model stand-alone or by executing WC/T in parallel with GOTHIC (see Section 3.4).

The WC/T LOCA M&E model output data is provided in a set of transient Mass – Energy release data tables. The mass-energy release data tables are provided for both ends of a double-ended guillotine break. The flow rate and enthalpy values that are provided in the data tables are written at user specified time intervals. These user specified time intervals may contain a large number of WC/T time step calculations. Therefore, the values provided in the data tables must be calculated such that the integrated WC/T mass and energy releases are conserved over each interval.

Code Implementation

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Figure 3-51 Calculation of Average M&E Release Rate in Specified Time Intervals

Model as Coded

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Code Validation

The WC/T LOCA M&E release output data files were verified by directly comparing the plots generated using the output data files with the WC/T time step data that is saved in the NSAPLOT output files.

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Figure 3-52 Implementation of the Containment Related Calculations in WC/T Logic

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Figure 3-53 Logic of the Containment Related Calculations in WC/T

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Figure 3-54 Containment Related Data Transfer to M&E Files or to GOTHIC

3.4 WCOBRA/TRAC RUNNING IN PARALLEL WITH GOTHIC

The containment pressure, temperature, and sump temperature response during a LOCA are dependent on the LOCA mass and energy releases. The LOCA mass and energy releases are dependent on the containment pressure and on the sump temperature when the residual heat removal 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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Model as Coded

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Code Validation

The interface transfer between WC/T and GOTHIC was validated by comparing the interface variable plots from the WC/T and GOTHIC sides. The results coincided identically.

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Figure 3-55 Schematic of the GOTHIC – WC/T Coupling

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Figure 3-56 Schematic of the GOTHIC – WC/T Parallel Execution

4 INPUT BIASING FOR THE CONTAINMENT DBA ANALYSES

As described in Section 1, several simplifying assumptions were made during the development of the currently approved LOCA M&E release methodology. Some of these assumptions are to be removed in the proposed new LOCA M&E release methodology as described below.

1. The new WC/T LOCA M&E release model can be coupled with a GOTHIC containment model to calculate the containment response into the post-reflood phase of the event. When running in this manner, it is not necessary to assume the containment backpressure remains at a constant value during blowdown or define conservative containment backpressure input values during the reflood and post-reflood phases. For large dry containment calculations, a reasonable containment backpressure will continue to be used if the WC/T LOCA M&E model is not coupled to a containment response model. For standalone ice condenser WC/T calculations, a bounding high backpressure will be used. The SG fluid, metal, and RCS metal energy remaining at the end of the WC/T calculation will be released, along with the decay heat, in the long-term containment response calculation.

2. The assumption that the vessel is automatically refilled to the bottom of the fuel at the end of the blowdown phase (just prior to reflood) is conservative, but 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 PIRT was developed to identify the important phenomena that need to be considered in the calculation (Reference 23, updated in Section 2). Next, an appropriate code was selected for the LOCA M&E release model. The Westinghouse best-estimate LOCA ECCS evaluation model uses the WC/T code (Reference 10) to calculate the RCS thermal-hydraulic response to a large pipe rupture. The LOCA ECCS evaluation model PIRT (Reference 29, Section 1.2.3) 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 LOCA M&E PIRT items (see Sections 2 and 3). The changes to the WC/T code were validated by comparison with SG test data from FLECHT. Finally, the calculated transient response from the proposed WC/T LOCA M&E release methodology was compared with the calculated transient response from the current LOCA M&E release methodology (see Sections 5, 6, and 7).

The LOCA 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. This document 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 review guidance provided in NUREG-0800, Section 6.2.1.3 is shown in Table 4-1. 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 4-2.

Table 4-1 NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

a,c

	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 4-1 NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations (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.		

a,c

[ ]^{a,c}

Table 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

a,c

	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 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

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 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

	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.		

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Table 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

	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.		

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Table 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

	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.		

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Table 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

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.		

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Table 4-1 (cont.) NUREG-0800, Section 6.2.1.3 Review Guidance for LOCA M&E Release Calculations

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	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 4-2 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 Metal* – 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.		

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Table 4-2 (cont.) ANS 56.4-1983 Recommendations

a,c

	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.		

Table 4-2 (cont.) ANS 56.4-1983 Recommendations

a,c

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.		

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Table 4-2 (cont.) ANS 56.4-1983 Recommendations

a,c

	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).		

Table 4-2 (cont.) ANS 56.4-1983 Recommendations

	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.		

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Table 4-2 (cont.) ANS 56.4-1983 Recommendations

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 4-2 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 4-2 (cont.) ANS 56.4-1983 Recommendations

	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.		

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Table 4-2 (cont.) ANS 56.4-1983 Recommendations

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 4-2 (cont.) ANS 56.4-1983 Recommendations

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 4-2 (cont.) ANS 56.4-1983 Recommendations

a,c

	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.		

Table 4-2 (cont.) ANS 56.4-1983 Recommendations

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 4-2 (cont.) ANS 56.4-1983 Recommendations

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.		

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Table 4-2 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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4.1 BIASING FOR PEAK PRESSURE/TEMPERATURE

The following changes must be made to bias a WC/T ECCS evaluation model input deck to calculate conservative LOCA M&E releases for the peak containment pressure and temperature calculation:
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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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4.2 BIASING FOR LONG-TERM EQ APPLICATION

The WC/T LOCA M&E model is used to calculate the break releases until both sides of the break reach saturation, i.e., there is no superheated steam release or until after the containment peak pressure is predicted to occur (whichever is longer). 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 boil-off calculation.

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4.3 BIASING FOR MINIMUM NPSHA APPLICATION

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5 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.

5.1 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 5-1. The modified WC/T steam generator noding diagram is shown in Figure 5-2.

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 5-1. The WC/T model has a slightly higher initial RCS fluid mass and lower energy than SATAN-VI. The WC/T model SG metal energy is also substantially higher than SATAN-VI. 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 5-1 Initial Steady State Mass and Energy Comparison

	WC/T Model	SATAN-VI Model
RCS Fluid Mass	574,800 lbm	567,400 lbm
RCS Fluid Energy	345 MBtu	347 MBtu
RCS Metal Energy	156 MBtu	114 MBtu
SG Secondary Fluid Mass	538,100 lbm	546,500 lbm
SG Secondary Fluid Energy	313 MBtu	313 MBtu
SG Secondary Metal Energy	96 MBtu	78 MBtu

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Figure 5-1 WC/T Steady State Noding Diagram (4-Loop Plant)

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Figure 5-2 WC/T Steam Generator Noding Structure for LOCA M&E

5.2 CONTAINMENT MODEL DESCRIPTION

The GOTHIC 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 2.76×10^6 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 5-3 and the key containment model input is given in Tables 5-2 and 5-3. [

]^{a,c} The fan cooler heat removal rate is input as a function of the containment saturation temperature as shown in Figure 5-4. [
]^{a,c}

The GOTHIC containment model runs in parallel 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 5-2 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 5-3	
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 5-4
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 5-3 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	1					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

a,c

Figure 5-3 GOTHIC Containment Model Noding Diagram

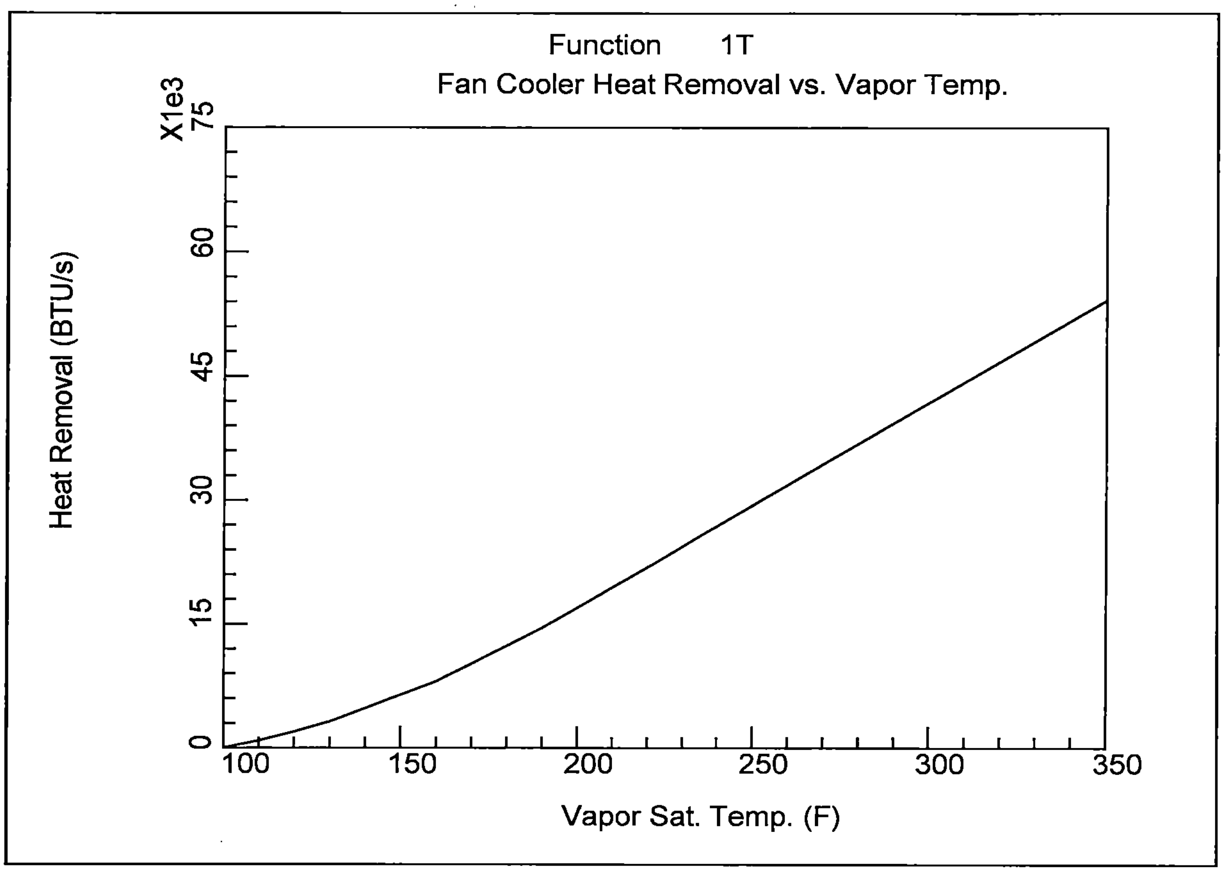

Figure 5-4 Fan Cooler Heat Removal Curve

5.3 DEPS LOCA BENCHMARK CASE RESULTS COMPARISON

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

]^{a,c}

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-P-A (Reference 2) benchmark case through sump recirculation. The integrated blowdown break mass and energy release comparison is shown in Figures 5-5 and 5-6. 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 5-7 and 5-8. 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 effect of the lower metal and SG energy release rates on the GOTHIC calculated containment pressure and temperature is shown in Figures 5-9 through 5-14. The blowdown peak pressure and temperature are about the same because the energy release rate is the same. However, 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-P-A LOCA M&E release model.

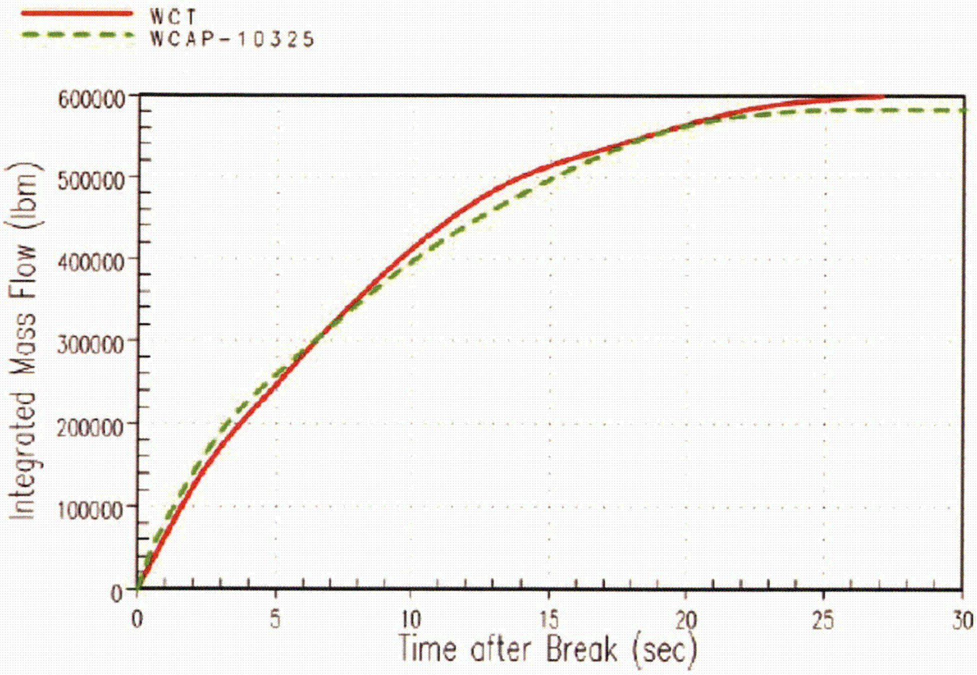

Figure 5-5 Integrated Blowdown Break Mass Release Comparison

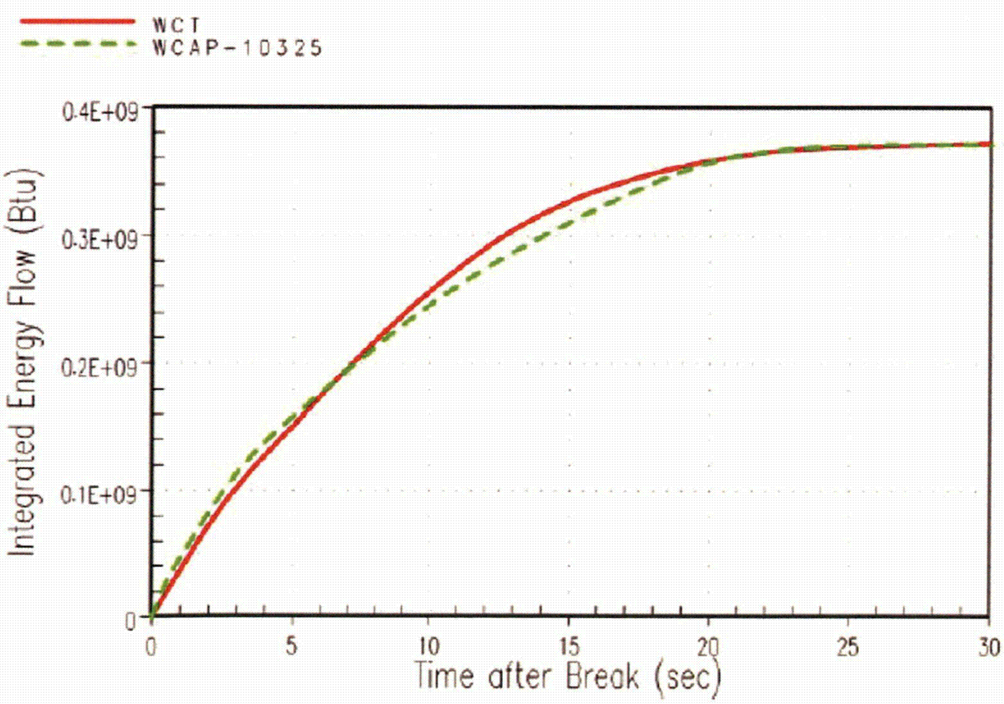

Figure 5-6 Integrated Blowdown Break Energy Release Comparison

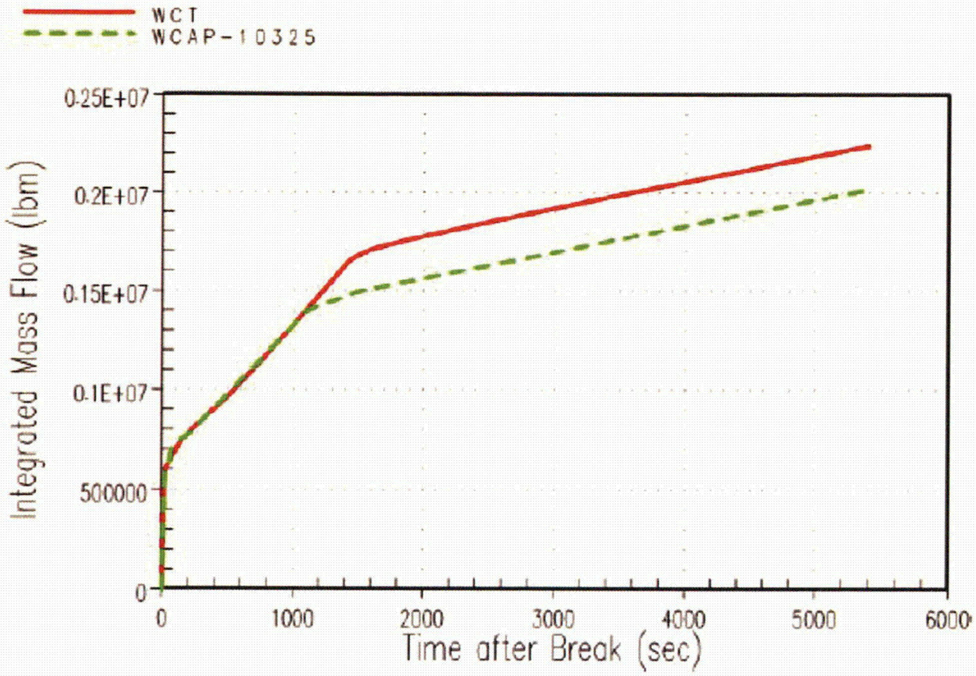

Figure 5-7 Integrated Long-term Break Mass Release Comparison

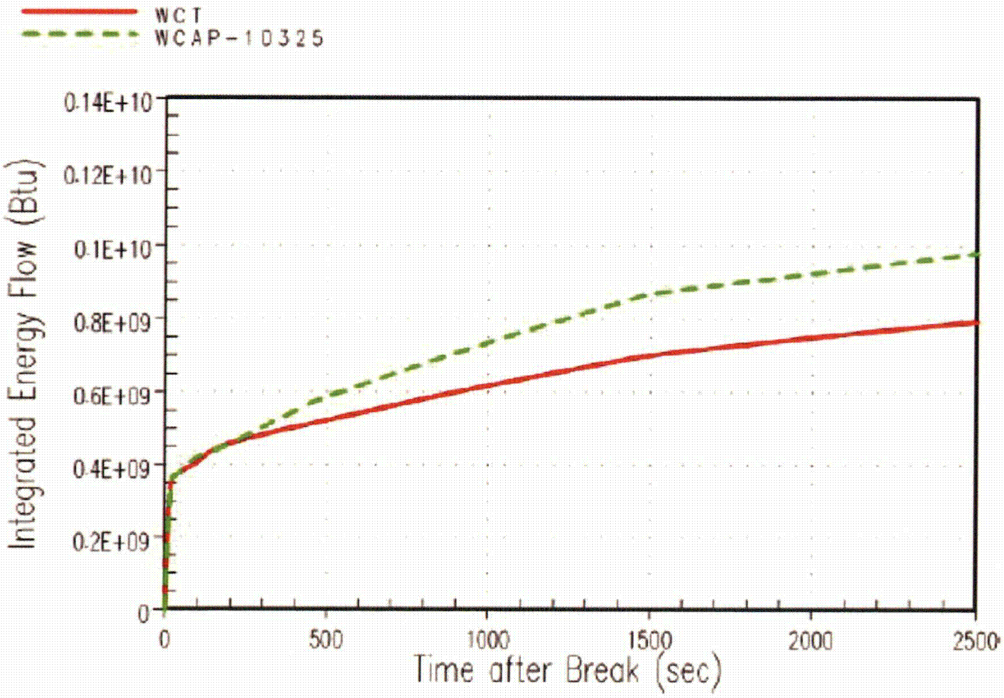

Figure 5-8 Integrated Long-term Break Energy Release Comparison

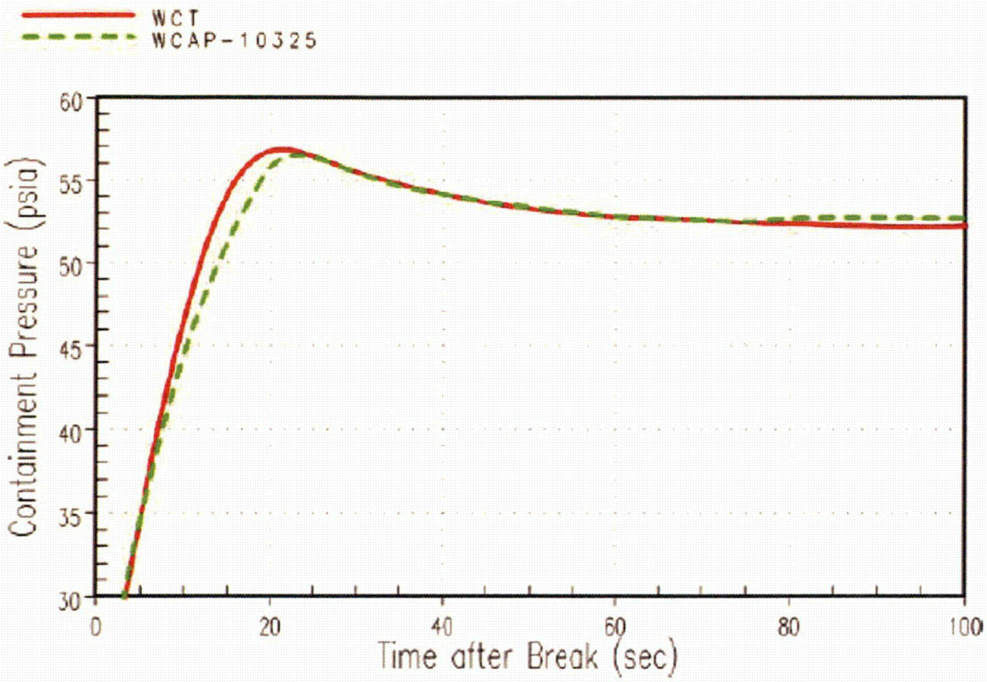

Figure 5-9 Blowdown Containment Pressure Comparison

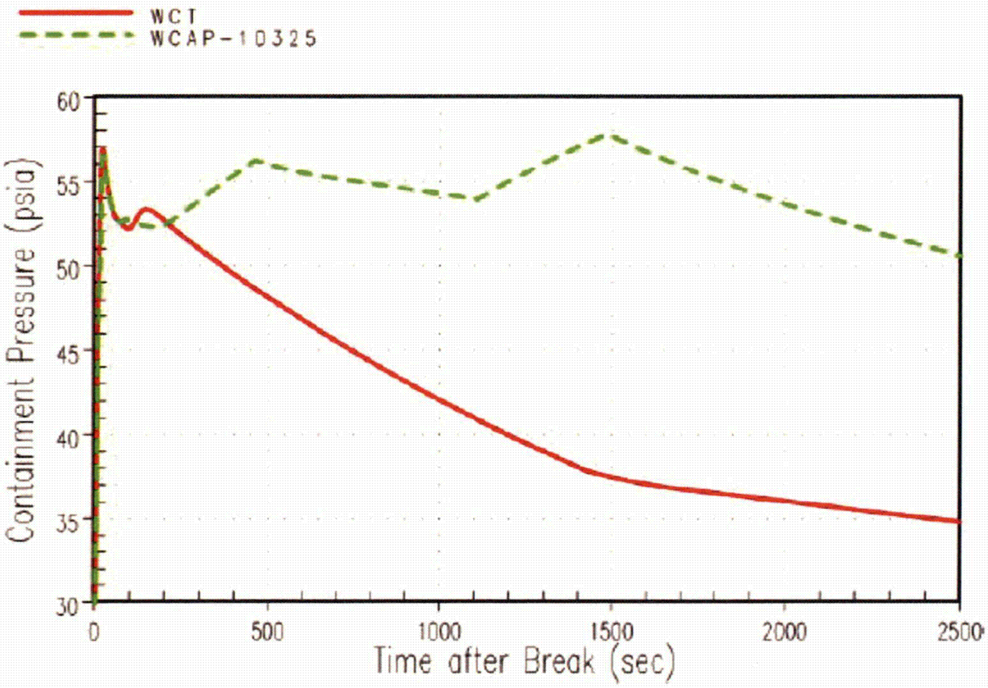

Figure 5-10 Long-term Containment Pressure Comparison

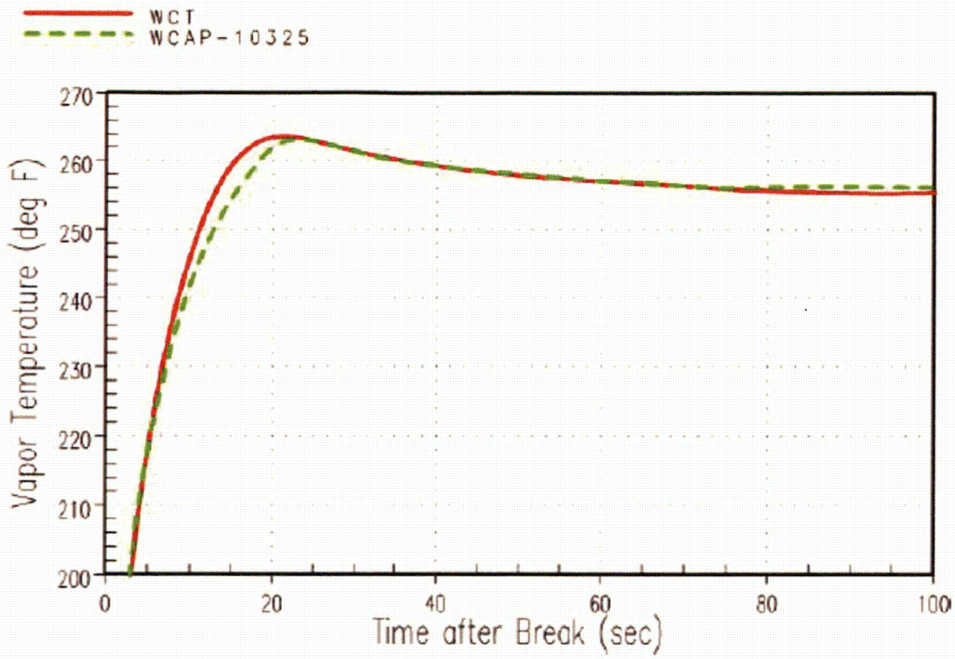

Figure 5-11 Blowdown Containment Vapor Temperature Comparison

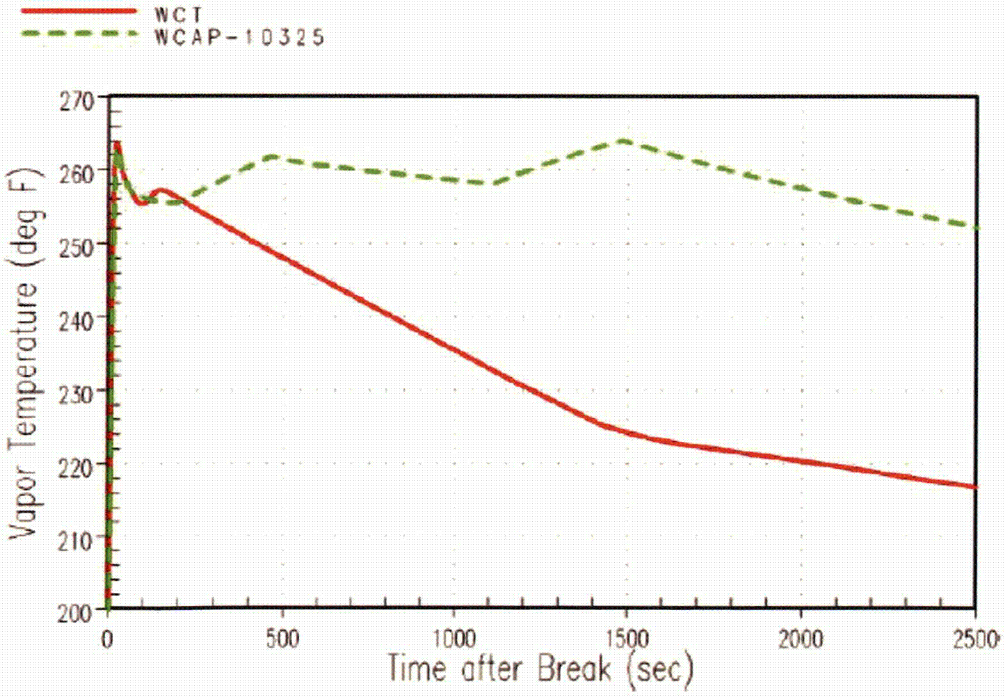

Figure 5-12 Long-term Containment Vapor Temperature Comparison

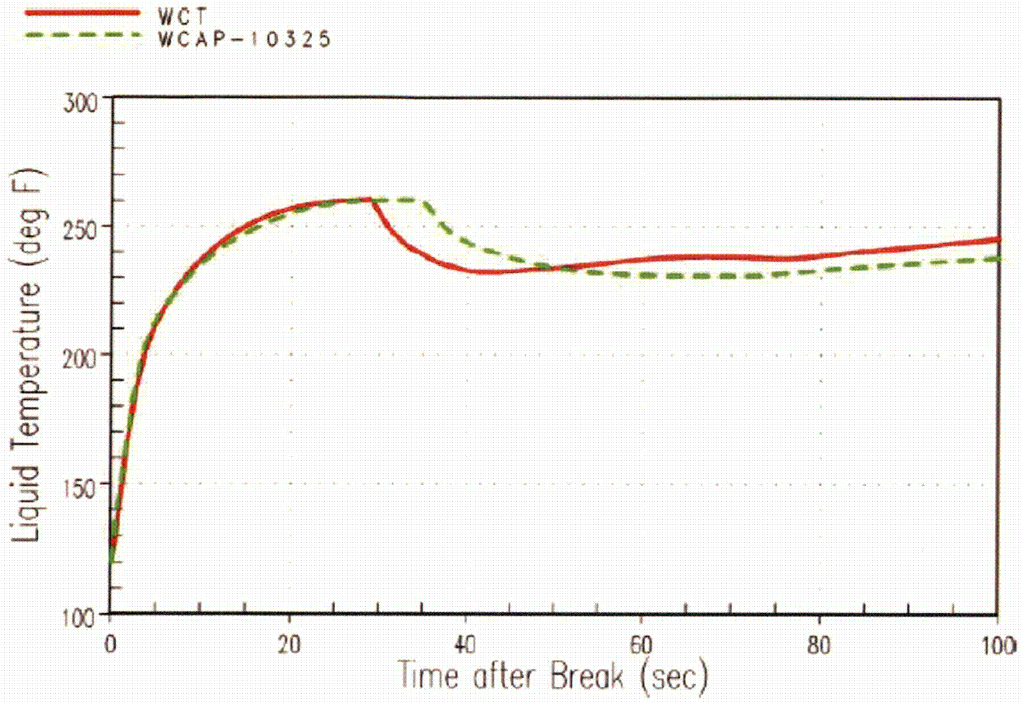

Figure 5-13 Blowdown Containment Sump Temperature Comparison

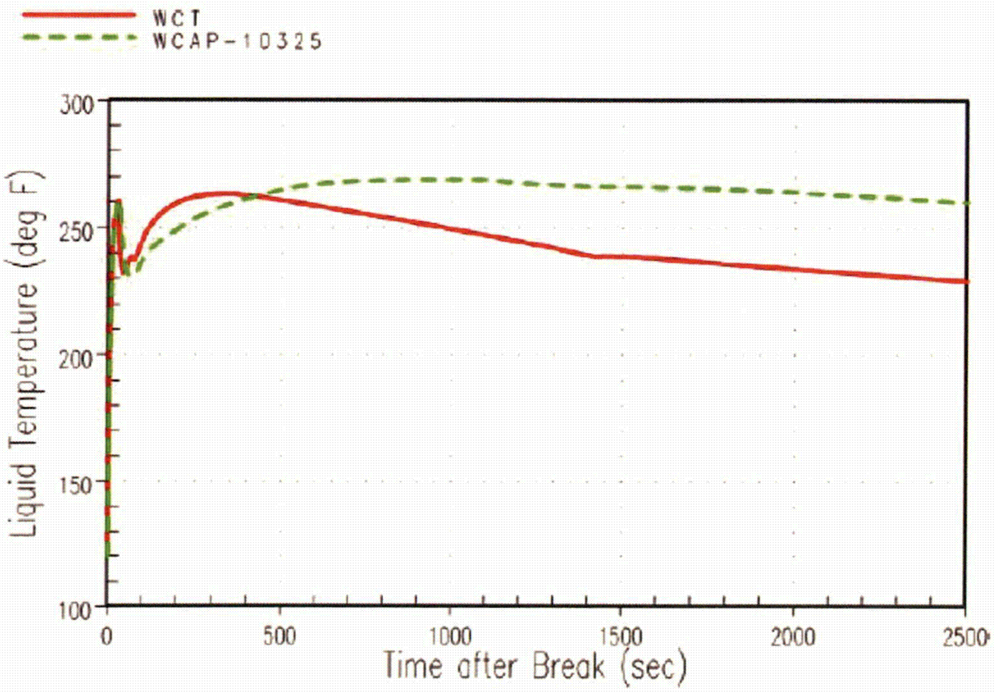

Figure 5-14 Long-term Containment Sump Temperature Comparison

5.4 DEHL LOCA BENCHMARK CASE RESULTS COMPARISON

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

]^{a,c}

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-P-A benchmark case. The integrated break mass and energy release comparison is shown in Figures 5-15 and 5-16. The WC/T model calculates a similar blowdown break mass and energy release. The containment response comparison is shown in Figures 5-17 through 5-19. The blowdown peak pressure and temperature are about the same because the energy release rate is nearly the same.

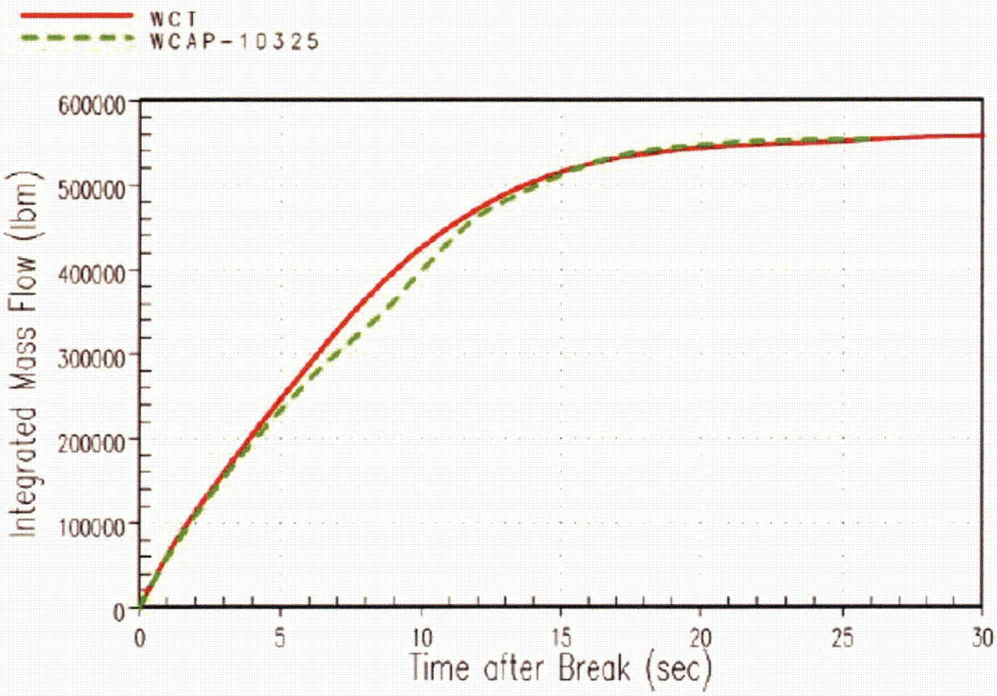

Figure 5-15 Integrated Break Flow Rate Comparison

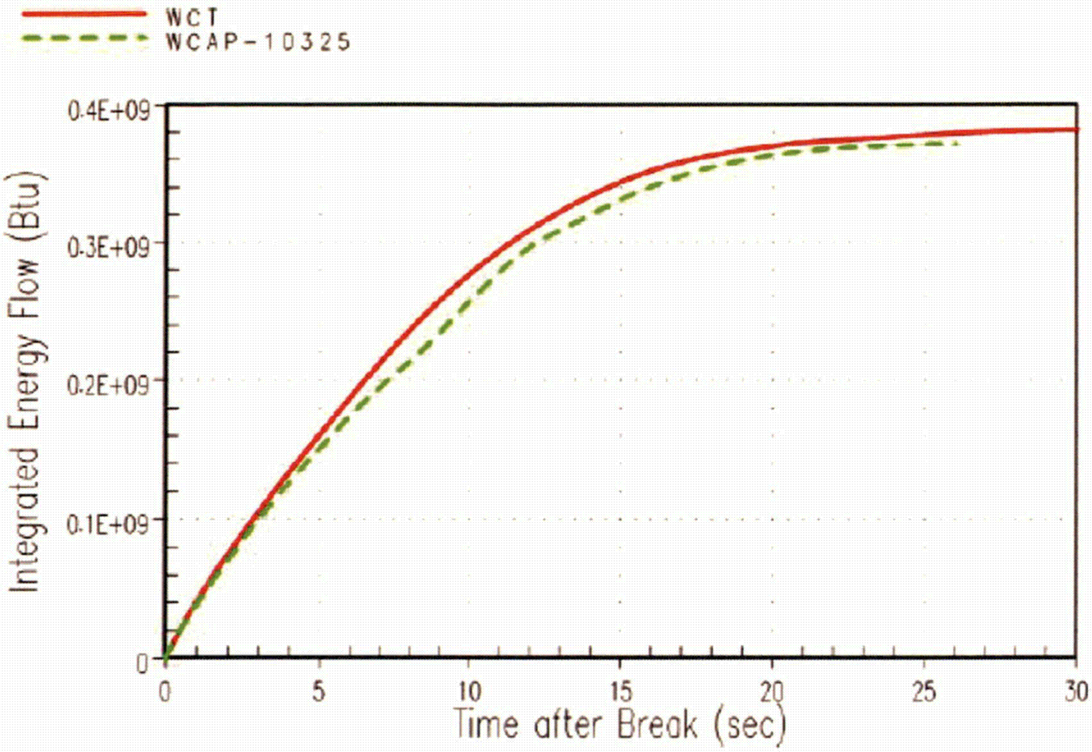

Figure 5-16 Integrated Break Energy Release Rate Comparison

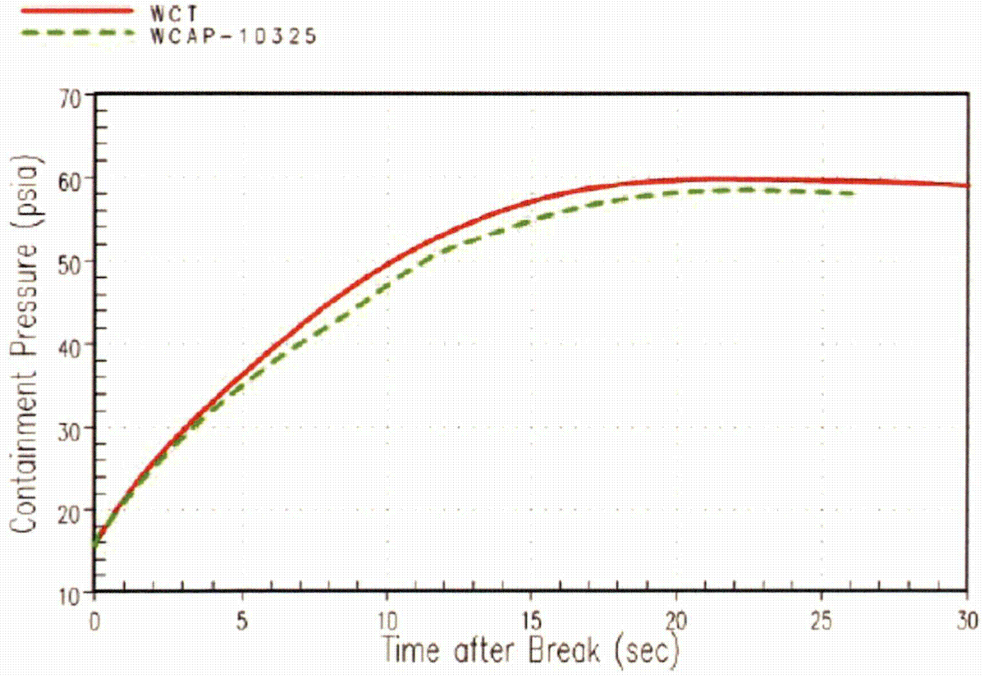

Figure 5-17 Containment Pressure Comparison

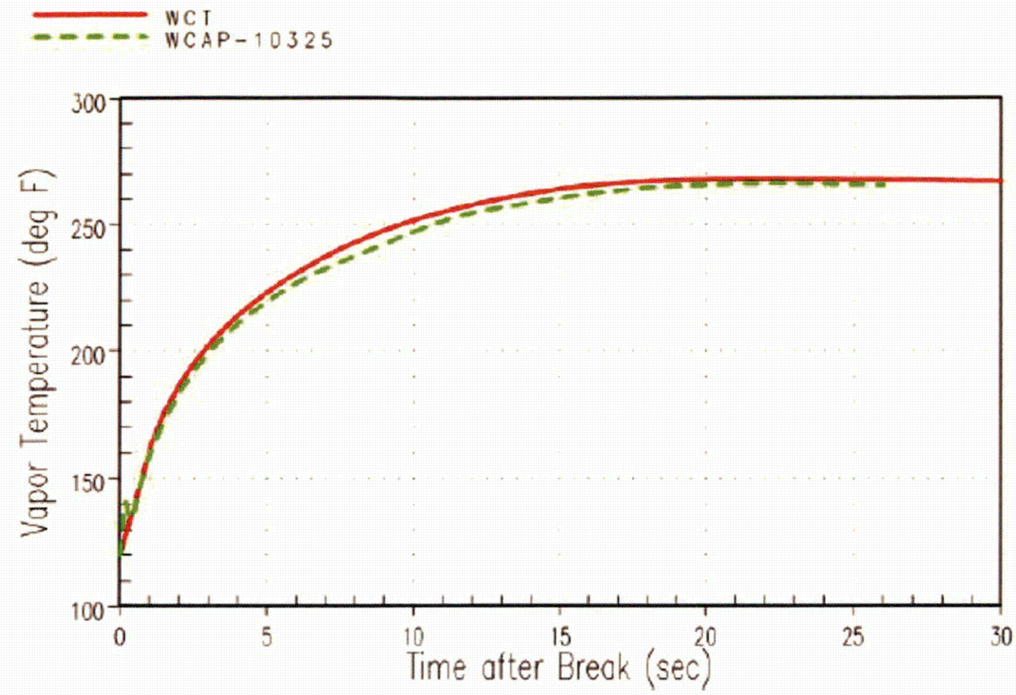

Figure 5-18 Containment Temperature Comparison

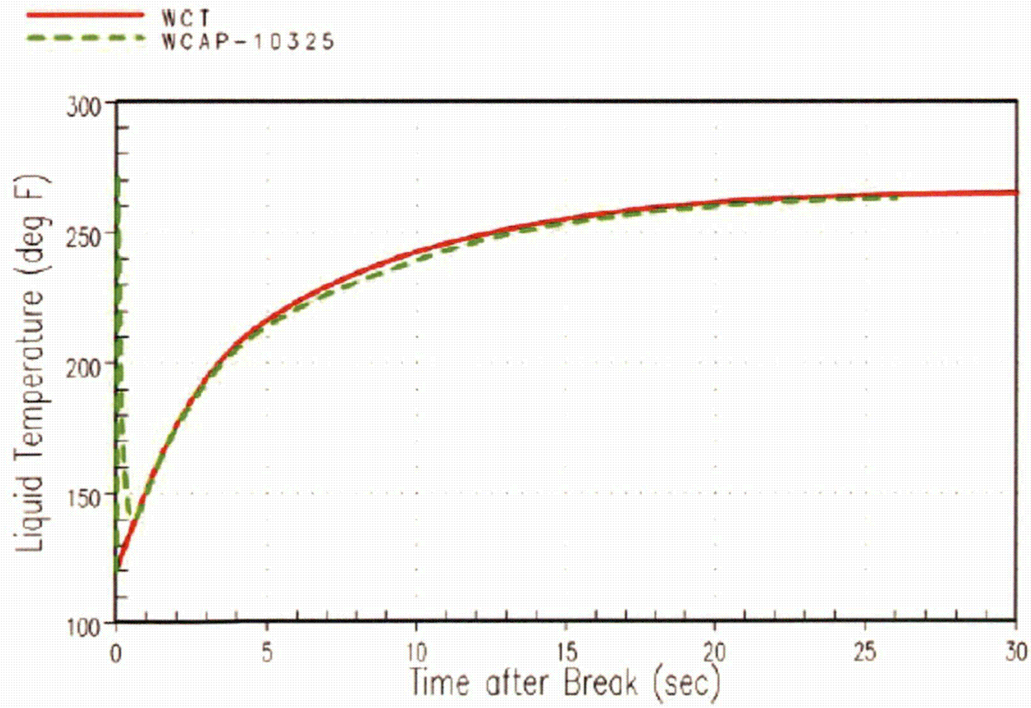

Figure 5-19 Containment Sump Temperature Comparison

6 SAMPLE CASES

The WC/T LOCA M&E release and containment model described in Section 5 was 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.

6.1 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. [

]^{a,c}

The containment pressure, temperature, and sump temperature for the three cases are compared in Figures 6-1 through 6-6. 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 6-7 and 6-8. 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 6-9 compares the average blowdown break enthalpy. The average break enthalpy for the DEHL case is higher than the others because the release is mostly steam; this causes the initial containment pressure for this case to be higher than the others.

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]^{a,c}

[

]^{a,c}

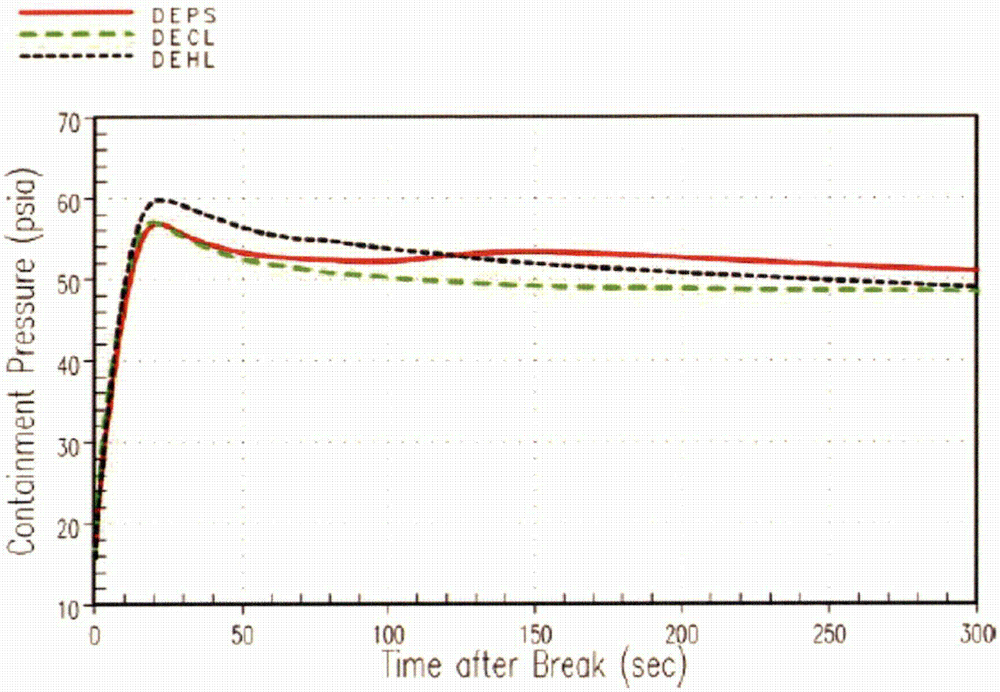

Figure 6-1 Peak Containment Pressure Comparison

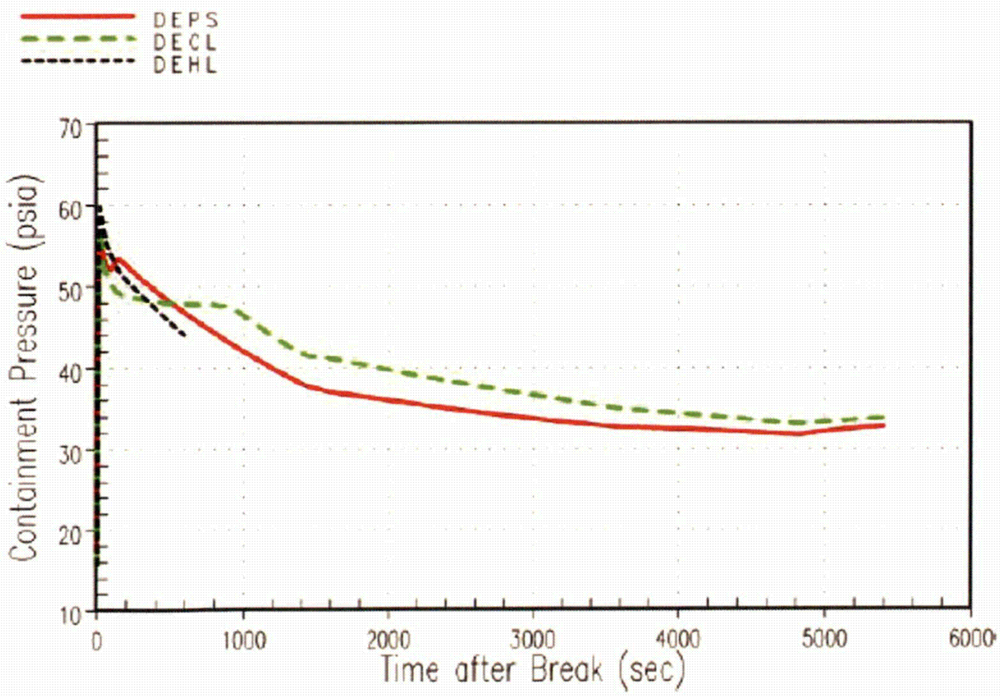

Figure 6-2 Long-term Containment Pressure Comparison

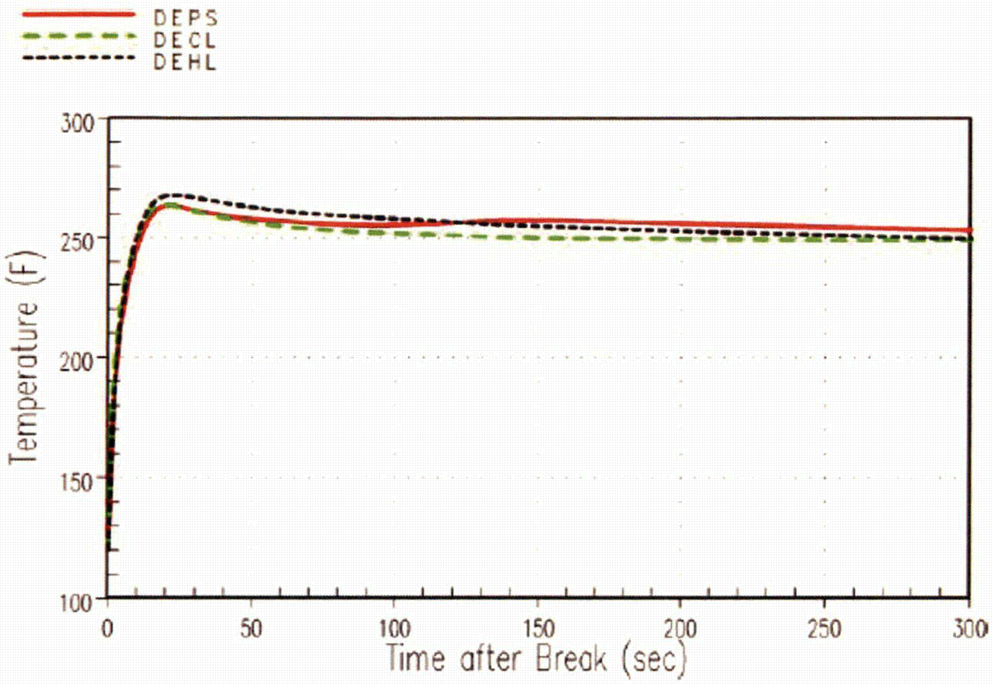

Figure 6-3 Peak Containment Temperature Comparison

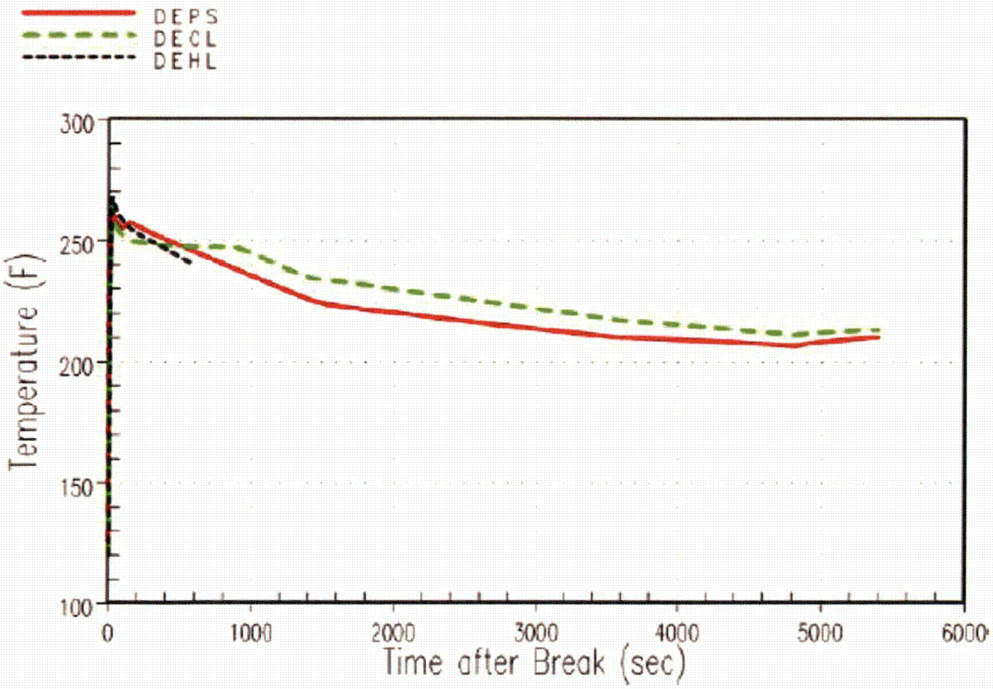

Figure 6-4 Long-term Containment Temperature Comparison

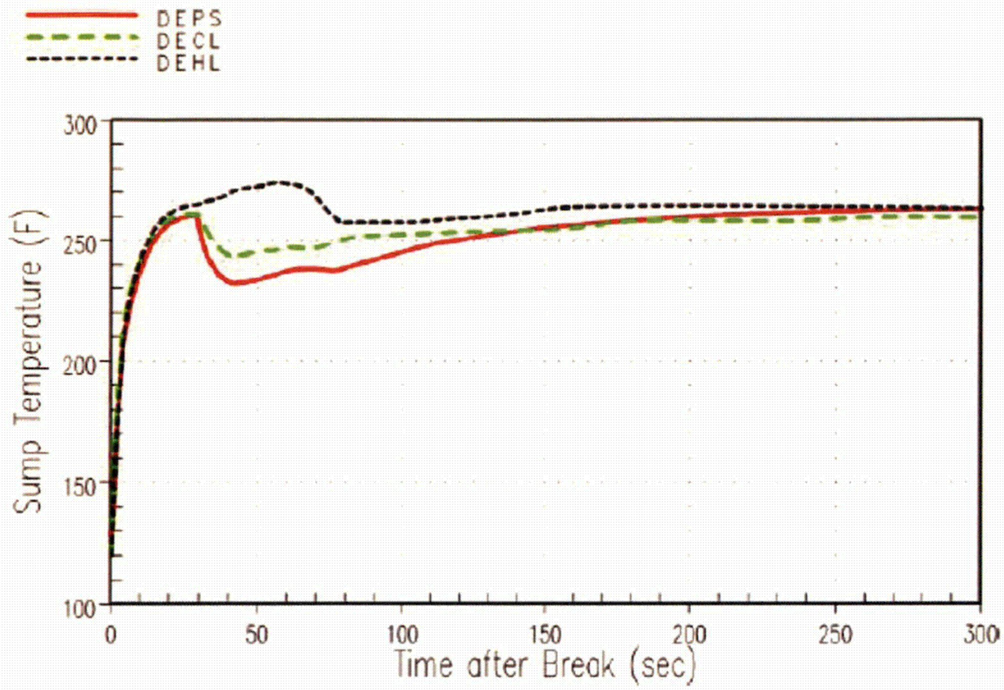

Figure 6-5 Peak Sump Temperature Comparison

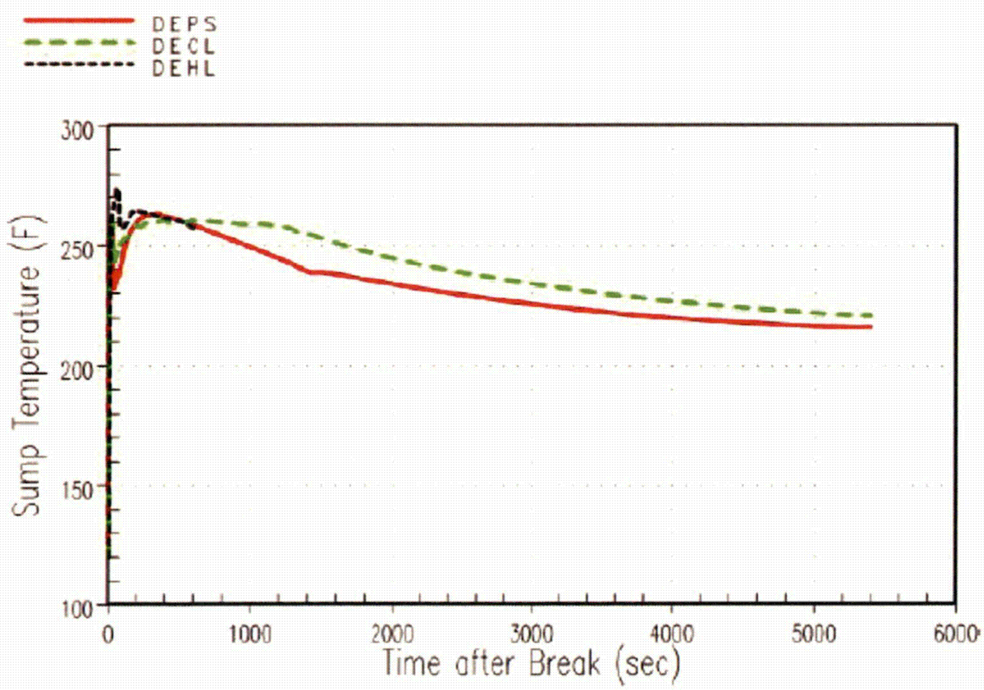

Figure 6-6 Long-term Sump Temperature Comparison

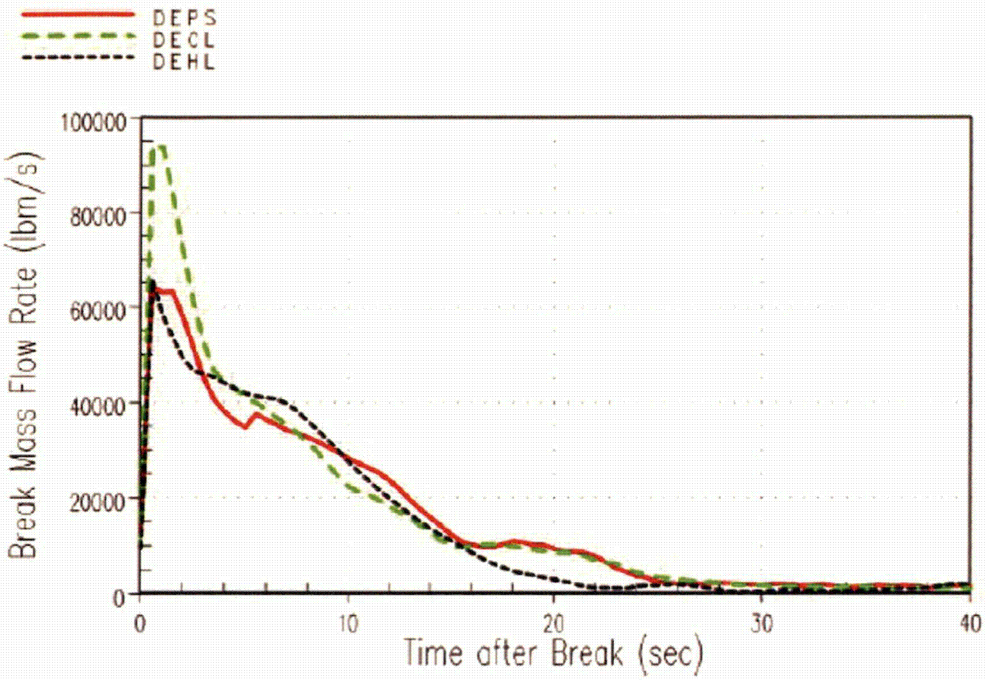

Figure 6-7 Blowdown Break Mass Flow Rate Comparison

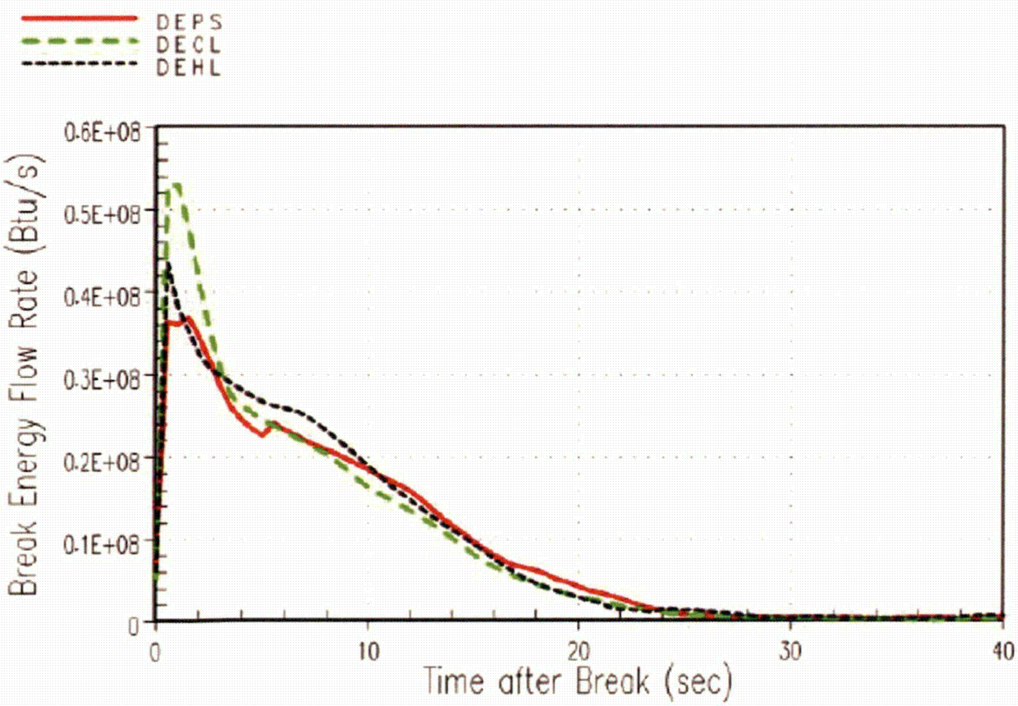

Figure 6-8 Blowdown Break Energy Flow Rate Comparison

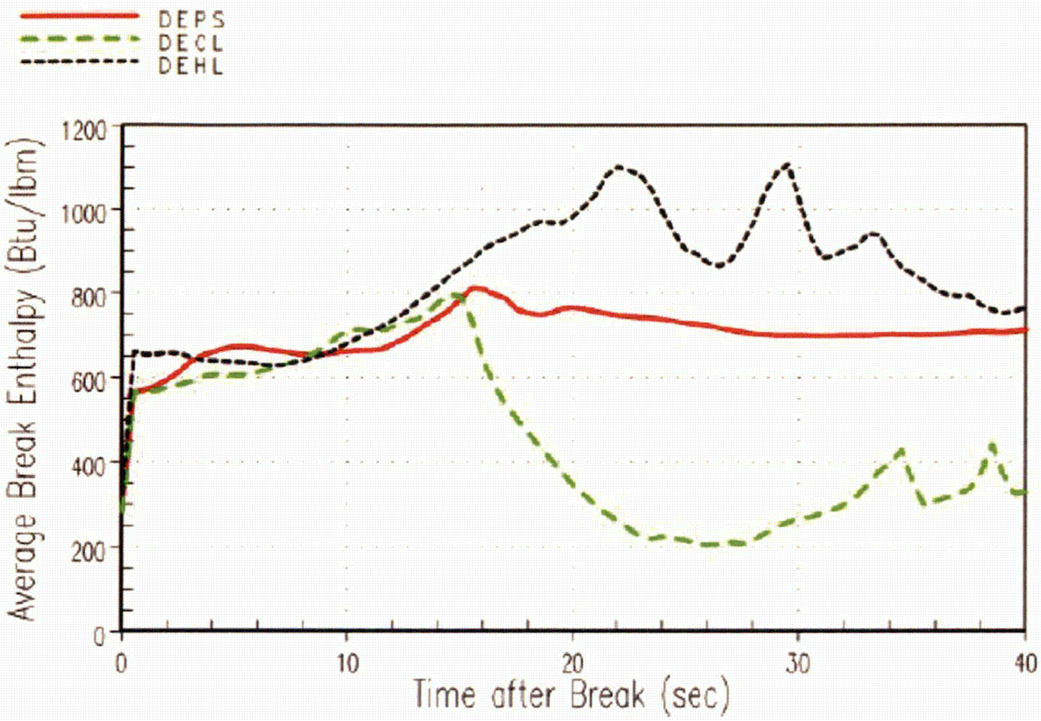

Figure 6-9 Blowdown Break Enthalpy Comparison

6.2 LONG-TERM EQ

As described in Section 4.2, the long-term LOCA steam release rate is maximized for the 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 6-10 and 6-11. 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 6-12 through 6-14. The containment pressure and temperatures decreased as the steam mass and energy release rate decreased.

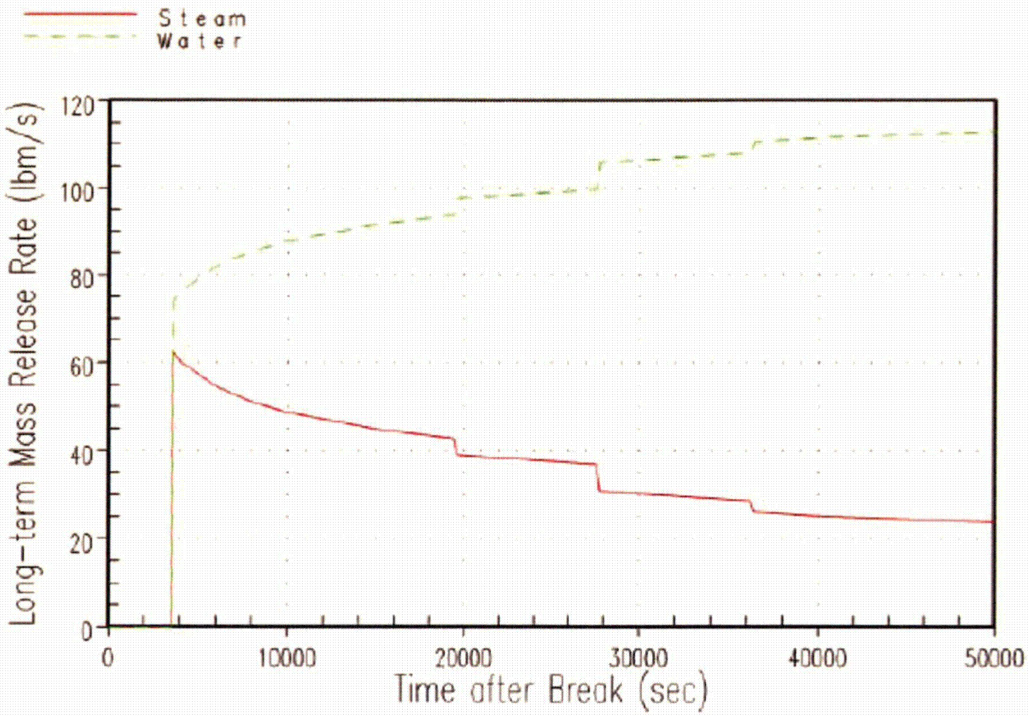

Figure 6-10 Long-term EQ Break Flow Rate

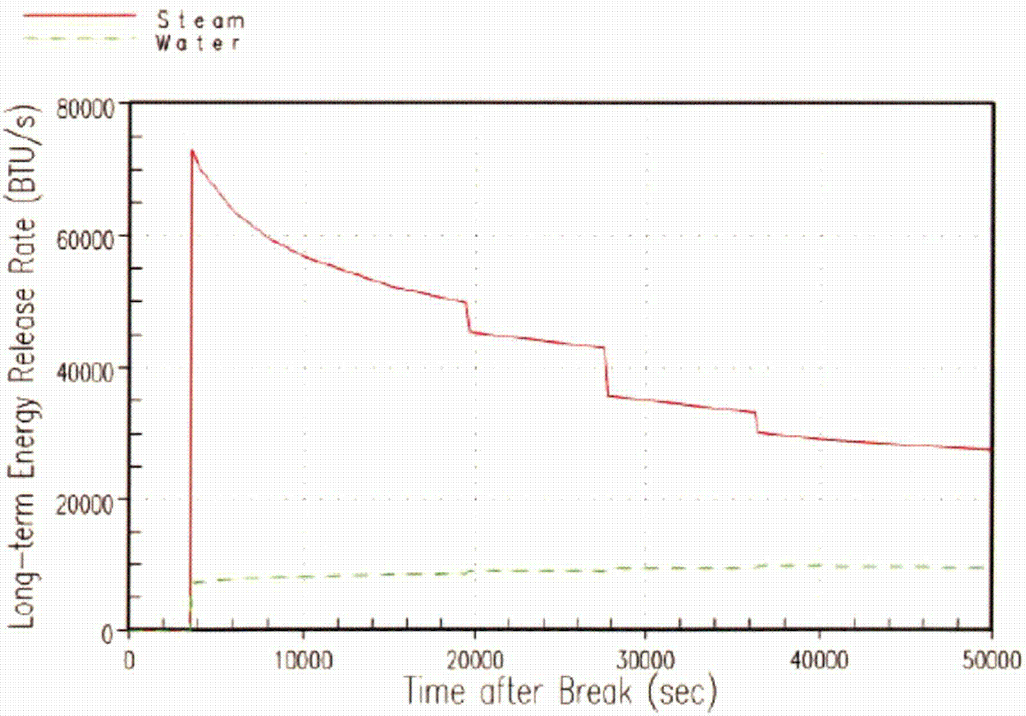

Figure 6-11 Long-term EQ Break Energy Flow Rate

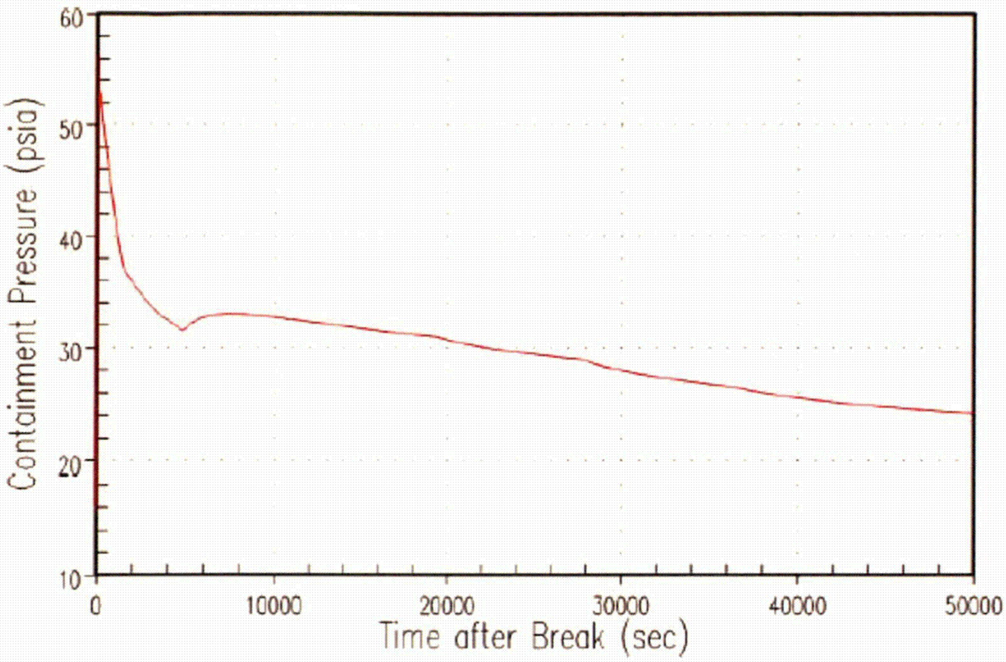

Figure 6-12 Long-term EQ Containment Pressure

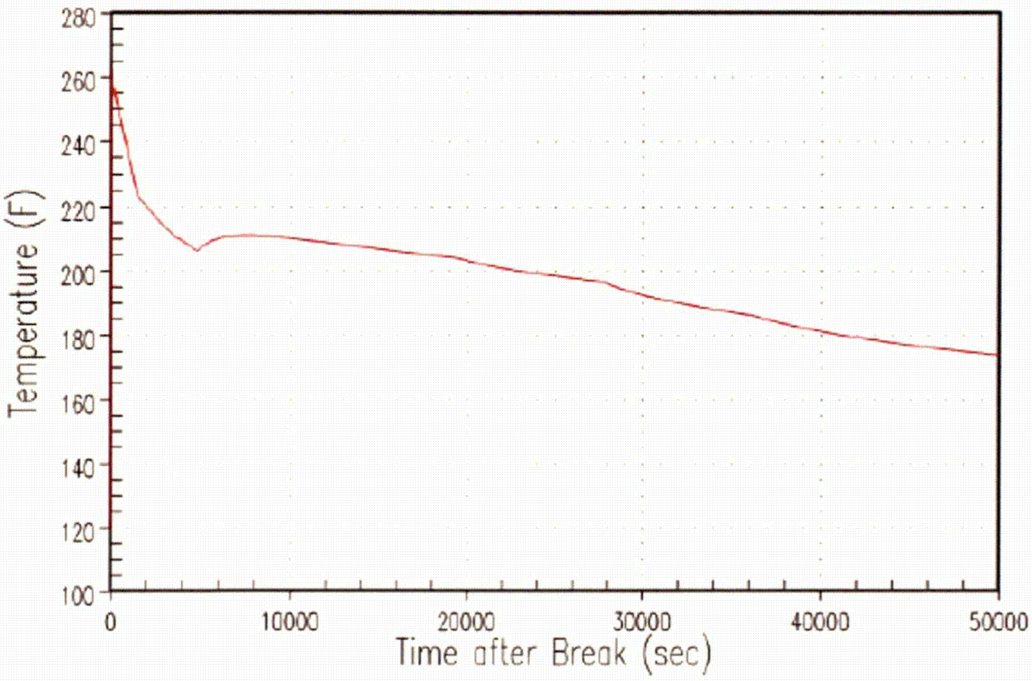

Figure 6-13 Long-term EQ Containment Temperature

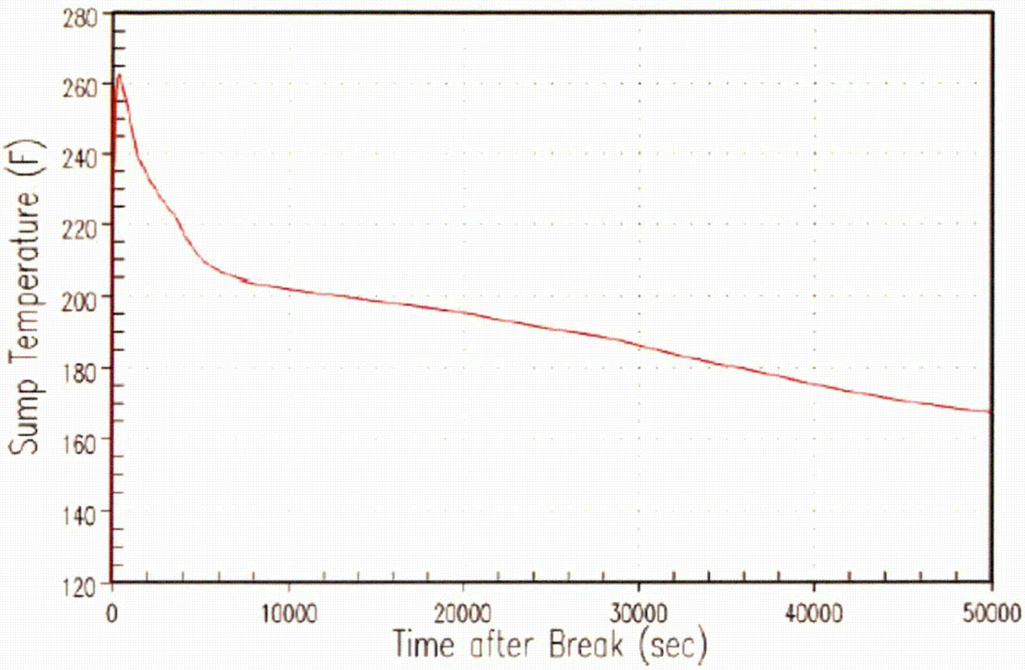

Figure 6-14 Long-term EQ Containment Sump Temperature

6.3 MINIMUM NPSHA

As described in Section 4.3, the LOCA steam release rate is minimized for the 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 6-15 and 6-16. 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 6-17 through 6-19. The containment pressure and temperature were slightly lower and the sump temperature was higher when compared with the long-term EQ sample case results.

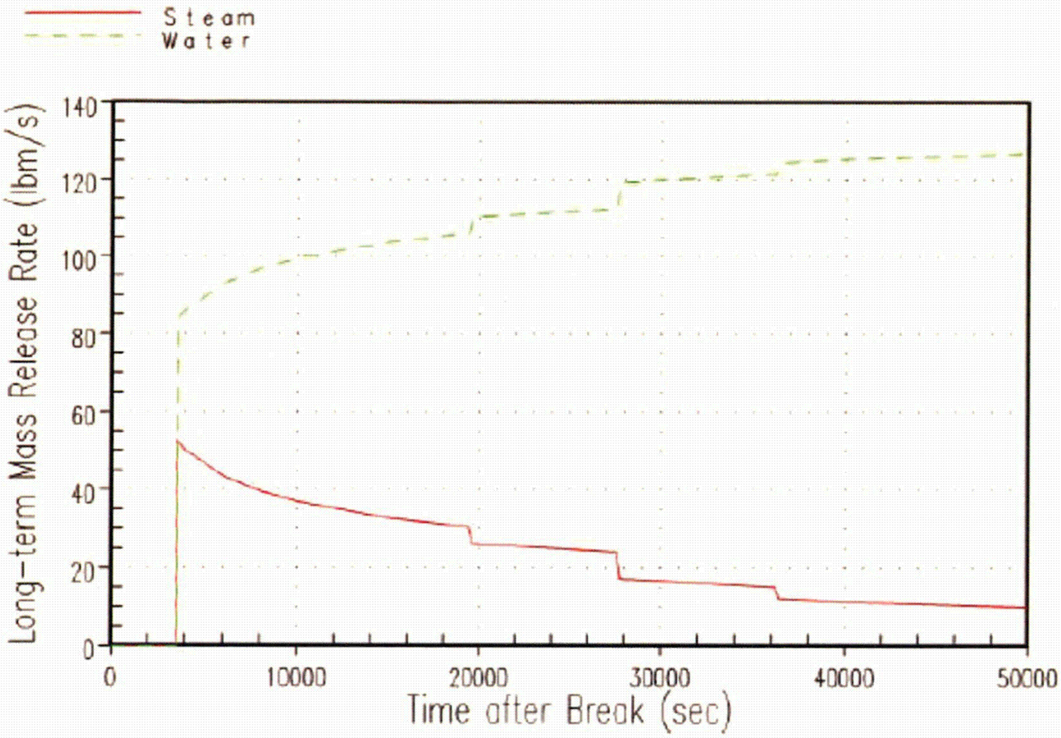

Figure 6-15 Minimum NPSHa Break Flow Rate

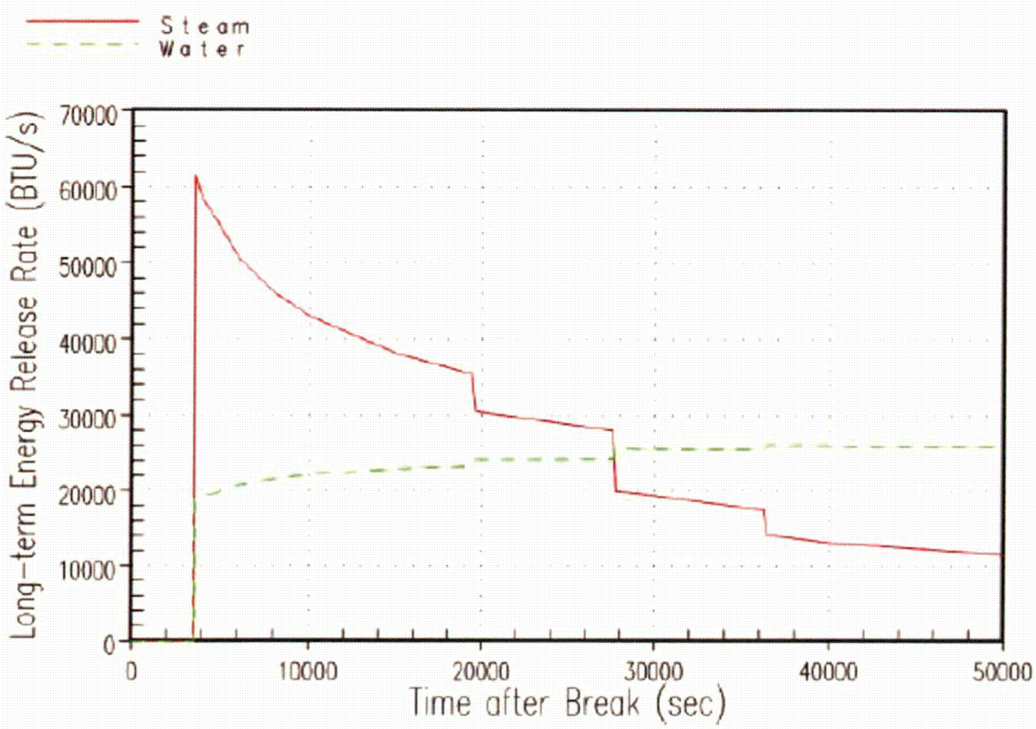

Figure 6-16 Minimum NPSHa Break Energy Flow Rate

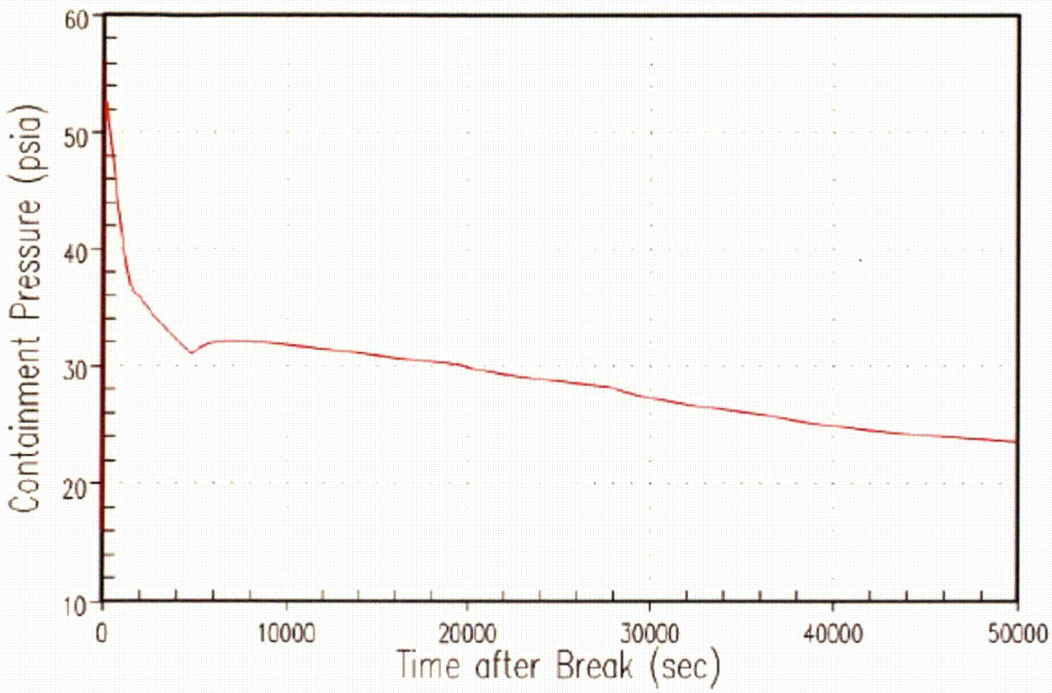

Figure 6-17 Minimum NPSHa Containment Pressure

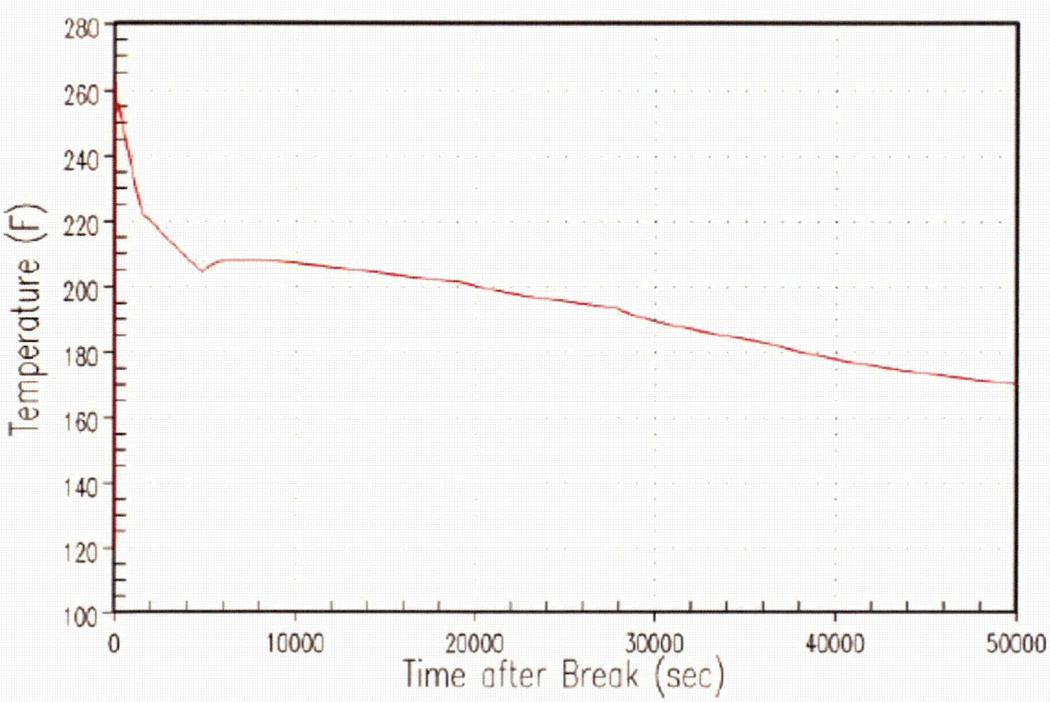

Figure 6-18 Minimum NPSHa Containment Temperature

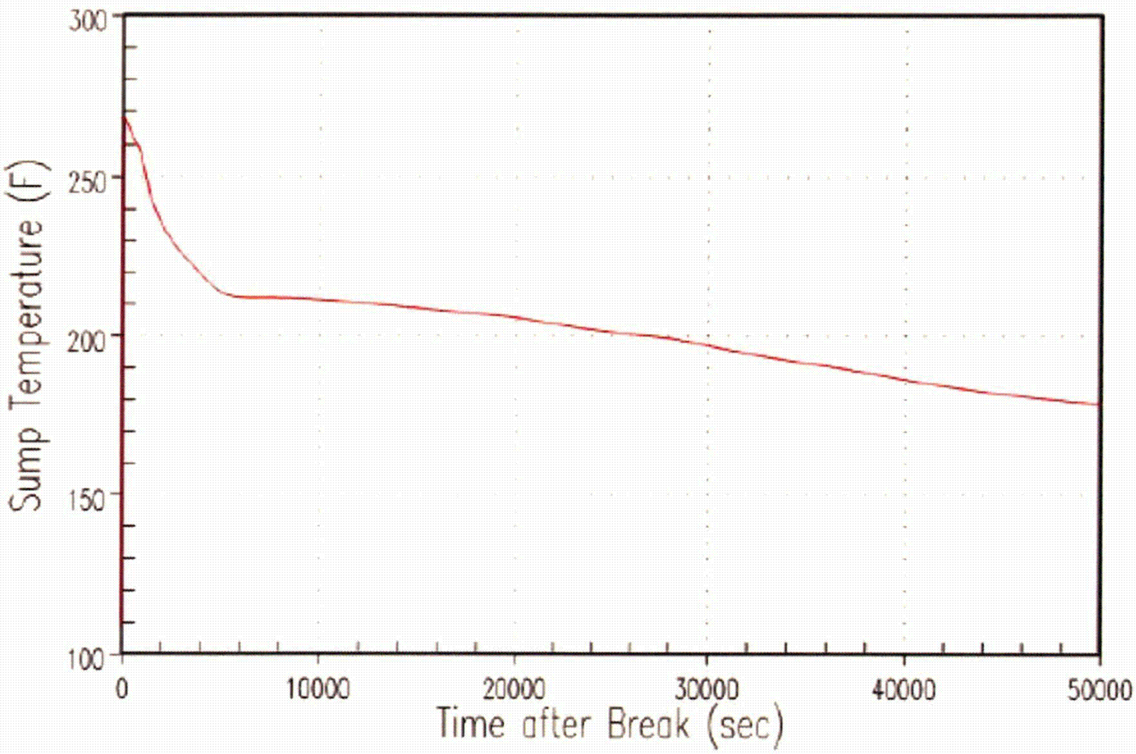

Figure 6-19 Minimum NPSHa Containment Sump Temperature

7 BENCHMARK COMPARISONS – ICE CONDENSER

This section compares the DEPS 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.

7.1 LOCA M&E MODEL DESCRIPTION

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

]^{a,c} The steady state loop noding diagram for the modified WC/T ECCS evaluation model is shown in Figure 7-1. The modified WC/T steam generator noding diagram is shown in Figure 5-2.

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 7-1. The WC/T model has a slightly lower initial RCS fluid mass and energy, but a substantially higher initial SG fluid mass and energy than SATAN-VI. 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 7-1 Initial Steady State Mass and Energy Comparison

	WC/T Model	SATAN-VI Model
RCS Fluid Mass	812848 lbm	800993 lbm
RCS Fluid Energy	353 MBtu	355 MBtu
RCS Metal Energy	148 MBtu	112 MBtu
SG Secondary Fluid Mass	690334 lbm	552900 lbm
SG Secondary Fluid Energy	389 MBtu	323 MBtu
SG Secondary Metal Energy	97 MBtu	57 MBtu

a,c

Figure 7-1 WC/T Steady State Noding Diagram (4-Loop Ice Condenser Plant)

7.2 CONTAINMENT MODEL DESCRIPTION

The LOTIC1 computer code is used to model the ice condenser containment, and LOTIC1 is discussed in detail in WCAP-8354-P-A. [

]^{a,c} Active containment cooling consists of 2 spray pumps and 2 RHR cooling loops; however, only 1 electrical train of active containment heat removal is assumed to be in operation. This leaves only 1 spray pump 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 level setpoint. The spray pump continues to draw from the RWST until the level reaches the low-low setpoint. After this, the spray pump suction is transferred from the RWST to the sump to provide recirculation spray.

The LOTIC1 model was developed following the methodology documented in WCAP-8354-P-A. Key containment model input is given in Tables 7-2 and 7-3. The containment model contains two break streams, one each for the high enthalpy and spilling sides. Steam from both break streams is directed towards the ice compartment, and liquid is spilled to the sump. Also, accumulator nitrogen release is modeled in LOTIC1.

The LOTIC1 model is run standalone. Mass and energy releases from the WC/T LOCA M&E release calculation are used as input to the LOTIC1 model. [

]^{a,c}

Table 7-2 Key LOTIC1 Model Input Values

Description	LOTIC1 Model
Containment Data	
Noding Structure	5 lumped volumes
Volume	1.27×10^6 ft^3
Initial Ice Mass	2.26×10^6 lb_m
Heat Sink Geometry – See Table 7-3	
Heat Transfer Coefficients – LOCA	Tagami
Initial Conditions	
Initial Pressure	15.0 psia
Initial Temperature	80°F (UC), 100°F (LC and DE), 15°F (IC)
Initial Humidity	10 %RH (UC, LC, DE), 100%RH (IC)
Boundary Conditions	
Break Flow Phase Separation	Steam is directed to the ice condenser, liquid to sump
Accumulator Nitrogen Release	Modeled for LOCA
Spray Flow Initiation	234 seconds
Spray Flow Rate	4000 gpm
Spray Flow Termination	Low-low RWST Level (4,575 ft^3 remaining)
LOCA Sump Recirculation Modeling	
Transfer to ECCS Recirculation	Low RWST Level (20,921 ft^3 remaining)
RHR Flow Rate	3,674 gpm injection/3160 gpm after switchover completion
RHR Heat Exchanger UA	1.666 MBTU/hr-°F
CCW Flow Rate	5,000 gpm
CCW Heat Exchanger UA	6.076 MBTU/hr-°F
Service Water Flow Rate	5200 gpm (spray)/6315 gpm (CCW)
Service Water Temperature	88°F
Deck Fans	
Number of Fans in Operation	1
Flow Rate	40,000 CFM
Initiation Time	600 seconds

Table 7-3 Heat Sink Geometry

	Area (ft^2)	Paint (in)	SS Steel (in)	CS Steel (in)	Concrete (in)	Steel and Insulation (in)	Total (in)
Operating Deck Slab 1	4880				12.8		12.8
Operating Deck Slab 2	18280	0.066			16.8		18.866
Operating Deck Slab 3	760	0.066			18.0		18.066
Operating Deck Slab 4	3840		0.25		18		18.25
Containment Shell Slab 5	56331	0.012		0.948			0.960
Operating Deck, Crane Wall, Interior Concrete Slab 6	31963				17.16		17.16
Operating Deck Slab 7	2830	0.066			13.2		13.266
Operating Deck Slab 8	760	0.066			21.0		21.066
Interior Concrete/Stainless Steel Slab 9	2270		0.25		24.0		24.25
Floor Slab 10	15921	0.066			19.2		19.266
Misc. Steel Slab 11	28500	0.012			0.79		0.802
Ice Baskets Slab 12	149600			0.080			0.080
Lattice Frame Slab 13	75865			0.260			0.260
Lower Support Structure Slab 14	28760			0.704			0.704
Ice Condenser Floor Slab 15	3336	0.066			4.0		4.066
Containment Wall Panels and Shell Slab 16	19100			0.75		12.0	12.75
Crane Wall Panels and Crane Wall Slab 17	13055				12.0	12.0	24.0

7.3 DEPS LOCA BENCHMARK CASE RESULTS COMPARISON

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

]^{a,c}

Figures 7-2 and 7-3 compare the integrated break mass and energy flow rates for WC/T and WCAP-10325-P-A. The integrated mass flow is less in the WCT run because of slight changes in SI flow rates due to pump performance assumptions (the flow change is ~10 lbm/s). The effect of this can also be seen in Figure 7-3. In the long term LOCA transient, the large majority of SI flow is spilled out of the pump side of the break, with flow to the core only making up for break flow from the steam generator side. Because of the mechanistic WC/T calculation, the overall steam flow to the lower compartment is reduced (Figure 7-4). The reduced steam flow rate causes the ice bed to melt out later for the WC/T containment response (7850 seconds vs. 3100 seconds), and the resulting peak pressure is reduced (Figure 7-5). Figures 7-6 and 7-7 indicate that the compartment temperatures are also reduced in the long term.

The sump temperature for the WC/T run (Figure 7-8) is generally higher than the WCAP-10325-P-A run. This is attributed to two factors. [

]^{a,c}

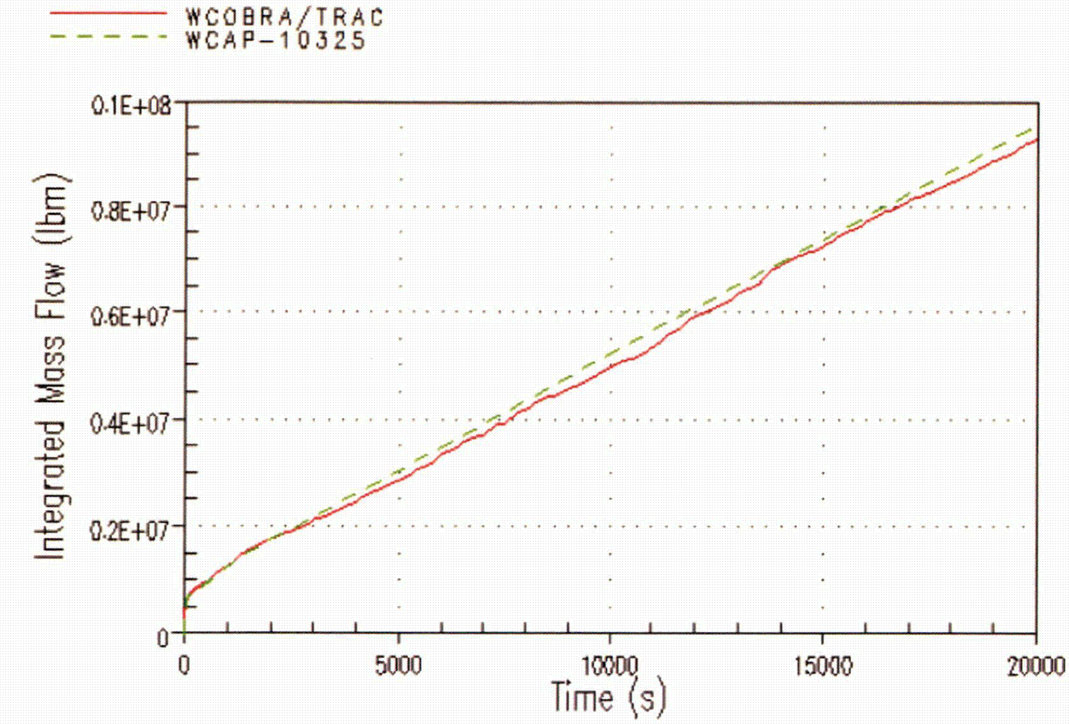

Figure 7-2 Integral Break Mass Comparison

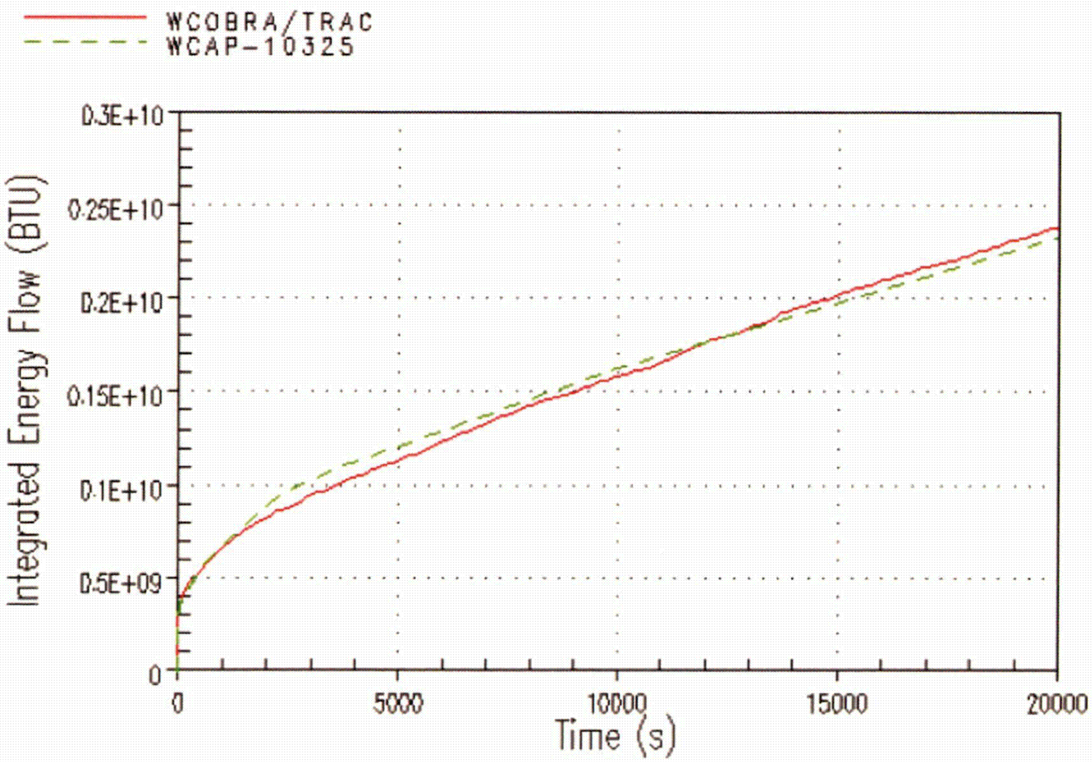

Figure 7-3 Integral Energy Comparison

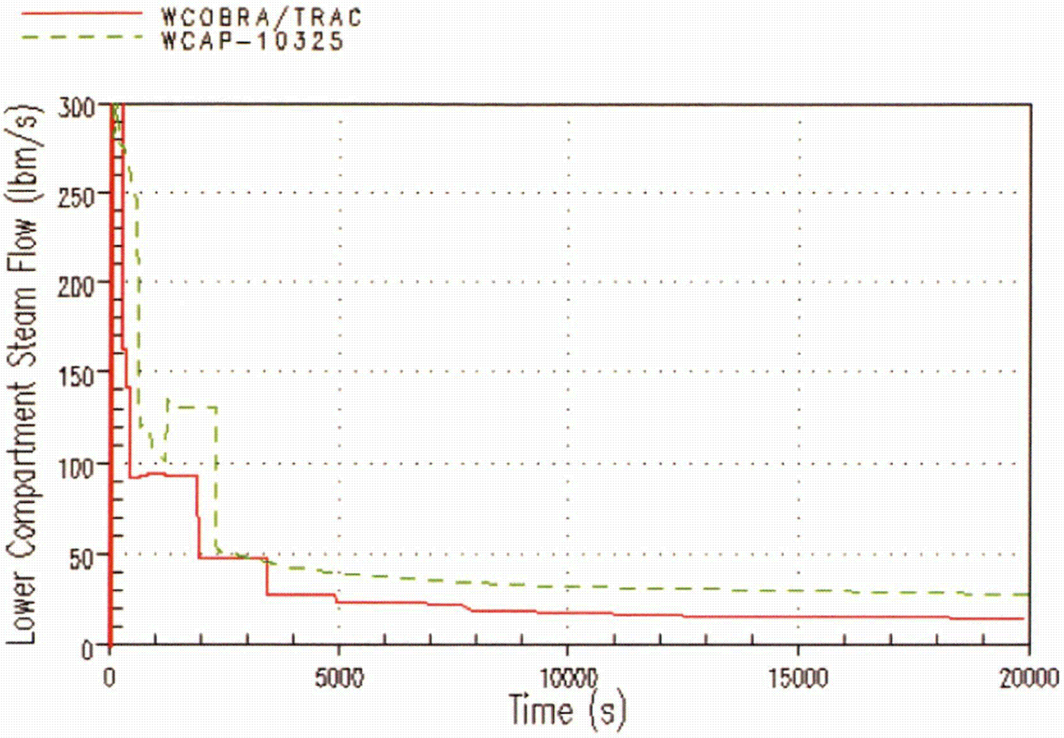

Figure 7-4 Lower Compartment Steam Flow Rate Comparison

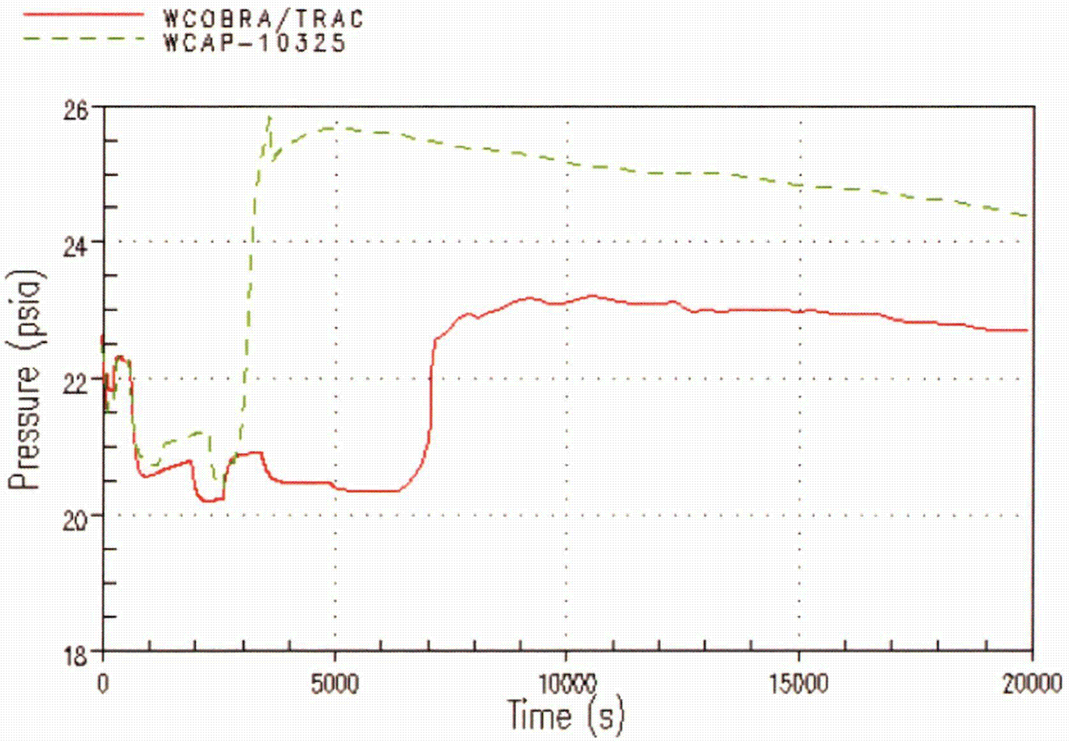

Figure 7-5 WC/T and WCAP-10325-P-A DEPS MIN Containment Pressure

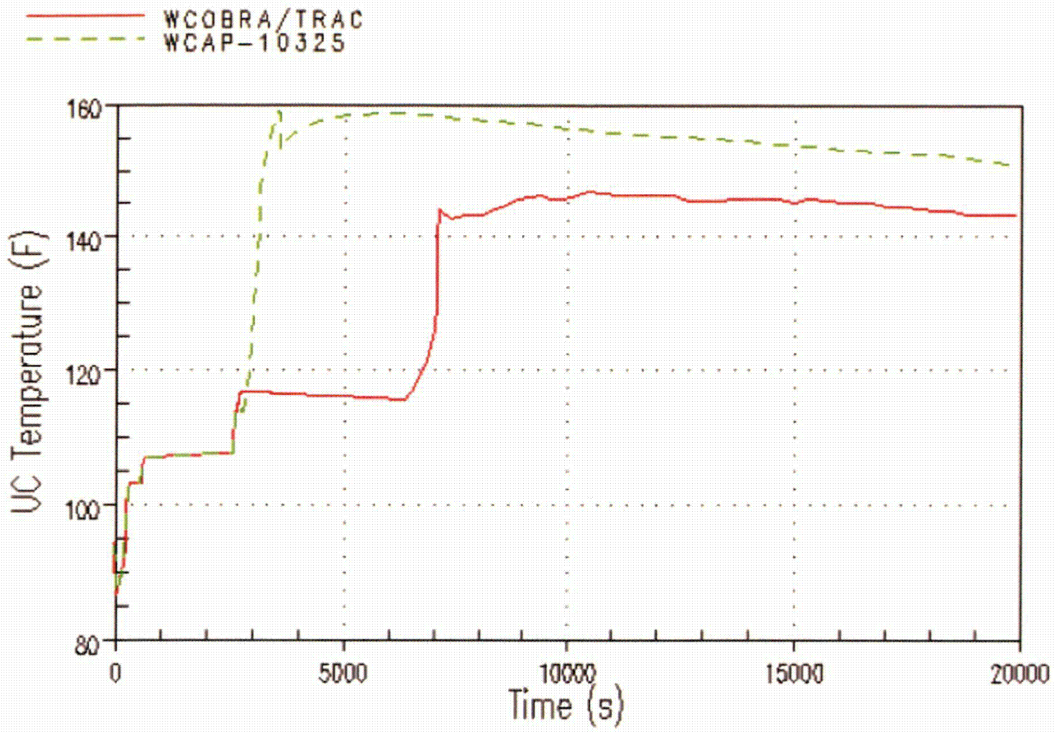

Figure 7-6 WC/T and WCAP-10325-P-A Upper Compartment Temperature

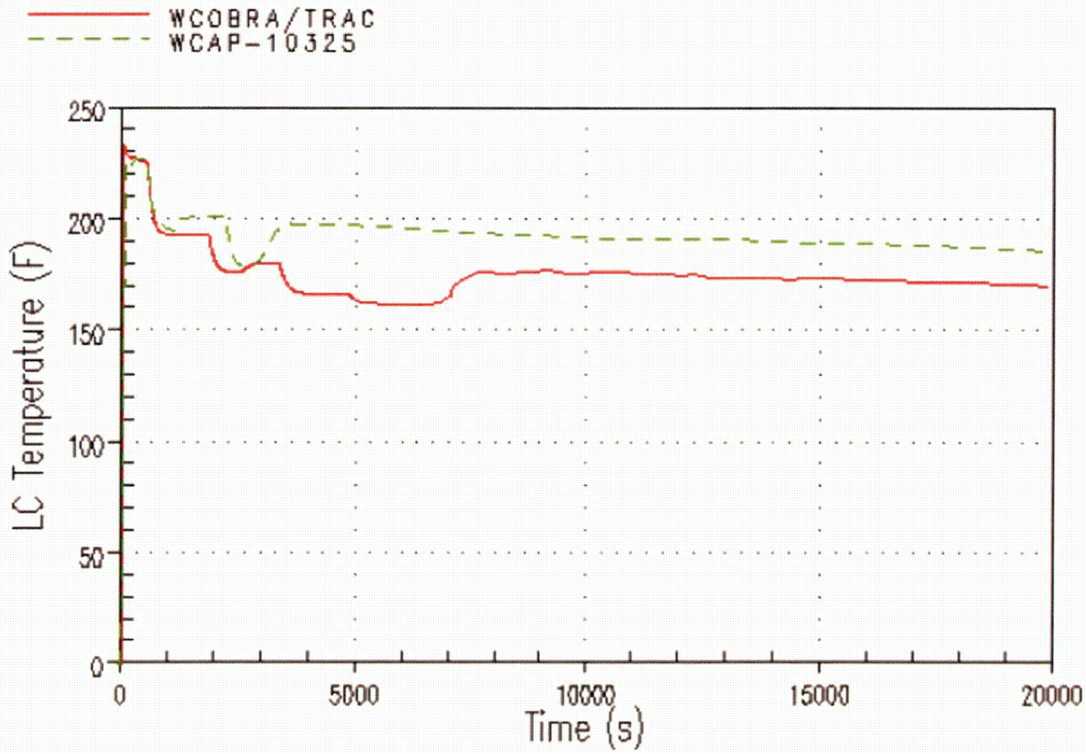

Figure 7-7 WC/T and WCAP-10325-P-A Lower Compartment Temperature

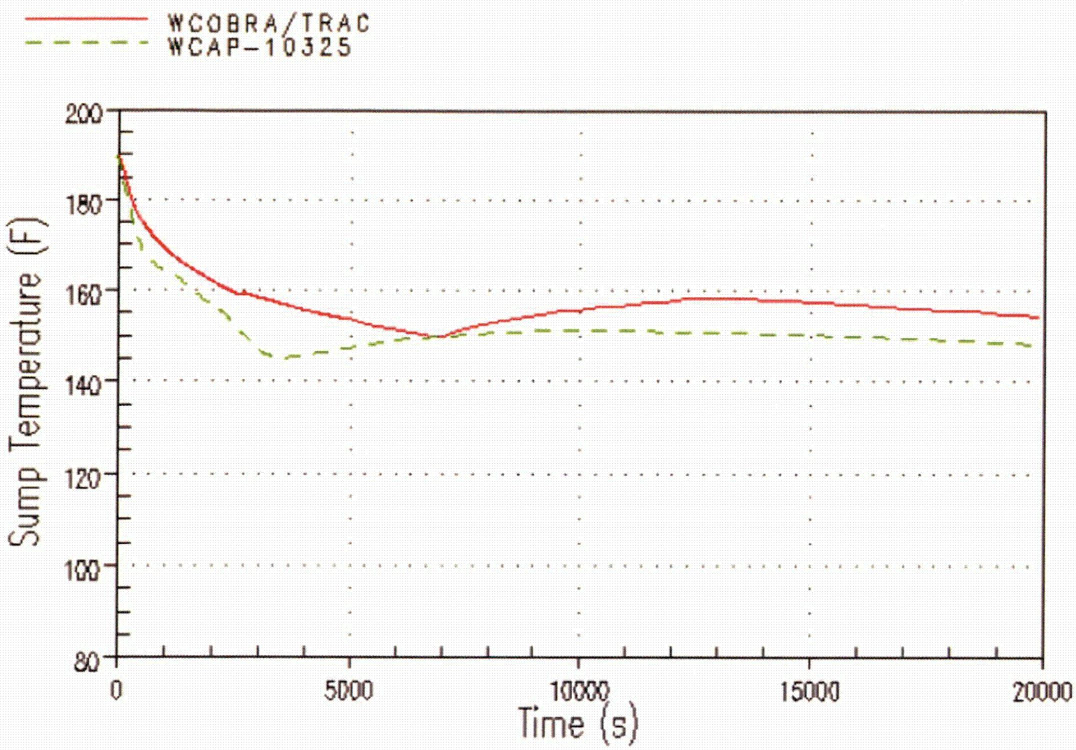

Figure 7-8 WC/T and WCAP-10325-P-A Sump Temperature

7.4 SAMPLE CASE – PEAK CONTAINMENT PRESSURE/TEMPERATURE

The WC/T LOCA M&E releases for the peak containment pressure/temperature application were generated for the DEPS and DECL LOCA events. The DEHL LOCA event was not included because it leads to little to no energy release from the steam generators, a major factor in the ice bed melt. [

]a,c

The containment pressure, upper compartment temperature, lower compartment temperature, and sump temperature are compared in Figures 7-9 through 7-12. Both the DEPS and DECL cases display elevated pressure early during the blowdown phase, but containment pressure decreases as safety systems are activated. Pressure is higher for the DECL case from approximately 500 seconds to 3200 seconds because for the DECL case the intact steam generators receive more flow from the core exit and release more secondary energy to containment. At 600 seconds, the deck fan begins to drive flow through the ice bed, thus the pressure reduction in both cases. Pressure again begins to increase slightly as the safety injection system switches to recirculation. The reduction in pressure at 2000 seconds is attributed to loop seal plugging and a change to a new portion of mass and energy release data in the LOTIC1 containment code. The increase and decrease in pressure between 2600 seconds and 2700 seconds is due to an interruption in spray flow while switchover to sump recirculation is taking place. Pressure remains low until the ice bed melts out at approximately 7800 seconds for the DEPS case and 8700 seconds for the DECL case. Because the DEPS case melts the ice bed sooner, when the decay heat rate is higher, the resulting peak

pressure is higher than the DECL case. This also leads to higher upper compartment and lower compartment temperatures.

Figures 7-13 through 7-15 show the overall mass and energy release is similar for the DECL and DEPS cases, as well as the total average break enthalpy. For the DECL case, however, the high enthalpy side break flow (flow that travels through the broken loop steam generator) encounters cold leg safety injection prior to exiting the RCS. Condensation of steam from the high enthalpy side of the DECL break leads to a higher sump temperature. Relative to peak containment temperature and pressure, the DEPS break is limiting for the ice containment design.

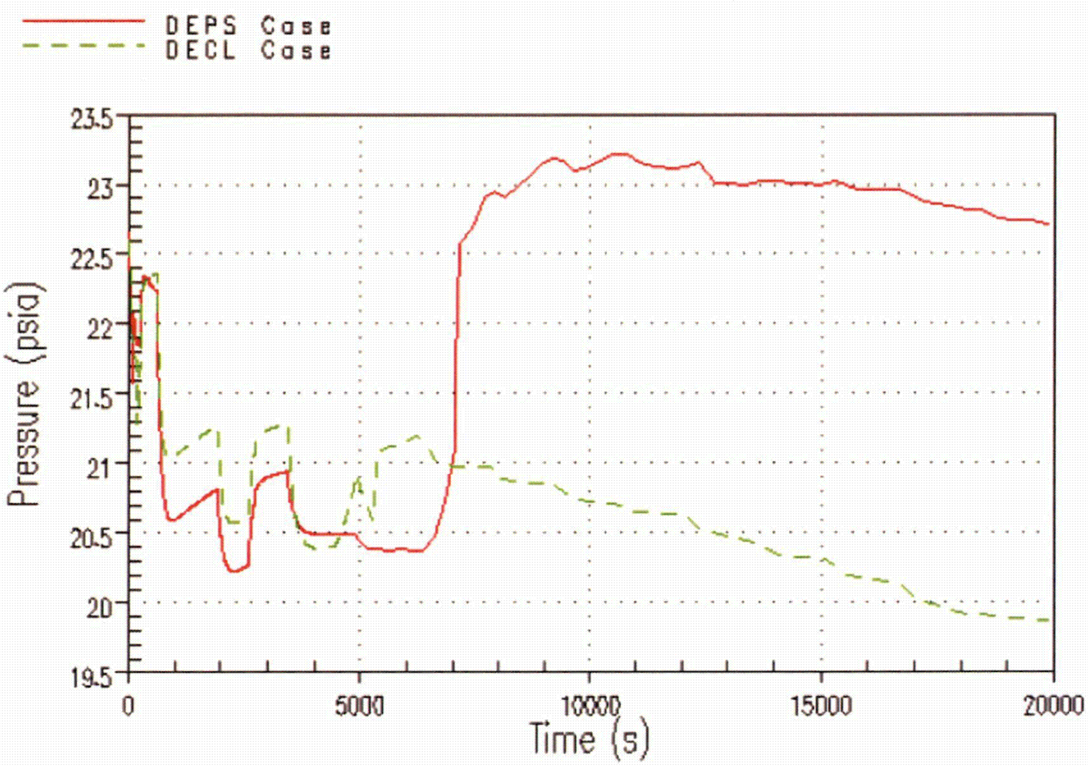

Figure 7-9 WC/T DEPS and DECL Containment Pressure

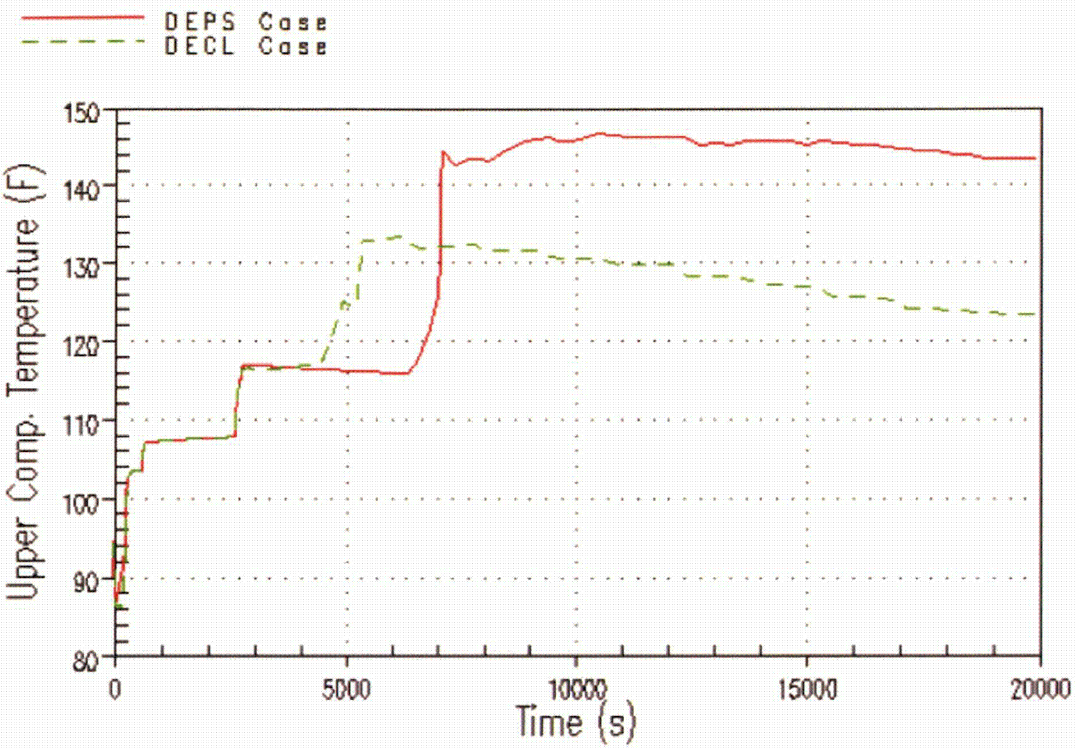

Figure 7-10 WC/T DEPS and DECL Upper Compartment Temperature

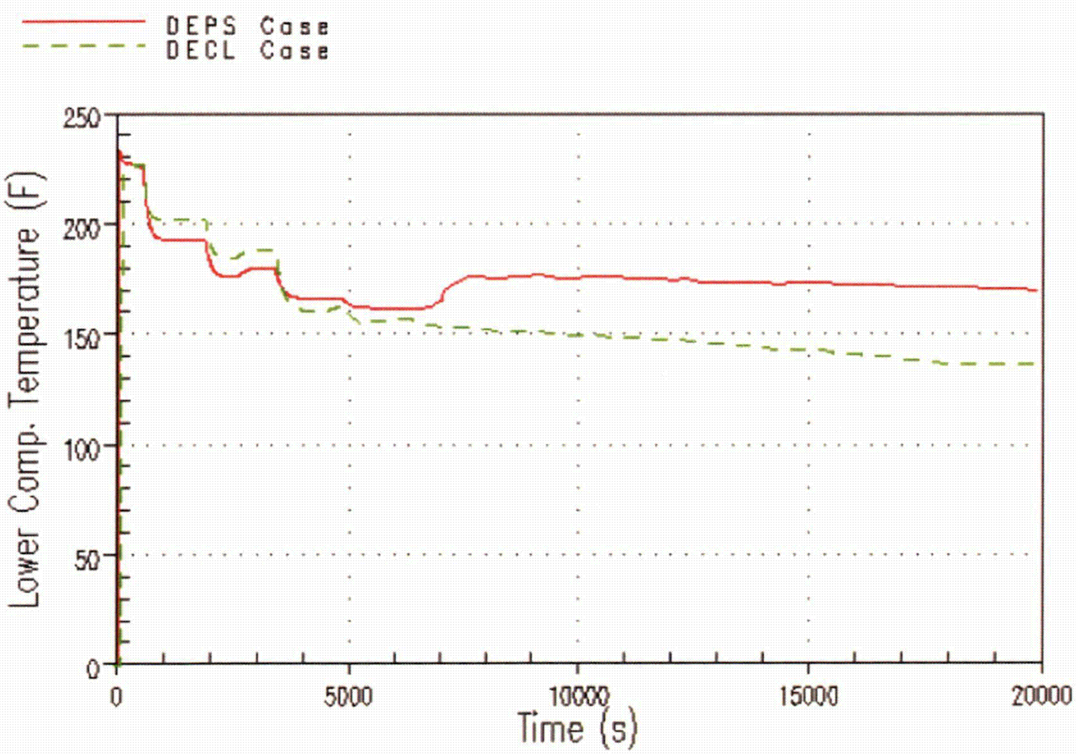

Figure 7-11 WC/T DEPS and DECL Lower Compartment Temperature

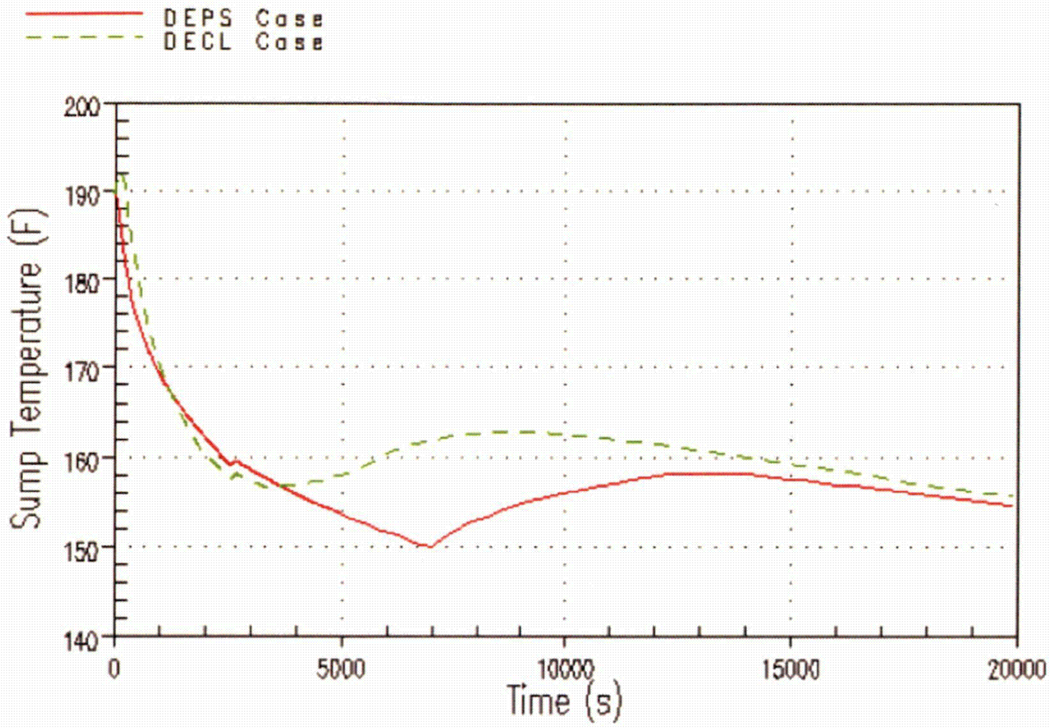

Figure 7-12 WC/T DEPS and DECL Sump Temperature

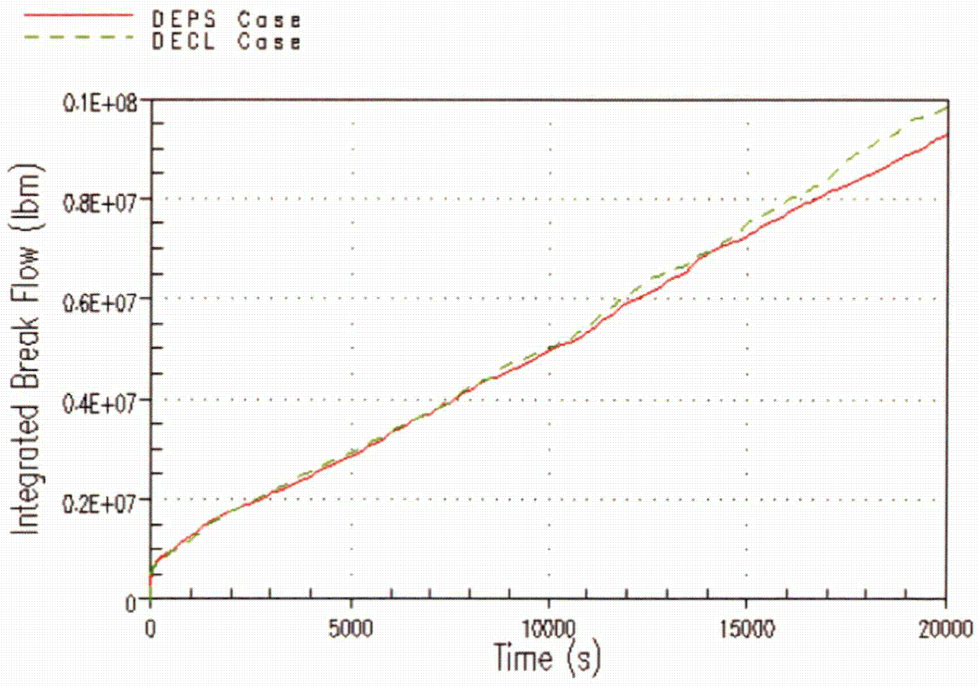

Figure 7-13 WC/T DEPS and DECL Integrated Break Flow

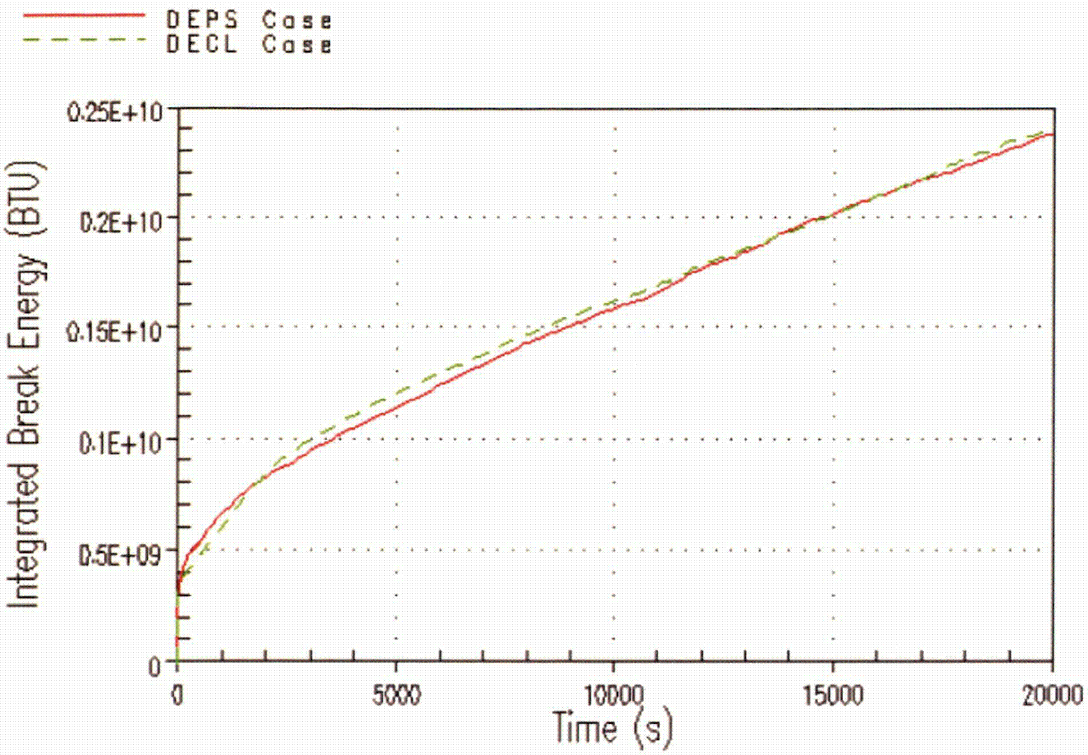

Figure 7-14 WC/T DEPS and DECL Integrated Energy Flow

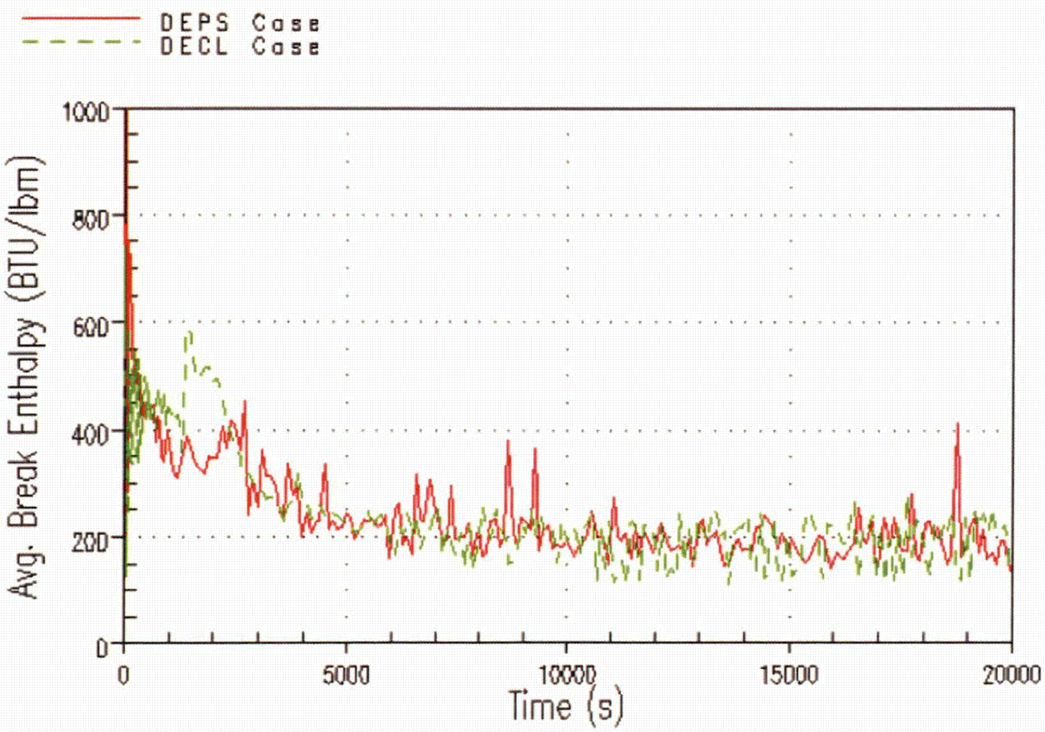

Figure 7-15 WC/T DEPS and DECL Average Break Enthalpy

8 CONCLUSIONS

The WC/T ECCS analysis model was modified, as described in this report, to allow it to produce LOCA M&E releases for the 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 release 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 can run in parallel with GOTHIC.

The WC/T ECCS evaluation model input was biased to produce conservative LOCA M&E releases in accordance with the acceptance criteria documented in NUREG-0800, Section 6.2.1.3 (Reference 27). The WC/T calculated LOCA M&E release data was compared with results from the currently approved model (Reference 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, combined with a long-term steaming release model, is capable of performing these types of calculations with analysis margins to the containment design limits.

Westinghouse intends to use this WC/T methodology to generate LOCA M&E releases for the Westinghouse NSSS Large Dry, Sub-Atmospheric, and Ice Condenser Containment Designs, and the CE/ABB NSSS Containment Designs.

9 REFERENCES

1. WCAP-8264-P-A, Rev. 1, "Westinghouse Mass and Energy Release Data for Containment Design," August 1975 (WCAP-8312-A, Rev. 2 is the non-proprietary version).

2. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design – March 1979 Version," May 1983 (WCAP-10326-A is the non-proprietary version).

3. WCAP-8302, "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," F. Bordelon, et al., June 1974 (WCAP-8306 is the non-proprietary version).

4. WCAP-8170, "Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," G. Collier, et al., June 1974 (WCAP-8171 is the non-proprietary version).

5. NUREG-0800, Section 6.2.1.1.A, "PWR Dry Containments, Including Sub-atmospheric Containments," Rev. 3, March 2007.

6A. Citation of the Reference for C-E LOCA Mass & Energy Release Methodology:

   a. CENPD-132P, Volumes 1 and 2, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

   b. CENPD-132P, Supplement 1, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

   c. CENPD-132-P, Supplement 2-P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

   d. CENPD-132, Supplement 3-P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for C-E and W Designed NSSS," June 1985.

   e. CENPD-132, Supplement 4-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," March 2001.

6B. Citation of the Reference for CEFLASH-4A Computer Code:

   a. CENPD-133P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," August 1974.

   b. CENPD-133P, Supplement 2, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," February 1975.

c. CENPD-133P, Supplement 4-P, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," April 1977.

d. CENPD-133P, Supplement 5, "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," June 1985.

7. FLOOD3 is an extension of the NRC approved FLOOD-MOD2 code referenced in the NUREG-0800, Standard Review Plan. "FLOOD-MOD2 – A Code to Determine the Core Reflood Rate for a PWR Plant with Two Core Vessel Outlet Legs and Four Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, November 1972.

8. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

9. WCAP-9724, Westinghouse Report No. 9, "PWR FLECHT SEASET Steam Generator Separate Effects Task Data Analysis and Evaluation Report," February, 1982.

10. WCAP-12945-P-A, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

11. "Modeling of the LOCA and Post-LOCA Steam Generator Heat Release with WCOBRA/TRAC," R. Macian et. Al, HTD-Vol. 251, 29th National Heat Transfer Conference, American Society of Mechanical Engineers, August 8-11, 1993.

12. "Calculation Study of Nonequilibrium Post-CHF Heat Transfer in Rod Bundle Test Using Modified RELAP5/MOD2," Y. A. Hassan, Nonequilibrium Transport Phenomena, Vol. 77, pp. 79-84, ASME HTD, New York, 1987.

13. "Heat Transfer Measurements of Evaporating Liquid Droplets," M. C. Yuen and L. W. Chen, Int. Heat Mass Transfer, Vol. 21, 1978.

14. "Dispersed Flow Heat Transfer Above a Quench Front During Reflood in a PWR after a LOCA," L. Richard, Ph.D Thesis, University of Maryland, 1982.

15. "Vapor generation Rate in Nonequilibrium Convective Film Boiling," S. W. Webb et al., Proc. 7th Intl. Heat Transfer Conf., Vol.4, pp. 437-442, Munich, 1982.

16. "Vapor Generation Rate Model for Dispersed Droplet Flow," C. Unal et al., ANS Proc. Nat. Heat Transfer Conf., pp 189-196, 1989.

17. "Axially Varying Vapor Superheats in Convective Film Boiling," D. Evans et al., ASME Journal of Heat Transfer, Vol. 107, pp 663-669, August, 1985.

18. "Forced Convective Nonequilibrium Post Critical Heat Flux Heat Transfer Experiments in a Vertical Tube," R. C. Gottula, et al., ASME-JSME Thermal Engineering Conference, Honolulu, March, 1983.

19. "Convective Film Boiling in a Rod Bundle: Axial Variation of Evaporation Ratio," C. K. Unal, Int. J. of Heat and Mass Transfer, Vol. 31, p. 2091, October, 1988.

20. "Transient Direct-Contact Condensation on Liquid Droplets," K. O. Pasamehmetoglu, Nonequilibrium Transport Phenomena, Vol. 77, PP. 47-56, ASME HTD, New York, 1987.

21. "Convective Boiling and Condensation," J. G. Collier, McGraw Hill, 1982.

22. "Advanced Programming in the UNIX Environment," W. Richard Stevens, Addison Wesley Publishing Company, Inc.

23. BE-2004 Proceedings, International Meeting on Updates in Best Estimate Methods in Nuclear Installation Safety Analysis, "PIRT for Large Break LOCA Mass and Energy Release Calculations," R. Ofstun, L. Smith, November 2004.

24. NAI 8907-02, Revision 17, "GOTHIC Containment Analysis Package User Manual, Version 7.2a," January 2006.

25. NAI 8907-06, Revision 16, "GOTHIC Containment Analysis Package Technical Manual, Version 7.2a, "January 2006.

26. NAI 8907-09, Revision 9, "GOTHIC Containment Analysis Package Qualification Report, Version 7.2a, "January 2006.

27. NUREG-0800, Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," Rev. 3, 2007.

28. WCAP-14449-P-A, Revision 1, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," October 1999.

29. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

30. Letter from E. Lenning (NRC) to J. Gresham (Westinghouse), "Request for Additional Information (RAI) Re: Westinghouse Electric Company Topical Report WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR [Pressurized Water Reactor] LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation Methodology," – Set 1 (Confirmatory Analysis) (TAC No. MF1797)," September 2014.

31. Letter from E. Lenning (NRC) to J. Gresham (Westinghouse), "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR [Pressurized Water Reactor] LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation Methodology," – Set 2 (Safety and Code Review Branch) (TAC No. MF1797)," October 2014.

32. Letter from E. Lenning (NRC) to J. Gresham (Westinghouse), "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR [Pressurized Water Reactor] LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation Methodology," – Set 3 (Containment and Ventilation Branch) (TAC No. MF1797)," October 2014.