U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

FINAL SAFETY EVALUATION

FOR TOPICAL REPORT WCAP-16747-P/NP, APPENDICES C AND D, REVISION 2,

"POLCA-T: SYSTEM ANALYSIS CODE WITH THREE-DIMENSIONAL CORE MODEL,

APPENDICES C AND D"

WESTINGHOUSE ELECTRIC COMPANY

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Enclosure 1

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1. INTRODUCTION

Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-16747-P/NP, Revision 2, "POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D" (Ref. 70)[1], for U.S. Nuclear Regulatory Commission (NRC) staff review and approval, that was accepted for the NRC review via letter dated September 25, 2024 (Ref. 73). This TR presents the qualification of the application of the POLCA-T methodology for Anticipated Operational Occurrence (AOO) Transient Analysis and Anticipated Transients Without Scram (ATWS) Analysis in boiling water reactors (BWR/2-6). This TR is not applicable to Advanced Boiling Water Reactor (ABWR) plant designs.

The POLCA-T methodology described in Westinghouse's TR WCAP-16747-P-A, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model," was reviewed and approved by the NRC staff's safety evaluation (SE) in 2010 (Ref. 68). This included the general POLCA-T computer code as well as the Appendix A for Control Rod Drop Accident Analysis and Appendix B for Stability Analysis. TR WCAP-16747-P/NP, Revision 2, "POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D," is a continuation of that previous work, with WCAP-16747-P/NP, Revision 2, Appendix C, containing Transient Analysis and Appendix D containing ATWS Analysis. Because they are all part of a single method, this SE should be considered an extension of the original SE given the initial POLCA-T methodology, and therefore this SE will add to the NRC's original SE by adding Sections 3.4 and 3.5.

The list of key correspondence between the NRC and Westinghouse is provided in Table 1 below.

Table 1: List of Key Correspondence

Sender	Document	Document Date	Reference
Westinghouse	TR Revision 2 Submittal Letter	September 23, 2024	70
NRC	Acceptance Letter	September 25, 2024	73
NRC	Request for Additional Information (RAI)	March 24, 2025	74
Westinghouse	RAI Response – Set 1	June 9, 2025	75
Westinghouse	RAI Response – Set 2	July 3, 2025	76
Westinghouse	RAI Response - Updated	June 2026	88

General information for each RAI including its number, its topic, its associated SE section, and the reference(s) of its response are given in Table 2 below.

[1] Reference 70 supersedes in its entirety Reference 69, as documented in the NRC acceptance letter dated September 25, 2024 (Ref. 73).

Table 2: Listing of RAIs

RAI	Subject	Main Associated SE Section	Reference of Response
12-1	Dryout Correlation	3.4.3.1.1	88
12-2	Counter current flow limitation revision	3.4.3.1.3	75
12-3	Steam separator model performance	3.4.3.1.4	88
12-4	[ ]	3.4.3.1.5	76
12-5	Reactor Internal Pump Model	3.4.3.1.6	75
12-6	Dynamic internal bypass channel void formation	3.4.3.1.7	76
12-7	Void reactivity coefficient biases and uncertainties	3.4.3.1.9	88
12-8	Lower Plenum Nodalization	3.4.3.1.10	88
12-9	[ ]	3.5.3.1.1	88
12-10	[ ]	3.5.3.1.2	75
12-11	Review of POLCA-T application to post-dryout conditions	3.5.3.1.3	88
12-12	Review of POLCA-T application to Delta over Initial CPR [critical power ratio] Versus Oscillation Magnitude (DIVOM) analysis	3.4.3.1.5	75
12-13	Review of POLCA-T application for exothermic metal-water reaction	3.5.3.1.7	88
12-14	Cladding rupture mode	3.4.3.2.6	75
12-15	[ ]	3.4.3.2.5	88
12-16	Interface to plant control system simulation tool SAFIR	3.4.3.2.2	75
12-17	Level measurement system model	3.4.3.2.3	88
12-18	Advanced control rod insertion mode	3.4.3.2.4	75
12-19	Model Uncertainties for the AOO transient Evaluation Model/Model Uncertainties for the ATWS Evaluation Model	3.4.3.3 3.5.3.3	75
12-20	POLCA-T for Anticipated Transient Without a Scram Assessment and Acceptance Criteria	3.5	76
12-21	Calculation of the decay ratio and oscillation frequency	3.5.3.2.5	75

2. REGULATORY EVALUATION

Direct quotations from the NRC staff final SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68), are listed below in *italics*:

> *Section 50.34 of Title 10 of the Code of Federal Regulations (10 CFR)* (Ref. 72) *"Contents of construction permit and operating license application; technical information," requires that the licensee (or vendors) provide safety analysis reports to the NRC detailing the performance of systems, structures, and components provided for the prevention or mitigation of potential accidents.*
>
> *Regulation 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 28, "Reactivity Limits," requires that reactivity control systems shall be designed with appropriate limits on the rate of reactivity insertion considering a potential rod dropout event.*
>
> *Regulation 10 CFR 50 Appendix A GDC 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.*
>
> *Regulation 10 CFR 50 Appendix A GDC 12, "Suppression of reactor power oscillations," requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.*
>
> *The intent of the current application is to review the POLCA-T methodology in terms of its generic capabilities and to specifically review the methodology to analyze control rod drop accidents to demonstrate compliance with GDC 28 and to evaluate BWR* [boiling water reactors] *stability to demonstrate compliance with GDC 10 and 12.*

In addition to the criteria identified above, the NRC staff considered other applicable regulatory requirements relevant to AOOs and ATWS. The regulatory basis associated with those topics is addressed in their respective sections of this SE, consistent with the structure of the review and the specific aspects of the POLCA-T methodology evaluated therein.

3. TECHNICAL EVALUATION

To maintain continuity with the NRC staff SE documented in the NRC's original review of WCAP-16747-P, Revision 0 (Ref. 68), this SE does not replace or recreate that earlier evaluation. Instead, it builds directly upon it so that the two SEs can be read together as one continuous, integrated document. For consistency, the numbering structure used in the original SE has been preserved, including continuation of the RAI sequence (with this SE beginning at RAI 12-X). The reference list from the earlier review has been carried forward and expanded only as needed, without renumbering or reorganizing citations. Direct quotations from the NRC staff's final SE for WCAP-16747-P, Revision 0 (Ref. 68), are presented in italics for clarity.

In keeping with the continuity of the original SE, the organization of the NRC staff's technical review, as presented in the initial evaluation, is reproduced below:

> *The NRC staff reviewed WCAP-16747-P in accordance with SRP* [Standard Review Plan*] 15.0.2 (Ref. 1). The technical review is documented in three sections. Section 3.1 documents the NRC staff review of the generic models and solution techniques, including the field equations, closure relationships, and physical models. Section 3.2 documents the NRC staff review of the specific application of POLCA-T to the analysis of control rod drop events, and Section 3.3 documents the NRC staff review of the application to BWR stability evaluation.*

To this initial SE, the NRC staff added Section 3.4 which documents the NRC staff review of the specific application of POLCA-T to the analysis of AOOs Transient Analysis, and Section 3.5 documents the NRC staff review of the application ATWS.

Additionally, the NRC staff has previously reviewed and approved a significant portion of the basis for the POLCA-T methodology in WCAP-17203-P-A, Revision 0-2, "Fast Transient and ATWS Methodology" (Ref. 73). This document augmented the existing approved methodology described in CENPDA-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (Ref. 1).

3.1. POLCA-T as a General Purpose Method

The NRC staff's review of this section is contained in Section 3.1 of the NRC staff's SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68).

3.2. POLCA-T for Control Rod Drop Accident Assessment and Acceptance Criteria

The NRC staff's review of this section is contained in Section 3.2 of the NRC staff's SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68).

3.3. POLCA-T for Stability Assessment and Acceptance Criteria

The NRC staff's review of this section is contained in Section 3.3 of the NRC staff's SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68).

3.4. POLCA-T for Anticipated Operational Occurrence Transient Analysis Assessment and Acceptance Criteria

3.4.1. Introduction

In 1996, Westinghouse obtained approval for performing safety analysis for BWRs in CENPD-300-P-A (Ref. 1). In 2010, Westinghouse obtained approval for inclusion of the POLCA-T code to be used with that methodology (Ref. 68). That approval was limited to using the POLCA-T code for control rod drop assessment and stability assessment. However, part of this review was the approval of POLCA-T as a general-purpose code with the NRC staff being aware that Westinghouse intended to submit additional appendices for POLCA-T's use in AOO Transient Analysis and for ATWS.

Since that review, Westinghouse has updated its Fast Transient and ATWS Methodology with the NRC review and approval of WCAP-17203-P-A, Revision 0-2 (Ref. 72). As stated in the accompanying NRC staff's SE:

> WCAP-17203-P/WCAP-17203-NP, Revision 0-2, provides a generic, code-independent methodology for evaluating only fast transients including both potentially limiting as well as nonlimiting events.

In summary, WCAP-17203-P-A is Westinghouse's currently approved evaluation model (EM) for fast transients. As many of the limiting AOOs are fast transients, WCAP-17203-P-A, Revision 0-2, contains portions of the background information justifying POLCA-T for AOOs including a description of the transient scenarios, the separation of the AOOs into four main categories, the specification of the acceptance criteria for each AOO, the specification of the figures-of-merit (FoMs) for each AOO, and a Phenomena Identification and Ranking Table (PIRT) for each AOO.

The EM described in WCAP-17203-P-A, Revision 0-2, was limited to fast transients. All transient events can be grouped into fast and slow transients based on the dynamic characteristics of the transient. Fast transients are those events of relatively short duration such that the impact of the spatial and temporal dynamics on the system nuclear and thermal hydraulics is important to the overall plant response. Slow transients are defined as those transients where the dynamic changes during the transient are sufficiently slow so that the assumption that steady state conditions are achieved at each time step is either realistic or conservative. WCAP-17203-P-A did not impact the slow transient evaluation methodology which is provided in CENPD-300-P-A (Ref. 1).

Following the distinction between fast and slow transients, **[**

].

The review of POLCA-T for AOOs was focused on the following:

1) Ensuring that the concerns identified by the NRC staff with respect to use of POLCA-T for AOO transient analysis have been satisfied.

2) Ensuring that POLCA-T is capable of modeling each important phenomenon identified in the approved EM (WCAP-17203-P-A) for fast transients.

3) Ensuring that POLCA-T is capable of modeling the combination of phenomena important to AOO transient analysis.

3.4.2. Applicable Regulatory Basis

GDC 10 requires that the reactor protection system must be capable of terminating any anticipated transients, including unstable power oscillations, prior to exceeding fuel design limits. GDC 10 states:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

WCAP-16747, Revision 2, will be applied to predict the core behavior following an AOO. The methodology uses these predictions to ensure that the minimum critical power ratio (MCPR), linear heat generation rate (LHGR), and peak reactor coolant pressure boundary (RCPB) pressure remain within limits and thus meet the requirements of GDC 10 under AOO conditions.

3.4.3. Technical Evaluation Regarding Anticipated Operational Occurrence Transient Analysis

3.4.3.1. NRC staff-identified deferred items with respect to POLCA-T's use as a general code for Anticipated Operational Occurrence Transient Analysis

The original submittal (Ref. 1) contained the specific application of POLCA-T to control rod drop accident analyses and stability evaluation. However, in the SE for WCAP-16747-P/NP-A, Rev. 0, the NRC staff performed a separate generic review of the POLCA-T methodology per Section III.5 of SRP 15.0.2 to minimize the potential for repeated review effort for multiple applications of POLCA-T, including the NRC staff's anticipated review of POLCA-T for AOO and ATWS analyses. In that original review, the NRC staff identified particular items needed before the acceptance of POLCA-T for AOO and ATWS analyses in Appendix D of the NRC staff's original SE (Ref. 68). Those deferred items are called review items. The review items from Appendix D that are associated with AOOs are addressed below. Some AOO review items involve plant response characteristics that are also relevant to ATWS behavior. Where this overlap exists, the discussions below note the applicability of the AOO review item to ATWS analyses so that the shared technical considerations are addressed once in a consolidated manner.

3.4.3.1.1. The application of dryout correlations in Anticipated Operational Occurrence Transient Analysis to predict Minimum Critical Power Ratio

The NRC staff's review of the dryout (i.e., critical power) correlations was discussed in Section 3.1.2.2.3 of the NRC staff's initial SE (Ref. 68), as well as RAIs 2-3, 5-4, 5-5, and 8-1. Dryout correlations are used to predict the MCPR during AOO Transient Analysis. Such correlations are reviewed and approved by the NRC for each fuel type, and for more details about such reviews, see NUREG/KM-0013 (Ref. 77). Westinghouse has obtained approval for many of these correlations for various fuel types, including SVEA-96 Optima2 (Refs. 78-80),

SVEA-96 Optima3 (Ref. 81), and TRITON11® (Ref. 82). These correlations are trained and validated against steady state data and then applied to unsteady transient analysis. While it has been commonly assumed that such dryout correlations developed for steady state conditions would conservatively predict transient behavior, transient validation is used to confirm this assumption.

In the review of each of the correlations, Westinghouse has provided validation demonstrating that the correlation accurately or conservatively predicts the CPR during a transient. For a CPR prediction to be accurate, the predicted CPR should be ≤1 when critical power has been reached (i.e., dryout occurs) in the test bundle and >1 when it has not. For a CPR prediction to be conservative, the predicted CPR should be ≤1 when critical power has been reached in the test bundle, but it can also be ≤1 if critical power has not been reached, as this would result in assuming the transient were more limiting than actual.

During the review of each CPR correlation, the NRC staff evaluated the correlation's ability to accurately predict CPR during transient conditions, based on the validation data submitted. The NRC previously reviewed and approved the critical power correlations used in the POLCA-T code for predicting the performance of SVEA Optima2, SVEA Optima3, and TRITON11 fuel designs. Unlike critical heat flux correlations - which typically require re-validation when implemented in a new computer code due to the presence of numerous closure models - critical power correlations involve fewer closure models. This characteristic generally facilitates its transferability between computational codes.

To demonstrate the performance of the critical power correlations in POLCA-T, Westinghouse provided plots comparing measured and predicted transient CPR values. In response to RAI-12-1 (Ref. 76), Westinghouse also supplied a crosswalk between the plots included in the WCAP-16747-P/NP, Revision 2 - generated with POLCA-T - and the original plots from the approval of each correlation in BISON. The NRC reviewed these comparisons and determined that POLCA-T and BISON exhibit very similar predictive behavior. Therefore, the NRC concludes that the CPR correlations originally validated in BISON, along with their associated uncertainty limits, are applicable when implemented in POLCA-T. Because Westinghouse has demonstrated this equivalency, the NRC staff considers this review item resolved.

3.4.3.1.2. Mixed core critical power evaluations

Since the initial review of POLCA-T, the NRC staff has reviewed and approved Westinghouse's methodology to determine the MCPR of mixed cores in WCAP-18032-P-A (Ref. 83). The methodology for mixed cores is mostly dependent on **[**

]. Because POLCA-T relies on the use of an approved CPR correlation, and Westinghouse now has an approved mixed core methodology which will result in approved CPR correlations for legacy fuel assemblies, the NRC staff considers this review item resolved.

3.4.3.1.3. Counter current flow limitation revision

The review of the counter current flow limit (CCFL) correlations was discussed in Section 3.1.2.2.5 of the NRC staff's final SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68), as well as RAI 8-4. The NRC staff determined the CCFL model to be in error. In response to RAI 8-4, Westinghouse provided a commitment to update the CCFL model for application to transients and ATWS. As counter current flow did not occur for control rod drop accident (CRDA) or stability analyses, **[ ]**. In Section C.4.15 of the TR, Westinghouse confirmed that this CCFL correlation was revised **[ ]** to be consistent with the model submitted **[ ]**.

In the event that CCFL must be modeled for a particular AOO or ATWS analysis, the NRC staff notes that POLCA-T can simulate CCFL using an NRC-approved CCFL model that has been validated for the applicable fuel design. This is reflected in the AOO condition and limitation #1: If CCFL is required in any AOO or ATWS analysis performed with POLCA-T, Westinghouse should use an NRC-approved CCFL correlation for the specific fuel type and only within its validated range of applicability. With this limitation and recognizing that CCFL is not expected to be a governing phenomenon in these transients, the NRC staff considers this review item resolved.

3.4.3.1.4. Steam separator model performance

The review of the steam separator for application to stability analysis where it is important in determining the core inlet subcooling was addressed in Section 3.1.2.3.4 of the NRC staff's initial SE (Ref. 68), as well as RAIs 6-33 and 8-8. For extended power uprate (EPU)/Maximum Extended Load Line Limit Analysis Plus (MELLLA+) plants, the core outlet quality may be very large. Therefore, the NRC staff reviewed the applicability of the steam separator model to

transient applications for EPU/MELLLA+ plants in the transient application where transient power increases may result in even higher core outlet qualities. In response to RAI 12-3 for WCAP-16747-P/NP, Revision 2 (Ref. 88), Westinghouse **[**

]. Because Westinghouse has provided the underlying validation data, **[**

], and because **[**

], the NRC staff considers this review item resolved.

3.4.3.1.5. Review of POLCA-T Application to Delta over Initial Critical Power Ratio Versus Oscillation Magnitude Analysis

The review of the critical power correlations was discussed in Section 3.1.2.2 of the NRC staff's initial SE as well as RAIs 2-3, 5-4, 5-5, 6-10, and 8-1 (Ref. 68). Evaluation of the CPR is required for transient analysis and for the calculation of the Delta over Initial CPR Versus Oscillation Magnitude (DIVOM) slope. These applications were outside the scope of the previously reviewed Appendices A (CRDA analysis) and B (stability analysis) of WCAP-16747-P/NP, Revision 0 (Ref. 68). Appendix B only presented a methodology to

calculate the decay ratio for stability region determination and did not include calculation of CPR during stability events.

In RAI 12-12, the NRC staff requested clarification on whether POLCA-T is intended to be used for DIVOM calculations as part of the AOO application in WCAP-16747-P/NP, Appendix C. In the response (Ref. 76), Westinghouse indicated an intent to use POLCA-T for developing DIVOM curves to be applied to the long-term stability solution (LTSS) methodology for plant-specific applications.

DIVOM calculations are a component of certain Option III LTSS methodologies and involve calculating the transient change in CPR (i.e., delta-CPR) experienced by fuel bundles subjected to an imposed time-dependent power oscillation. This is used to determine a correlation between the power oscillation magnitude and the delta-CPR response, which is referred to as the DIVOM (delta-CPR over initial CPR versus oscillation magnitude) curve.

A separate aspect of these Option III LTSS methodologies is the determination of the hot channel oscillation magnitude (HCOM), which conservatively quantifies the maximum power oscillation magnitude that can be expected in a given plant prior to oscillation suppression via scram. This considers the plant-specific grouping of local power range monitors (LPRMs) into oscillation power range monitors (OPRM) channels, the plant-specific stability detection algorithm settings, and other physical and plant-specific aspects. The initial SE for WCAP-16747-P/NP, Revision 0 (Ref. 68), identified multiple technical items, such as bypass voiding, which may impact the sensitivity of LPRMs to oscillations. Such considerations would impact the HCOM portion of the LTSS calculation. However, since Westinghouse intends to use POLCA-T only for the DIVOM portion of the LTSS calculation and not for HCOM, the NRC staff did not review POLCA-T with respect to LPRM sensitivity and LPRM prediction capability during oscillations.

To evaluate POLCA-T's suitability for calculation of DIVOM curves, the NRC staff reviewed the POLCA-T transient CPR qualification, **[ ]**, provided in Appendix C of WCAP-16747-P/NP, Revision 2 (Ref. 70). In Section 3.4.3.1.1 of this SE, the NRC staff concluded that the CPR correlations originally validated in BISON, along with their associated uncertainty limits, are applicable when implemented in POLCA-T. That evaluation considered the transient CPR validation of POLCA-T against **[ ]**. Additional transient CPR validation cases were provided in Section C.3.1.4 of WCAP-16747-P/NP, Revision 2, which compared POLCA-T predictions against experimental **[ ]**. An evaluation of these tests is provided here to determine the acceptability of POLCA-T's transient CPR predictions during **[ ]**, including DIVOM applications. Since these tests applicable to the AOO analysis **[ ]**, an evaluation of these aspects is provided here. These aspects are also applicable to anticipated transient without scram with instability (ATWS-I). The open review item associated with these aspects for ATWS-I is identified in Section 3.5.3.1.3 of this SE.

[

]. This is a common and well-established experimental approach for **[ ]** of systems codes such as POLCA-T. Although **[**

] at which the **[**

] test setup. This is due to the absence of neutronic reactivity feedback which would otherwise allow **[** **]**. Therefore, the use of **[** **]** is expected and necessary, and consistent with the validation basis of **[** **]** which the NRC staff have previously approved.

The core axial power shape during BWR oscillations can vary depending on cycle exposure and transient conditions, with highly bottom-peaked typically most limiting in terms of oscillation severity during AOO and ATWS-I stability events. The axial power shape is an important factor impacting the axial void and pressure drop distributions and the stability behavior of BWR bundles. Therefore, the **[** **]** provides necessary confidence that POLCA-T is able to reasonably predict the stability behavior across the range of expecting axial power and void conditions.

Section C.3.1.4 of WCAP-16747-P/NP, Revision 2 (Ref. 70), provided **[**

]. The NRC staff requested additional details on the **[** **]** in RAI 12-11 in order to perform its evaluation. The RAI 12-11 response (Ref. 88) provided direct comparisons of the measured and calculated **[** **]** in each test **[** **]**. Measured and calculated **[** **]** for each test were also included. These showed the expected trends with respect to **[** **]** and provided assurance that POLCA-T produced reasonable stability behavior. **[** **]** consistent with cyclic **[** **]** can be observed in the tests, which is an expected phenomenon in many ATWS-I events and demonstrates the **[** **]** are qualitatively behaving as expected. The **[** **]** was predicted earlier in POLCA-T than in the experiments, with larger **[** **]** compared to the measured data. These trends are conservative and indicate that POLCA-T yields a reasonable prediction of **[** **]** for ATWS-I applications.

A discussion of the applicability of **[** **]** calculations is provided in Section 3.5.3.1.3 of this SE. Based on the NRC staff's review of the information provided in support of these models, including the validation of **[** **]**, the NRC staff determined that POLCA-T adequately represents the expected **[** **]** for AOO DIVOM applications. Based on **[** **]** exhibited by POLCA-T in the same validation cases, the NRC staff determined that **[** **]** is adequately represented as well. Therefore, the NRC staff concludes that this review item is resolved.

3.4.3.1.6. Reactor Internal Pump Model

The review of the reactor internal pump model was discussed in Section 3.1.2.3.6 of the NRC staff's initial SE (Ref. 68). Reactor internal pumps are not present in BWR/2–6 designs, and the scope of POLCA-T is limited to application in those plants. As a result, no reactor internal pump modeling is required, and therefore the NRC staff considers this review item resolved.

3.4.3.1.7. Dynamic Internal Bypass Channel Void Formation

The review of the dynamic internal bypass channel void formation was addressed in RAIs 2-4, 6-6, and 6-33 of the NRC staff's initial SE (Ref. 68). POLCA-T has **[** **]**, up to the degree of modeling individual internal bypass flow channels and several **[**

]. Therefore, the licensee must submit an evaluation of the impact of bypass void formation **[** **]** as part of the license amendment request or prior to implementing the method. This is reflected in the AOO condition and limitation #2 in this SE: Justification of the **[** **]** modeling for both AOO and ATWS applications must be provided for plant-specific POLCA-T applications.

Because Westinghouse must demonstrate a plant specific evaluation to confirm that bypass void sensitivity has been appropriately addressed, the NRC staff considers this review item resolved.

3.4.3.1.8. Reload Licensing Methodology for Mixed Core Applications

The review of the reload licensing methodology was addressed in RAIs 2-2 and 2-13 of the NRC staff's initial SE (Ref. 68). At the time of the NRC staff's review, the reload licensing methodology was being updated. In lieu of review of the generic reload licensing methodology, the NRC staff SE recommended specific review of implementation of POLCA-T in any fuel transition applications. Since the initial review of POLCA-T, the NRC staff has reviewed and approved Westinghouse's methodology to determine the MCPR of mixed cores in WCAP-18032-P-A (Ref. 83). Because Westinghouse now has an approved methodology for determining the safety limit for mixed cores, the NRC staff concludes that this review item has been resolved.

3.4.3.1.9. Void Reactivity Coefficient Biases and Uncertainties

The review of the void reactivity coefficient biases and uncertainties was addressed in RAIs 5-7 and 6-2 of the NRC staff's initial SE (Ref. 68). The responses to these RAIs showed that the decay ratio uncertainties are established through **[**

]. Similarly, **[**

]. The NRC staff determined that by performing this analysis, Westinghouse has appropriately accounted for the void reactivity uncertainty. Therefore, the NRC staff concluded that this review item has been resolved.

3.4.3.1.10. Lower Plenum Nodalization

The NRC staff's review of the void reactivity coefficient biases and uncertainties was addressed in RAI 6-26 of the NRC staff's initial SE (Ref. 68). In response to RAI 6-26, Westinghouse clarified that limiting feedwater and core flow conditions are established at initial steady state conditions for stability analyses. Since these limiting conditions are the initial conditions, the model did not require any transient modeling of the evolving temperature distribution in the lower plenum, or at the bundle inlet. In the initial SE, the NRC staff agreed with this assessment but concluded that the lower plenum nodalization would need to be reviewed for future applications of the POLCA-T methodology to transients or ATWS evaluations. Therefore, the NRC staff reviewed the adequacy of the lower plenum nodalization as part of this WCAP-16747-P/NP, Revision 2, review, and determined RAI 12-8 was necessary (Ref. 88).

In response to RAI 12-8 (Ref. 88) Westinghouse provided a sensitivity study which compared the impact of various lower plenum nodalizations to the behavior of multiple FoM. This study demonstrated that the main figures of merit **[** **]**. This was true across various nodalizations in the lower plenum. The largest apparent impact of nodalization occurred for **[** **]** in the core during the ATWS event. Westinghouse stated that this difference was due primarily to differences in **[** **]** which can impact **[** **]**. Further, Westinghouse stated that these variations occurred well after **[** **]** the core and effectively shut down the reactor, and, therefore, **[**

].

Additional **[** **]** considerations are evaluated in Section 3.5.3.1.1 of this SE. This includes evaluation related to **[**

].

The lower plenum nodalization also impacts ATWS events. An ATWS-I event was additionally analyzed, which showed **[**

]. The information provided in RAI 12-8 provides evidence to indicate that key figures of merit for AOOs and ATWS events may be **[** **]**. However, the lower plenum geometry, vessel configuration (e.g., SLCS injection location), and fluid conditions (flow rates, temperature, etc.) during events vary on a plant-specific basis. This is reflected in the AOO condition and limitation #3 in Subsection 4.4, "Anticipated Operational Occurrences Specific," of Section 4.0, "Summary of Conditions and Limitations," of this SE that reads:

[

].

Because this condition and limitation **[** **]**, this review item has been resolved.

3.4.3.2. Additional items for the Anticipated Operational Occurrence Transient Evaluation Model

Aside from the items mentioned above that were identified in the NRC staff's initial review of POLCA-T TR (Ref. 68), TR WCAP-16747-P/NP, Revision 2 (Ref. 70), contains details of the evaluation model in Section C.3 which the NRC staff addresses below.

3.4.3.2.1. Transient Critical Power Ratio

The transient CPR is addressed in Section C.3.1 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). As this topic has already been discussed in Section 3.4.3.1.1 of this SE, no further review is necessary.

3.4.3.2.2. Interface to Plant Control System Simulation Tool SAFIR

The interface to plant control system simulation tool SAFIR is addressed in Section C.3.2 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). SAFIR was previously reviewed and approved by the NRC staff in the final SE for WCAP-17079-P-A (Ref. 86). In response to RAI 12-16 (Ref. 75), Westinghouse confirmed the discussion provided in Appendix C of the TR was for illustration purposes only and no new review or approvals for SAFIR were requested. Therefore, the NRC staff determined that no further review is necessary.

3.4.3.2.3. Level Measurement System Model

The water level measurement system model is addressed in Section C.3.3 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). In response to RAI 12-17 (Ref. 88), Westinghouse confirmed that **[**

]. Because the SAFIR model has been previously reviewed and approved by the NRC staff in WCAP-17079-P-A (Ref. 86), and because Westinghouse has demonstrated that POLCA-T produces substantially the same results as that approved version, the NRC staff concludes that the POLCA-T water level measurement model is acceptable.

3.4.3.2.4. Advanced Control Rod Insertion Model

The advanced control rod insertion model is addressed in Section C.3.4 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). As confirmed by Westinghouse in response to RAI 12-18 (Ref. 75), approval of the advanced control rod insertion model is outside of the scope of this review because it is not applicable to BWR/2-6; therefore, the NRC staff did not review the advanced control rod insertion model.

3.4.3.2.5. **[ ]**

[

].

3.4.3.2.6. Cladding Rupture Model Uncertainties

The review of the cladding rupture model uncertainties was addressed in RAI 11-13 of the NRC staff's initial SE (Ref. 68). The cladding rupture model is **[**

]. In response to RAI-12-14 (Ref. 75), **[**

].

The NRC staff acknowledges that ongoing rulemaking activities related to Fuel Fragmentation, Relocation, and Dispersal may affect future applications of POLCA-T. However, these considerations were outside the scope of this review and would require additional NRC evaluation and approval before implementation.

3.4.3.3. Model Uncertainties for the Anticipated Operational Occurrence Transient Evaluation Model

The model uncertainties are addressed in Section C.4 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). In response to RAI 12-19 (Ref. 75), Westinghouse confirmed this section provided a description of the uncertainty quantification methodology approved in WCAP-17203-P-A (Ref. 72) and was not requesting review or approval of any new or changed methodology. Therefore, the NRC staff has confirmed that no further review is necessary.

3.4.4. Conclusions Regarding Anticipated Operational Occurrence Transient Analysis

Based on the NRC staff's initial review of the POLCA-T methodology in TR WCAP-16747, Revision 0, which was approved in 2010 (Ref. 68), the resolution of the deferred review items identified in that original POLCA-T SE reviewed in this SE in Section 3.4.3.1, the NRC staff's evaluation of the additional AOO-specific items in Section 3.4.3.2 above, and the applicability of the approved fast transient methodology in WCAP-17203-P-A (Ref. 72), the NRC staff concluded that POLCA-T has demonstrated the capability to reasonably, accurately, or conservatively model the phenomena important to AOO transient analysis subject to the conditions and limitations listed in Subsection 4.4 of this SE.

3.5. POLCA-T for Anticipated Transient Without a Scram Assessment and Acceptance Criteria

3.5.1. Introduction

WCAP-17203-P-A (Ref. 72) established Westinghouse's methodology for analyzing both limiting and non-limiting fast transients in initial and reload cores for currently operating BWRs. This included ATWS events as well as AOOs. WCAP-16747-P/NP, Revision 2, Appendices C and D demonstrate POLCA-T's ability to analyze AOO and ATWS events, respectively. Westinghouse intends to be able to reference this submittal in licensing applications for BWR/2-6 plants with homogeneous or mixed-core designs.

WCAP-16747-P/NP, Revision 2, builds upon the existing approval of POLCA-T (Ref. 68) which includes the previously approved applications to CRDA and AOO stability analysis. Appendix D

provides qualification studies and calculation methodology specific to ATWS, although the application to ATWS relies also on qualification studies provided in Appendix C for AOOs. The NRC staff considered both sets of qualification studies in its review of POLCA-T's applicability to ATWS.

3.5.2. Applicable Regulatory Basis

The NRC staff considered SRP Section 15.8, "Anticipated Transients without Scram" (Ref. 87), to identify areas of review and analysis requirements for the application of POLCA-T to ATWS analyses. Regulatory requirements for ATWS are provided in 10 CFR 50.62, "Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants." 10 CFR 50.62 defines an ATWS as an AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20.

The following acceptance criteria for ATWS are identified in SRP Section 15.8:

- 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy.
- 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry.
- GDC 12, found in Appendix A to 10 CFR Part 50, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed.
- GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary.
- GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents.
- GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reaction must be limited to negligible amounts.
- GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment.
- GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

GDC 35 requires that fuel and clad damage that could interfere with continued core cooling must be prevented (the "core coolability" requirement) and that clad metal-water reaction be limited to negligible amounts. Section 46 of 10 CFR Part 50 defines three specific core coolability criteria: (1) peak clad temperature shall not exceed 2200 degrees Fahrenheit, (2) maximum cladding oxidation shall not exceed 17 percent the total cladding thickness before oxidation, and (3) maximum hydrogen generation shall not exceed 1 percent of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen. Satisfying the 10 CFR 50.46 criteria is a way to demonstrate the core coolability requirement of GDC 35.

3.5.3. Technical Evaluation Regarding Anticipated Transient Without a Scram

3.5.3.1. NRC staff-identified Deferred Items with Respect to POLCA-T's Use as a General Code for Anticipated Transients without Scram Analysis

The original TR WCAP-16747-P, Revision 0 (Ref. 1), contained the specific application of POLCA-T to control rod drop accident analyses and stability evaluation, the NRC staff performed a separate generic review of the POLCA-T methodology per Section III.5 of SRP 15.0.2 to minimize the potential for repeated review effort for multiple applications of POLCA-T, including the NRC staff's anticipated review of POLCA-T for AOO and ATWS analyses. In that original review, the NRC staff deferred particular items needed before the acceptance of POLCA-T for AOO and ATWS analyses in Appendix D of the NRC staff's original SE (Ref. 68). Those items from the appendix that relate to POLCA-T use in ATWS are listed below.

3.5.3.1.1. Review of **[ ]**

The review of the **[ ]** was discussed in Section 3.1.2.2 of the NRC staff's initial SE (Ref. 68). However, as it was not required to simulate CRDA or reactor oscillations, its review was not completed in the initial review of POLCA-T.

Injection of **[ ]** is a key mitigation action taken during ATWS events to suppress core power and therefore limit the heatup of the suppression pool. Timely **[ ]** in sufficient quantities to maintain core subcriticality is especially important for ATWS isolation events. In such events, core power must be removed to the suppression pool without the benefit of turbine bypass heat removal.

In the PIRT for fast transients and ATWS given in Table 5-2 of WCAP-17203-P-A (Ref. 72), **[ ]**. As identified in Table 5-3 of WCAP-17203-P-A, **[ ]** effects during ATWS.

[ ]

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[ ]

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Section D.4.7 of the TR WCAP-16747-P/NP, Revision 2, provided a demonstration of this **[ ]** based on a simulated main steam line isolation valve closure (MSIVC) transient in an ABWR plant. **[ ]**. Because the ABWR injects boron above the top of active fuel, the delay time is expected to be significantly shorter for BWR/2-6 plants which inject boron into the lower plenum directly. BWR/2-6 plants which inject above the top of the active fuel may differ from the ABWR timing due to plant-specific design differences. **[ ]**.

Two key simulation uncertainties are the impacts of numerical diffusion and 3D modeling effects. Both of these uncertainties can impact the timing and slope of the boron reactivity response over time.

Numerical diffusion increases the spatial “spreading” of the boron distribution as it traverses the control volumes in the vessel. This can increase the effective speed of the boron propagation front and unphysically, non-conservatively decrease the time at which boron first reaches the core. This effect depends on the timestep size, node size, and fluid velocity (via the Courant-Friedrichs-Lewy number) and is more pronounced for larger node sizes. The NRC staff issued RAIs 12-8 and 12-9 in part to address this concern regarding numerical diffusion and boron.

The response to RAI 12-8 provides ATWS results for different upper and lower plenum nodalizations, demonstrating that the impact on the ATWS FoMs **[ ]**. The NRC staff reviewed the provided results and determined that the assessment is supported by the information presented.

[

]. The NRC staff determined that this plant-specific approach is reasonable because it accounts for plant-specific variations in lower plenum geometry, SLCS injection location, and relevant flow velocities and fluid conditions during the ATWS event. **[**

].

Regarding 3D spatial transport effects, a given concentration of boron will produce a greater (negative) reactivity impact in high-power regions of the core than in low power regions. Because the SLCS solution is injected from the radial outer portion of the lower plenum, a higher boron concentration could be expected to reach the outermost, lower power rows of core assemblies than the inner, higher power assemblies. Past studies in the literature have indicated that such spatial heterogeneities can significantly reduce the total reactivity insertion of

a given mass of boron, compared to if a homogeneous spatial distribution were assumed. This may be true for the lower plenum as well, where modeling the lower plenum as a 1D vertical flow with no radial or azimuthal dependence imposes a spatial uniformity on the boron entering the core. **[**

]. This is reflected in the ATWS condition and limitation #2:

[

] diffusion and 3D transport effects on the boron transport time and core reactivity response.

Because **[** **]** is supported by appropriate validation data, and because ATWS condition and limitation #2 ensures that **[** **]** will be appropriately considered, this review item is resolved.

[**]**

[

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[

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[

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[

], and the NRC staff determined that this approach is reasonable.

Based on the above justifications, the NRC staff concluded that **[** **]** are applicable to POLCA-T ATWS calculations, provided that justification is provided **[**

]. This is reflected in the ATWS condition and limitation #2 of this SE.

Because **[** **]** supported by appropriate validation data, and because ATWS condition and limitation #2 ensures that **[** **]** will be appropriately considered, this review item is resolved.

3.5.3.1.2. Review of **[** **]**

The review of the **[** **]** was discussed in Section 3.1.2.2 of the NRC staff's initial SE (Ref. 68) and RAI 5-8. However, the use of the **[** **]** was not required to simulate CRDA or reactor oscillations. Nevertheless, the NRC staff review contained in Appendix A of the original SE indicates that there is an assurance that the model is acceptable, however, its application would likely require treatment of uncertainty. In the staff's SE in WCAP-17203-P-A (Ref. 72), the NRC staff concluded that **[** **]**; this conclusion was further discussed in response to RAI 12-10 (Ref. 75). Because of the demonstration of the significance of the **[** **]**, this review item has been resolved.

3.5.3.1.3. Review of POLCA-T application to post-dryout conditions

The review of the post dryout heat transfer models was discussed in Section 3.1.2.2 of the NRC staff's initial SE (Ref. 68) and RAI 3-5. Post dryout heat transfer was not required to simulate CRDA or reactor oscillations. Since transient analyses are intended to demonstrate margin to critical power, the use of post dryout heat transfer is limited to ATWS applications. In the original SE, the NRC staff had reviewed information provided in response to RAI 3-5 regarding the **[** **]** qualification of the heat transfer models in POLCA-T up to the condition of dryout; however, the NRC staff's review did not constitute approval of these models generically.

Westinghouse provided additional discussion of the post-dryout modeling in response to RAI 12-11 (Ref. 76). The POLCA-T **[**

]. The post-dryout modeling consists of three main aspects: **[**

].

[**]** determination relies on the CPR correlation which is approved by the NRC on a fuel type specific basis. As discussed in Section 3.4.3.1.1 of this SE, the NRC staff concluded that the CPR correlation for the SVEA-96 Optima2 and 3 and TRITON11 fuel types provides acceptable predictions of transient CPR. **[**

]. These models

were approved for BISON, and the NRC staff concludes that the models represent post-dryout phenomena in a realistic manner and remain acceptable for POLCA-T ATWS application.

In BISON, **[**
]. How a methodology is chosen depends on processes described in WCAP-17203-P-A (Ref. 72). In RAI 12-11 (Ref. 88), the NRC staff requested information on what approach was used for POLCA-T. Westinghouse indicated in the response to RAI 12-11 that an unbiased approach was used for all ATWS calculations and qualification studies in Appendix D of WCAP-16747-P/NP, Revision 2 and in the RAI responses. The NRC staff reviewed the RAI response and determined that this approach is appropriate because it allows for unbiased comparisons of POLCA-T's predictions against the qualification data. POLCA-T's predictions for **[**

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In **[**

].

In RAI 12-20 (Ref. 88), the NRC staff requested POLCA-T demonstration results for a representative limiting ATWS-I event in order to better assess POLCA-T's ability to predict key ATWS-I physical phenomena in a reasonable manner. In the response to RAI 12-20, Westinghouse provided POLCA-T results for **[**

]. Westinghouse provided these results for illustrative purposes only and it was not intended to represent an ATWS-I event at any particular plant.

Upon review, the NRC staff determined that this simulation showed general similarities to expected plant response during realistic ATWS-I events. This included the general behavior of reduction in core flow rate early in the event, gradual increase in core inlet subcooling throughout the event, and oscillations which increase in magnitude leading to cyclic dryout-rewet and potentially failure-to-rewet. However, **[**
].
Core oscillations typically occur either as in-phase oscillations, with the flow rate in all bundles oscillating in phase with each other, or out-of-phase oscillations such that one half of the core oscillates out of phase with the other half, separated by a symmetry line through the middle of the core. This common form of out-of-phase oscillations is associated with the first harmonic mode of the neutron flux distribution. The behavior seen in the Westinghouse's response to RAI 12-20 with **[**

].

Another important consideration is that the POLCA-T simulation **[**
] established. Failure-to-rewet, if it occurs during ATWS-I, leads to significantly increased PCT which in some plants may challenge the coolable geometry

acceptance criterion of 10 CFR 50.46. Analysis codes must demonstrate an ability to perform reasonable prediction of the cladding heatup behavior in the failure-to-rewet regime for this reason. **[**

]. This is reflected in the ATWS condition and limitation #1 focused on MELLLA+ application in Section 4.5 of this SE.

Section 3.5.3.1.4 of this SE provides the NRC staff's evaluation of POLCA-T's transient CPR modeling **[**

].

Based on the **[** **]** test and **[** **]** analyses, the NRC staff concluded that POLCA-T can accurately or conservatively predict **[**

]. However, the idealized nature of the ATWS-I simulations provided in response to RAI 12-20 (Ref. 88) limited the NRC staff's ability to evaluate POLCA-T's behavior under realistic ATWS-I conditions. Furthermore, insufficient information was provided to evaluate the code's **[**

] should such conditions arise, as part of plant-specific POLCA-T ATWS-I applications. This is reflected in the ATWS condition and limitation #1 focused on MELLLA+ application.

Because of the **[** **]** of the POLCA-T **[** **]** behavior, and because review of **[** **]** and limitation #1, this review item has been resolved.

3.5.3.1.4. Counter current flow limitation revision

The counter current flow limitation (CCFL) correlation was addressed in Section 3.4.3.1.3 of this SE above. No further review was required for modeling of the CCFL revision for ATWS.

3.5.3.1.5. Steam separator model performance

The steam separator model performance was addressed in Section 3.4.3.1.4 of this SE above. No further review was required for steam separator model performance for ATWS.

3.5.3.1.6. Reactor Internal Pump Model

The reactor internal pump model was addressed in Section 3.4.3.1.6 of this SE above. No further review was required for the reactor internal pump model for ATWS.

3.5.3.1.7. Review of POLCA-T application for exothermic metal-water reaction

The review of the exothermic metal-water reaction model was discussed in Sections 3.1.2.6 and 3.2.1.7 of the NRC staff's initial SE (Ref. 68), as well as RAIs 9-1 and 11-12. Use of the exothermic metal water reaction model is only required for applications where significant, gross fuel cladding heat up occurs. This situation is limited to the ATWS application.

[

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[

]. Because Westinghouse is using previously approved models with appropriate uncertainty quantification, the NRC staff considers this review item resolved.

3.5.3.1.8. Dynamic internal bypass channel void formation

The dynamic internal bypass channel void formation was addressed in Section 3.4.3.1.7 of this SE. No further review was required for modeling of the dynamic internal bypass channel void formation for ATWS.

3.5.3.1.9. Reload licensing methodology

The reload licensing methodology was addressed in Section 3.4.3.1.8 of this SE. No further review was required related to the reload licensing methodology for ATWS.

3.5.3.1.10. Void reactivity coefficient biases and uncertainties

The void reactivity coefficient biases and uncertainties were addressed in Section 3.4.3.1.9 of this SE. No further review was required related to the void reactivity coefficient biases and uncertainties for ATWS.

3.5.3.1.11. Lower Plenum Nodalization

The lower plenum nodalization was addressed in Section 3.4.3.1.10 of this SE. That evaluation considered the impact of lower plenum nodalization for both AOOs and ATWS events. However, additional review of **[ ]** was required to address all concerns regarding ATWS events. This additional review was related to lower plenum nodalization as well as **[ ]**. This review is provided in Section 3.5.3.1.1 of this SE. No further review was required related to the lower plenum nodalization.

3.5.3.2. Additional items for the ATWS Evaluation Model

The following contains modifications or additions to the ATWS EM that were not described in the original submittal of POLCA-T (Ref. 68).

3.5.3.2.1. **[ ]**

The **[ ]** is described in Section D.3.1 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). The NRC staff's evaluation of this topic is contained in Section 3.5.3.1.1 of this SE. No further review was required for the **[ ]** for ATWS.

3.5.3.2.2. **[ ]**

The **[ ]** is described in Section D.3.2 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). The NRC staff's evaluation of this topic is contained in Section 3.5.3.1.1 of this SE. No further review was required for the **[ ]** for ATWS.

3.5.3.2.3. Reactivity model of bypass channel coolant density

The reactivity model of bypass channel coolant density is addressed in Section D.3.3 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). The NRC staff's evaluation of this topic is contained in Section 3.4.3.1.7 of this SE. No further review was required for the reactivity model of bypass channel coolant density for ATWS.

3.5.3.2.4. Post dryout and rewet model

The post dryout and rewet model is contained in Section D.3.4 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). The NRC staff's evaluation of these topics is contained in Sections 3.4.3.1.5 and 3.5.3.1.3 of this SE. No further review was required for the post dryout and rewet model for ATWS.

3.5.3.2.5. Calculation of the decay ratio and oscillation frequency

The calculation of the decay ratio and oscillation frequency is addressed in Section D.3.5 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). This method is applied as a post-processing step after the POLCA-T simulation is complete to quantify the decay ratio and frequency of the oscillations that were calculated for the event. Because it is a post-processing step, it does not impact the POLCA-T analysis itself and has no effect on the cladding temperature evolution and other ATWS event outcomes. In response to RAI 12-21 (Ref. 75), Westinghouse confirmed that the decay ratio and oscillation frequency calculation is performed for informational purposes only within the ATWS methodology.

As part of the NRC staff's previous approval of POLCA-T for stability calculations in Appendix B of WCAP-16747-P-A (Ref. 68), the NRC staff approved the decay ratio calculation methodology for the determination of stability region boundaries. **[**

]. The decay ratio and frequency calculation method in Section D.3.5 of TR WCAP-16747-P/NP, Revision 2 (Ref. 70), includes an ability to predict these quantities **[**

]

related approaches commonly used in the literature.

[

]. However, the response to RAI 12-21 clarified that the details of this **[ ]** decay ratio calculation were provided for informational purposes. It does not impact the calculation of key FoMs such

as peak cladding temperature which determine whether the acceptance criteria are satisfied. Therefore, the NRC staff did not perform a detailed review of the methodology in Section D.3.5. Instead, the currently approved decay ratio calculation methodology based on Appendix B of WCAP-16747-P-A is to be used for stability region calculations.

[

], the NRC staff considers this review item to be resolved.

3.5.3.2.6. Other phenomena relevant to ATWS

Other phenomena not identified in Appendix D but relevant to ATWS are addressed in Section D.3.6 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). This includes **[**

]

[

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Based on the justifications given above, the NRC staff concluded that the “other phenomena” described in Section D.3.6 of WCAP-16747-P/NP, Revision 2, are reasonably modeled and therefore the methodology is acceptable.

3.5.3.3. Model Uncertainties for the Anticipated Transients Without Scram Evaluation Model

The model uncertainties are addressed in Section D.4 of the TR WCAP-16747-P/NP, Revision 2 (Ref. 70). In response to RAI 12-19 (Ref. 75), Westinghouse confirmed this section was providing a description of the uncertainty quantification methodology approved in TR WCAP-17203-P-A (Ref. 72) and was not requesting review or approval of any new or changed methodology. Therefore, the NRC staff has confirmed that no further review is necessary.

3.5.4. Conclusions Regarding Anticipated Transient Without a Scram

Based on the NRC staff’s initial review of the POLCA-T methodology (Ref. 68), the resolution of the previously deferred review items identified in the original POLCA-T SE given in Section 3.5.3.1, and the NRC staff’s evaluation of the additional ATWS-specific items in Section 3.5.3.2, the NRC staff determined that POLCA-T has demonstrated the capability to reasonably, accurately, or conservatively model the phenomena important to ATWS analysis subject to the conditions and limitations listed in Section 4.5 of this SE.

4. SUMMARY OF CONDITIONS AND LIMITATIONS

Direct quotation from the NRC staff final SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68) is provided in *italics* below:

This section of the SE provides a comprehensive listing of the conditions, limitations, and restrictions listed in the body of this SE. The conditions, limitations, and restrictions are divided into sections according to their applicability, either general or application specific.

While the conditions and limitations are grouped into five sections based on the aspect of POLCA-T they most directly address, each condition should be considered applicable to all uses of POLCA-T where the underlying issue is relevant. The section in which a condition appears should not be interpreted as restricting its scope.

4.1. General Usage

The NRC staff's review of this section is contained in Section 4.1 of the NRC staff's original SE (Ref. 68).

4.2. Control Rod Drop Accident Specific

The NRC staff's review of this section is contained in Section 4.2 of the NRC staff's original SE (Ref. 68).

4.3. Stability Specific

The NRC staff's review of this section is contained in Section 4.3 of the NRC staff's original SE (Ref. 68).

4.4. Anticipated Operational Occurrences Specific

1. If CCFL is required in any AOO or ATWS analysis performed with POLCA-T, Westinghouse should use an NRC-approved CCFL correlation for the specific fuel type and only within its validated range of applicability.

2. Justification of the **[** **]** modeling for both AOO and ATWS applications must be provided for plant-specific POLCA-T applications.

3. Justification of the **[**

].

4.5. Anticipated Transients Without Scram Specific

1. Plant-specific application of POLCA-T to MELLLA+ operation must include a review of the limitations and conditions (L&Cs) in the MELLLA+ TR (NEDO-33006-A, Revision 3, ML091800513 (Ref. 89)), including the seven L&Cs within Section 1.0 of the TR and the 24 L&Cs within the NRC staff's SE. This review must include a dispositioning of which L&Cs are applicable to the POLCA-T MELLLA+ application and indicating how these L&Cs have been addressed. For example, justification

must be provided that the plant-specific ATWS stability calculations appropriately model all plant design and operational features that impact the ATWS stability response as described in L&C 12.19 of the MELLLA+ TR SE. The NRC anticipates that, if a POLCA-T MELLLA+ application provides a disposition of this L&C with a technical basis broadly applicable across multiple plants, that resolution may be relied upon in future submittals and, where still applicable, may be referenced in further applications to demonstrate L&C compliance. **[**

] Section C.4.24 of WCAP-16747-P/NP, Revision 2.

2. Plant-specific applications of POLCA-T must justify **[**
] calculations based on the plant-specific vessel configuration and the ATWS **[** **]** predicted for that plant. This must include justification that **[**
] and core reactivity response.

5. CONCLUSIONS

The NRC staff has completed its review of Westinghouse TR WCAP-16747-P/NP, Appendices C and D, Revision 2 (Ref. 70), which documents the qualification of the POLCA-T system analysis code for application to AOO transient analyses and ATWS analyses. This SE builds upon the previously approved POLCA-T methodology and supplements the original NRC approval of WCAP-16747-P-A, Revision 0 (Ref. 68), by assessing the additional capabilities introduced in Appendices C (AOO) and D (ATWS).

For the AOO evaluation model, the NRC staff reviewed the resolution of all previously deferred review items from the original POLCA-T approval, evaluated the code's ability to model the phenomena identified in the approved fast transient methodology of WCAP-17203-P-A, Revision 0-2 (Ref. 72), and assessed additional AOO-specific modeling elements. Based on this review and the stated conditions and limitations, the NRC staff concludes that POLCA-T has demonstrated the capability to reasonably, accurately, or conservatively model the phenomena important to AOO transient analyses.

For the ATWS evaluation model, the NRC staff reviewed resolution of the deferred ATWS-related items from the original approval, evaluated the ATWS-specific models provided in Appendix D (including **[**
]), and considered the applicability of the WCAP-17203-P-A, Revision 0-2, fast transient methodology to ATWS analyses. Based on the qualification studies, supplemental RAI responses, and the limitations imposed for plant-specific applications, the NRC staff concludes that POLCA-T has demonstrated the capability to reasonably, accurately, or conservatively model the key phenomena important to ATWS events.

Therefore, the NRC staff concluded that TR WCAP-16747-P/NP, Appendices C and D, Revision 2, "POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D," is acceptable for referencing in licensing applications for U.S. BWR/2–6 plants with homogeneous or mixed-core designs, subject to the conditions, limitations, and restrictions identified in Section 4, "Summary of Conditions and Limitations," of this SE. Any

future application of the methodology must address these conditions and confirm that plant-specific modeling assumptions and configurations remain within the validated range of applicability.

If NRC criteria, regulatory requirements, or acceptance bases applicable to this methodology change in the future such that the conclusions of this SE are invalidated, licensees referencing TR WCAP-16747-P/NP, Appendices C and D, Revision 2, "POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D," will be expected to either revise and resubmit their respective documentation or justify continued applicability of the methodology without revision.

6. REFERENCES

The NRC staff's References 1-67 are listed in Section 6 of the NRC staff's SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68). Specific references from References 1-67 that are cited in this SE for WCAP-16747-P, Revision 2, "POLCA-T System Analysis Code with Tree-Dimensional Core Model" (Ref. 70), are reproduced here *in italics* for reference purposes.

1. *WCAP-16747-P, "POLCA-T: Systems Analysis Code with Three-Dimensional Core Model," Westinghouse Electric Company, March 2007, LTR-NRC-07-19 (Agencywide Documents Access and Management System (ADAMS) No. ML071370710.*

8. *CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB CE, July 1996 (ADAMS Accession No. ML072250429).*

23. *NRC, Section 15.0.2, Revision 0, "Review of Transient and Accident Analysis Method," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," December 2005 (ML053550265).*

References 2 through 7, 9 through 22, and 24 through 67 are listed in Section 6 of the NRC staff's SE for WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Ref. 68).

References unique to this SE begin with Reference 68.

68. Letter from Gresham, J. A. (Westinghouse) to NRC, "Submittal of Approved Version of WCAP-16747-P-A, Revision 0 & WCAP-16747-NP-A, Revision 0, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model,'" September 14, 2010 (ML102770087).

69. Letter from Harper, Z. (Westinghouse), to NRC, "Submittal of Westinghouse Topical Report WCAP-16747-P/-NP Appendices C and D, Revision 2, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D' (Proprietary/Non-Proprietary)," LTR-NRC-24-21, June 19, 2024 (ML24172A260).

70. Letter from Harper, Z. (Westinghouse), to NRC, "Resubmittal of Submittal of Westinghouse Topical Report WCAP-16747-P/-NP, Appendices C and D, Revision 2, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D' (Proprietary/Non-Proprietary)," LTR-NRC-24-21 Revision 1, September 23, 2024 (ML24267A195; ML24267A194 Package).

71. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”

72. Letter from Gresham, J. A. (Westinghouse) to NRC, “Submittal of WCAP-17203-P-A/NP-A, Revision 0-2, “Fast Transient and ATWS Methodology,” November 5, 2018 (ML18318A109).

73. Letter from George, G. (NRC), to Westinghouse, “Acceptance for Review of Westinghouse Electric Company Topical Report WCAP-16747-P/NP, Revision 2, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D’ (EPID L-2024-TOP-0025)’ (Proprietary/Non-Proprietary),” September 25, 2024 (ML24267A286).

74. Email from Lenning, E. (NRC), to Jerrod Ewing (Westinghouse), “U.S. NRC Requests for Additional Information for Westinghouse Topical Report WCAP-16747-P/NP, Revision 2, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D’ (EPID: L-2024-TOP-0025),’” March 24, 2025 (ML25083A073 Package).

75. Letter from Ewing, J. (Westinghouse), to NRC, “Submittal of Set 1 of Responses to Requests for Additional Information on Westinghouse Topical Report WCAP-16747-P/NP Appendices C and D, Revision 2, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D’ (Proprietary/Non-Proprietary),” LTR-NRC-25-29, June 9, 2025 (ML25161A188 Package).

76. Letter from Ewing, J. (Westinghouse), to NRC, “Submittal of Set 2 of Responses to Requests for Additional Information on Westinghouse Topical Report WCAP-16747-P/NP, Appendices C and D, Revision 2, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D’ (Proprietary/Non-Proprietary),” LTR-NRC-25-35, July 3, 2025 (ML25184A393 Package).

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78. WCAP-16081-P-A, “10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2,” Westinghouse, March 2005 (ML003676083).

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80. WCAP-16081-P-A, Addendum 1-A, “SVEA-96 Optima2 CPR Correlation (D4): High and Low Flow Applications,” Westinghouse, March 2009.

81. Westinghouse, Submittal of WCAP-17794-P-A and WCAP-17794-NP-A, Revision 0, “10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 Optima3,” October 2020 (ML20283A815 Package)

82. Westinghouse, Submittal of WCAP-18904-P/NP-A, Revision 0, “Critical Power Experiments and D6 CPR Correlation for TRITON11 Fuel,” October 2025 (ML25318A739).

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84. Letter from Ewing, J. (Westinghouse), to NRC, “Submittal of Westinghouse Topical Report WCAP-18950-P/NP, Revision 0, Westinghouse BWR ECCS Evaluation Model: Supplement 6 to Code Description, Qualification and Application to TRITON11 Fuel Design,” LTR-NRC-25-20, March 26, 2025 (ML25085A311).

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87. NRC, Section 15.8, Revision 2, “Anticipated Transients without SCRAM,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2007 (ML070570008).

88. Letter from Ewing, J. (Westinghouse), to NRC, “Submittal of Updated Responses to Requests for Additional Information on Westinghouse Topical Report WCAP-16747-P/NP, Revision 2, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model - App C and D’ (Proprietary/Non-Proprietary)” LTR-NRC-26-23, June 2026 (ML26153A187).

89. GE Hitachi Nuclear Energy, “Accepted Version of GE Licensing Topical Report NEDO-33006-A/NEDC-33006P-A, Revision 3,” June 2009 (ML091800530 Package).

U.S. Nuclear Regulatory Commission Resolution of

Westinghouse Electric Company Proprietary Markings and Voluntary Comments

By letter dated October 7, 2026, Westinghouse Electric Company (Westinghouse) submitted proprietary markings and voluntary comments (ML26282A000; LTR-NRC-26-51) following the U.S. Nuclear Regulatory Commission (NRC) issuance on August 6, 2026 (ML26160A244 - Cover Letter), of the proprietary draft safety evaluation (SE) for proprietary review.

The NRC staff reviewed and incorporated Westinghouse's proprietary markings requests for additional markup, and voluntary minor editorial comments into the final SE.