

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

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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 framework¹ 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 model², 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of the information provided in WCAP-17721-P/NP, Revision 0, was previously submitted in WCAP-16608, Appendix C, but was withdrawn before the completion of the review.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.

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 merit³. 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.

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
<i>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.</i>
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
<i>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.</i>
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 nodding 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
<i>All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.</i>
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
<i>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.</i>
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
<p><i>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.</i></p> <p>SRP Chapter 6.2.1.3, Sub-Section III</p>

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.

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 ANS/ANS-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 increase 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, ANSI/ANS-51.1-1983, and ANSI/ANS-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 ANSIIANS-51.1-1983 and ANSIIANS-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
<p><i>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.</i></p>
<p>ANS-56.4-1983, Sub-Section 3.2.4.5.1</p>

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
<p><i>For blowdown period analysis, the primary containment backpressure is unimportant because the break flow is critical virtually throughout the blowdown period.</i></p> <p><i>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.</i></p> <p style="text-align: right;">ANS-56.4-1983, Sub-Section 3.2.4.6.1</p>

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

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 Aktieföretaget
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)	<i>Dispute</i> RAIs are asked when the staff understands the information, but believes the information is incorrect.
2	Skepticism (RAI-SK)	<i>Skeptical</i> RAIs are asked when the staff understands the information, but is unsure if the information is correct.
3	Lacking Documentation (RAI-L)	<i>Lacking Documentation</i> RAIs are asked when the staff understands the information, but considers the supporting documentation to be inadequate.
4	Clarification Necessary (RAI-C)	<i>Clarification Necessary</i> 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)	<i>Basic Information</i> RAIs are focused on obtaining some basic information not provided in the submittal.
6	Editorial Change (RAI-E)	<i>Editorial Change</i> 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:

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

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	
<i>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.</i>	
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	
<i>The initial SI (RWST) and accumulator water temperature. (2 values)</i>	
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	
<i>Specify the water volume held in the RWST at the beginning of the event. (1 value)</i>	
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	
<i>During switchover, is any period of no injection assumed? If so, how long is that period of no injection?</i>	
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	
<i>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)</i>	
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	
<i>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)</i>	
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	
<i>Provide plots of the SG pressure for the broken and intact loops. (4 plots)</i>	
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	
<i>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)</i>	
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	
<i>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.</i>	
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	
<p><i>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?</i></p> <p style="text-align: right;">2.2 RAI-3 – Break size (Reference 10)</p>	
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	
<p><i>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.</i></p> <p style="text-align: right;">2.3 RAI-3 – Refill (Reference 10)</p>	
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	
<p><i>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.</i></p> <p style="text-align: right;">2.4 RAI-3 – Refill (Reference 10)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.4.1
Comment	<p>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.</p>
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	
<p><i>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.</i></p> <p style="text-align: right;">2.5 RAI-3 – Core flooding (Reference 10)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.1
Comment	<p>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.</p>
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	
<i>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.</i>	
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	
<i>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.</i>	
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	
<p><i>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.</i></p>	
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	
<p><i>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.</i></p> <p style="text-align: right;">2.9 RAI-3 – Steam quenching (Reference 10)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.5.3 and 3.2.6.2
Comment	<p>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.</p>
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

EQ and NPSHa	
<p><i>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.</i></p>	
2.10 RAI-3 – EQ and NPSHa (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.2.6.2
Comment	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.
Reference of Response	1. 2.10 RAI-3 – Equipment Qualification (EQ) and Net Positive Suction Head Analysis (Reference 14)

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	
<p><i>Provide a table which contains the following:</i></p> <ol style="list-style-type: none"> 1. <i>The phase of the event (e.g., Blowdown, Refill, Reflood)</i> 2. <i>The conditions which define the beginning of that phase.</i> 3. <i>The conditions which define the end of that phase</i> 4. <i>An approximate duration of that phase (in seconds)</i> 5. <i>An approximate starting time of that phase (in seconds – with 0 being the event initiation)</i> 6. <i>A description of how the phase is simulated (e.g., mechanistically in WC/T, conservatively using certain approximations)</i> <p><i>Additionally, provide a second table which contains a description of the energy sources which impact each of the phases listed in the above table:</i></p> <ol style="list-style-type: none"> 1. <i>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.</i> 2. <i>The approximate initial energy of that energy source at the beginning of the event (in kW).</i> 3. <i>The approximate amount of energy which is released during phase 1 (include both kW and %)</i> 4. <i>The approximate amount of energy which is released during phase 2 (include both kW and %)</i> <p><i>The approximate amount of energy which is released during every other phase of the event (include both kW and %)</i></p> <p style="text-align: right;">2.12 RAI-3 – Event definitions (Reference 10)</p>	
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	
<i>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.</i>	
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	
<i>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.</i>	
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	
<i>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.</i>	
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	
<i>Are measurement uncertainties considered for the values of the initial safety injection tank water volume and water temperature?</i>	
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

Nodalization	
<p><i>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.</i></p>	
2.17 RAI-3 – Nodalization (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.1
Comment	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.
Reference of Response	1. 2.17 RAI-3 – Nodalization (Reference 14)

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	
<i>Which steam tables are used in the M&E evaluation model? Are those steam tables consistent with the 1967 ASME Steam Tables?</i>	
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

Flow modeling	
<p><i>Confirm that the following effects have been taken into account in the flow modeling used in the M&E evaluation model:</i></p> <ul style="list-style-type: none"> <i>(1) temporal change of momentum,</i> <i>(2) momentum convection,</i> <i>(3) forces due to wall friction,</i> <i>(4) forces due to fluid pressure,</i> <i>(5) forces due to gravity, and</i> <i>(6) forces due to geometric head loss effects (for example, contractions, expansions, bends, and pump losses).</i> <p><i>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.</i></p> <p><i>Additionally confirm that if an uncertainty in a pressure loss exists, the pressure loss shall be conservatively minimized.</i></p>	
2.19 RAI-3 – Flow Modeling (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.3
Comment	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.
Reference of Response	1. 2.19 RAI-3 – Flow Modeling (Reference 15)

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	
<i>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.</i>	
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

Downcomer condensation	
<p><i>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.</i></p> <p style="text-align: right;">2.21 RAI-3 – Downcomer condensation (Reference 10)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.3
Comment	<p>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.</p>
Reference of Response	1. SCVB-RAI-11, Downcomer Condensation (Reference 13)

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	
<i>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.</i>	
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	
<i>Demonstrate that the assumption to ignore any hot leg condensation is also appropriate for NPSHa and EQ analysis.</i>	
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

Dynamic pump model	
<i>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.</i>	
2.24 RAI-3 – Dynamic pump model (Reference 10)	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.4.4
Comment	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.
Reference of Response	1. 2.24 RAI-3 – Dynamic pump model (Reference 15)

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	
<p><i>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.</i></p>	
<p>2.25 RAI-3 – GOTHIC time step sensitivity (Reference 10)</p>	
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	<ol style="list-style-type: none"> 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	
<i>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.</i>	
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

Heat transfer correlations	
<p><i>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.</i></p> <p style="text-align: right;">2.27 RAI-3 – Heat transfer correlations (Reference 10)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	3.3.7.1, 3.3.7.2, and 3.3.7.6
Comment	<p>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.</p>
Reference of Response	1. 2.27 RAI-3 – Heat transfer correlations (Reference 16)

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	
<i>Is heat transfer from the primary and secondary metal to containment directly calculated and if not why is this appropriate?</i>	
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	
<i>Define inactive metal and discuss how it is treated.</i>	
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	
<p><i>Provide validation for Unal’s correlation over its application domain as used in the M&E evaluation model.</i></p> <p style="text-align: right;">2.30 Unal’s correlation (Reference 10)</p>	
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	<ol style="list-style-type: none"> 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	
<i>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.</i>	
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

FLECHT heat release rate	
<p><i>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.</i></p>	
2.32 FLECHT heat release rate (Reference 10)	
Disagreement Level	Level 2 – Skepticism
SE Section	3.3.7.3
Comment	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.
Reference of Response	<ol style="list-style-type: none"> 1. 2.32 RAI-2 – FLECHT (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)

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	
<i>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.</i>	
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	
<i>Provide the definition for the following acronyms: PCWG, DEPSG, EQ, NPSHa, DEHLG, GENF</i>	
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	
<i>The first full paragraph on page 3-5 does not make sense. Revise this paragraph and re-submit it.</i>	
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	
<p><i>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.</i></p> <p style="text-align: right;">(Reference 20)</p>	
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	
<p><i>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.</i></p>	
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	
<i>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.</i>	
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	
<i>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.</i>	
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	
<p><i>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.</i></p> <p style="text-align: right;">SCVB-RAI-4, Break Spectrum (Reference 11)</p>	
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	
<p><i>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.</i></p> <p style="text-align: right;">SCVB-RAI-5, GOTHIC Topical Report (Reference 11)</p>	
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	
<i>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.</i>	
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	
<p>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.").</p>	
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	
<p><i>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).</i></p>	
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	
<i>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.</i>	
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	
<i>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?</i>	
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”

Conformance of Regulatory Guide (RG) 1.203, “Transient and accident analysis methods”	
<p><i>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.</i></p>	
<p>Conformance of Regulatory Guide (RG) 1.203, “Transient and Accident Analysis Methods” (Reference 11)</p>	
Disagreement Level	Level 3 - Lacking Documentation
SE Section	Various
Comment	None
Reference of Response	<ol style="list-style-type: none"> 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)

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

Audit	
<p>Provide the following calculations for audit via electronic reading room:</p> <ol style="list-style-type: none"> 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. <p style="text-align: right;">SCVB-RAI-12, Audit (Reference 11)</p>	
Disagreement Level	Level 5 - Basic Information
SE Section	Various
Comment	None
Reference of Response	1. Audit performed - see Audit Summary (Reference 12)

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.