FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16747-P

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

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

EXECUTIVE SUMMARY

Westinghouse Electric Company (Westinghouse) submitted WCAP-16747-P —POLCA-T: Systems Analysis Code with Three Dimensional Core Model" Topical Report (TR), for U.S. Nuclear Regulatory Commission (NRC) staff review for applications to boiling-water reactor (BWR) control rod drop accident (CRDA) analysis and stability evaluation (Reference 1). The submittal includes a primary document that provides a description of the field equations, closure relationships, the numerical solution techniques, and references to the origins of those models. The submittal also includes Appendices A and B. Appendix A of the TR includes the methodology description for the application of POLCA-T to CRDA analysis and an assessment that provides the basis for the uncertainty determination and associated acceptance criteria. Appendix B of the TR includes the methodology description for the application of POLCA-T for stability evaluation and an assessment that provides the basis for the uncertainty determination and associated acceptance criteria.

The POLCA-T methodology is based, in part, on the combination of codes and methods that have previously been approved by the NRC. The neutronic methodology is based on POLCA7 and the steam line models and SCRAM system based on BISON. Westinghouse requested that the NRC staff review the POLCA-T methodology as a general purpose methodology with specific applications to CRDA analysis and stability evaluation.

The POLCA-T code structure, in a basic sense, iteratively couples the POLCA7 nodal diffusion code with the RIGEL thermal-hydraulics code. The two codes are run iteratively to couple the transient thermal-hydraulic fluid state to the transient core power distribution and reactivity. As RIGEL has not been previously approved, the staff reviewed the RIGEL thermal-hydraulic models as part of this review and determined these models are to be acceptable for the current application purposes.

The POLCA-T neutronic model is derived from POLCA. POLCA is a three-dimensional diffusion theory code used to predict the core and pin-wise power distribution. POLCA is a steady state code using the analytical nodal method (Reference 39) with assembly discontinuity factors and a thermal-hydraulic model based on the CONDOR code (Reference 40). While the neutronic model of POLCA is utilized in POLCA-T, the thermal-hydraulic model in POLCA-T is based on RIGEL.

The POLCA-T neutronic model is used to determine the core reactivity and power distribution. The fundamental engine for solving the power distribution is the POLCA code. The POLCA code, however, receives nuclear parameter input from the upstream two-dimensional lattice analysis code PHOENIX.

The POLCA-T thermal-hydraulic model is based on a two-fluid approach. The energy and mass are solved for both the liquid and vapor phases that may be in thermal non-equilibrium. Additionally POLCA-T includes optional models for **[** **]** POLCA-T is based on a five-equation, one-dimensional thermal hydraulic approach.

The development of an evaluation model for use in reactor safety licensing calculations requires a substantial amount of documentation. This documentation includes/covers (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program. The NRC staff conducted a review of the POLCA-T code in accordance with Section 15.0.2, —Review of Transient and Accident Analysis Methods", of NUREG-0800, –Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP 15.0.2). (Reference 23)

Section III.5 of SRP 15.0.2 states that: —often a general purpose transient analysis computer program is developed to analyze a number of different events for a wide variety of plants. These codes can constitute the major portion of an evaluation model for a particular plant and event."

Generic reviews are often performed for these codes to minimize the amount of work required for plant- and event-specific reviews.

Noting that the POLCA-T code is a general purpose code, the NRC staff conducted its review of the generic elements of the code as well as separate detailed reviews of the specific application of the POLCA-T method to perform CRDA analyses and stability evaluations. In particular cases where the NRC staff could not complete its generic review of particular aspects of the POLCA-T computer program, the NRC staff provided a description of the basis for the NRC staff deferral of the particular aspect and documented these items in Appendix D of this safety evaluation (SE).

During the conduct of this review the NRC staff identified conditions, limitations, and restrictions to the method. These conditions, limitations, and restrictions may be specific to the type of analysis performed or may be of a generic nature.

In terms of analysis specific conditions, limitations, and restrictions, the NRC staff identified key analysis inputs that are necessary to appropriately evaluate the conditions specific to that particular analysis. As an example, the CRDA analyses require that the control rod drop velocity be input to assess the rate of reactivity insertion. The NRC staff has imposed a condition on this parameter due to differences between the advanced BWR (ABWR) and BWR/2-6 control blade designs. Another example is in the stability application. Time domain stability analyses are sensitive to the numerical solution technique and inappropriate controls on the time integration may result in numerical damping of simulated reactor oscillations.

In terms of generic conditions, the NRC staff often imposes conditions, limitations, and restrictions regarding the use of particular models within the code when several options are available. For example, a systems analysis code such as POLCA-T may include different models for the fuel rod performance that are identified by code input. As an example, the NRC staff imposed the condition that those fuel models consistent with the NRC approved STAV7.2 code be used. Additionally, the NRC staff understands that during routine code maintenance activities that particular aspects of the code may be updated or changed that do not impact the methodology. However, the NRC staff often imposes conditions on code changes to provide clarification of the criteria of Title 10 of the Code of Federal Regulations (10 CFR) 50.59 as it applies to code changes. As an example, the NRC staff finds that code updates that render models in the code inconsistent with the approved TR documentation, constitute a departure from a method of evaluation.

The NRC staff has clearly marked the conditions, limitations, and restrictions throughout the body of the SE and repeated these conditions in each subsection of the SE. When the POLCA-T methodology is exercised within these conditions, limitations, and restriction, the NRC staff finds that the method is acceptable for reference to perform licensing calculations for CRDA analysis and stability evaluation.

CONTENTS

1 INTRODUCTION

Westinghouse Electric Company (Westinghouse) submitted WCAP-16747-P —POLCA-T: Systems Analysis Code with Three Dimensional Core Model" TR, for NRC staff review for applications to boiling-water reactor (BWR) control rod drop accident analysis and stability evaluation (Reference 1). The submittal includes a primary document that provides a description of the field equations, closure relationships, the numerical solution techniques, and references to the origins of those models. The submittal also includes Appendices A and B. Appendix A of the TR includes the methodology description for the application of POLCA-T to control rod drop accident (CRDA) analysis and an assessment that provides the basis for the uncertainty determination and associated acceptance criteria. Appendix B of the TR includes the methodology description for the application of POLCA-T for stability evaluation and an assessment that provides the basis for the uncertainty determination and associated acceptance criteria.

The POLCA-T methodology is based, in part, on the combination of codes and methods that have previously been approved by the NRC. The neutronic methodology is based on POLCA7 and the steam line models and SCRAM system based on BISON. Westinghouse requested that the NRC staff review the POLCA-T methodology as a general purpose methodology with specific applications to CRDA and stability evaluation. Westinghouse intends to supplement WCAP-16747-P with additional appendices to address the application of the general purpose methodology to transient anticipated operational occurrence (AOO) analysis and anticipated transient without SCRAM (ATWS) analysis.

2 REGULATORY EVALUATION

Section 50.34 of Title 10 of the *Code of Federal Regulations* (10 CFR) —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 stability to demonstrate compliance with GDC 10 and 12.

3 TECHNICAL EVALUATION

The NRC staff reviewed WCAP-16747-P in accordance with Standard Review Plan (SRP) Section 15.0.2 —Review ofTransient and Accident Analysis Methods" (Reference 23). 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.

3.1 POLCA-T as a General Purpose Method

While the licensing topical report (Reference 1) contains 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 AOO and ATWS analyses. In cases where the NRC staff has deferred the review of particular items, these deferrals and relevant Westinghouse commitments are summarized in Appendix D of this SE.

3.1.1 Documentation

The development of an evaluation model for use in reactor safety licensing calculations requires a substantial amount of documentation. This documentation includes/covers (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program. In accordance with SRP 15.0.2 the NRC staff conducted a review of the documentation of the POLCA-T code.

The NRC staff performed a detailed audit of the code documentation, including theory and user manuals, code change assessment reports, and internal Westinghouse procedures that govern the quality assurance processes for POLCA-T. The results of the NRC staff audit of these documents are provided in Reference 47**.** The NRC staff found that the code documentation was complete in its description of the theory, assessment, and user guidance. The audit also covered topics related to the code stewardship quality assurance program. The outcome of the NRC staff audit regarding the quality assurance plan is briefly summarized in Section 3.1.5 of this SE.

3.1.2 Evaluation Model

The POLCA-T code structure, in a basic sense, iteratively couples the POLCA7 nodal diffusion code with the RIGEL thermal-hydraulics code. The two codes are run iteratively to couple the transient thermal-hydraulic fluid state to the transient core power distribution and reactivity.

3.1.2.1 Neutronic Model

POLCA is a three-dimensional diffusion theory code used to predict the core and pin-wise power distribution. POLCA is a steady state code using the analytical nodal method (Reference 39) with assembly discontinuity factors and a thermal-hydraulic model based on the CONDOR code (Reference 40).

The POLCA-T neutronic model is used to determine the core reactivity and power distribution. The fundamental engine for solving the power distribution is the POLCA code. The POLCA code, however, receives nuclear parameter input from the upstream two-dimensional lattice analysis code PHOENIX.

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The NRC staff reviewed and approved the original version of PHOENIX in 1985 (Reference 19). **[**

] In 2000 the NRC staff reviewed an update to the PHOENIX code (Reference 9). **[**

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No changes have been submitted to the NRC for the PHOENIX methodology as part of this application; therefore, the NRC staff finds that the PHOENIX code is acceptable for use as an upstream lattice parameter analysis tool for the POLCA-T code system when exercised within the conditions and limitations specified in the NRC staff's safety evaluation report (SER) for Reference 9. The NRC staff notes that the cross section functional fitting process accounts for exposure, void, void history, and Doppler effects as well as including nuclide tracking capabilities for cross section adjustments and includes terms for control blade corrections and control blade history corrections. All of these models have previously been reviewed and accepted by the NRC staff (Reference 9).

The original POLCA method was reviewed by the NRC staff with the original version of PHOENIX as documented in Reference 19. **[**

] (Reference 20).

Substantial improvements were made in the POLCA code and the methodology was submitted to the NRC for review and approval in 2000. The NRC staff review of the methodology identified the following as substantial improvements in the POLCA methodology comprising the POLCA7 method:

- An enhanced cross section treatment based on microscopic and macroscopic cross sections to accurately accommodate the impact of various effects including those due to control rods and spacer grids. The spectral history is specifically accounted for by solving depletion chains for heavy nuclides and fission products. This treatment allows a substantially more accurate treatment of spectral and burnup effects.

- The new POLCA version utilizes a full two-group diffusion theory model rather than the modified one-group model. The use of discontinuity factors and burnup dependent spatial cross section variation provides accurate nodal power distributions and a firm basis for pin power reconstruction.

- The new POLCA version has the capability of utilizing pin power reconstruction to accommodate the effect of neutron leakage from adjacent assemblies on pin powers.

The use of the PHOENIX4/POLCA7 code system was previously approved for BWR reload licensing analysis with the following limitations and conditions as specified in the NRC staff's SE (Reference 9)[1]:

- When applying PHOENIX4/POLCA7 to transition cores, CENP should use fuel-specific data to model the thermal and hydraulic behaviors of the non-ABB/CE fuel and confirm that the uncertainties derived for ABB fuel are applicable to the non-ABB/CE fuel.

- PHOENIX4/POLCA7 are approved for analysis of ABB/CE fuel types up to and including 10X10 lattices with a maximum enrichment of 5 weight percentage (w/o). Non-ABB/CE fuel types may be analyzed assuming that analyses are performed consistent with the above condition. The code is approved for application to fuel with burnable absorbers composed of a mixture of uranium oxide and gadolinia with concentrations up to 9 w/o. Application of the code to non-oxide fuel or the fuel using burnable poisons other than gadolinia will need to be justified.

- When applying the PHOENIX4/POLCA7 code to fuel other than what is approved in this SE, the NRC should be informed by letter of this application and be provided an opportunity for review.

- PHOENIX4/POLCA7 contains several models for BWR analysis not used to generate the information contained in the topical report. If CENP determines that one of these models is needed for a licensing analysis, the NRC staff should be informed of the application and be given an opportunity for review.

Based on the nature of the integration of POLCA7 in the POLCA-T methodology, the NRC staff finds that these historical limitations and conditions are likewise directly applicable to POLCA-T. The POLCA-T methodology, however, is built upon several codes that have been previously reviewed and approved by the NRC staff besides PHOENIX4 and POLCA7. Therefore, the NRC staff imposes a generalized condition on POLCA-T in regards to these encompassed codes.

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[1] CENP [Combustion Engineering] and ABB [ASEA Brown Boveri] have been procured by Westinghouse. Reference to these entities herein is only for consistency with the previous SE language.

Applicability of Conditions and Limitations on Encompassed Codes

Licensees implementing POLCA-T should provide justification that STAV7.2, PHOENIX4, POLCA7, and PARA computer codes and methodology, when approved in the licensing basis for use, are utilized in a manner that is in compliance with the conditions identified in the NRC staff Safety Evaluation (SE). The exception to this is called out in the response to RAI 11-15 (Reference 67).

If a specific plant has not been licensed for the use of the computer codes and methodology that are utilized by POLCA-T, then that licensee will need to take appropriate licensing action for application of these computer codes. Licensees will need to verify that the conditions and limitations imposed on each of the NRC approved codes (STAV7.2, PHOENIX4, POLCA7, and PARA), encompassing the POLCA-T methodology will continue to be satisfied each time the POLCA-T methodology is utilized.

In request for additional information (RAI) 5-3, the NRC staff requested additional qualification data if approval of POLCA-T was sought for application to mixed oxide (MOX) fuel. The response stated that review for MOX applications is not being sought by the current application (Reference 48). Therefore, the NRC staff imposes the following restriction.

Mixed oxide Restriction[2]
POLCA-T is not approved to analyze cores containing MOX fuel.

The bundle power uncertainty for the PHOENIX4/POLCA7 code system was assessed against
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The POLCA-T TR (Reference 1) references the PHOENIX4/POLCA7 neutronic methods as described in Reference 9. The assessment databases provided in References 9 and 19 are used to determine uncertainty values for those particular parameters included in the determination of cycle-specific specified acceptable fuel design limits (SAFDLs).

In general, those uncertainties that are related to the efficacy of the neutronic methods are the bundle power and pin power uncertainties. The uncertainties used in the evaluation of SAFDLs are a combination of the calculational uncertainty associated with the neutronic solver as well as uncertainty factors related to the core monitoring methodology and plant instrumentation (including failed instruments and core power shape measurement effects).

For the determination of safety limits, such as the safety limit minimum critical power ratio (SLMCPR) per Reference 8, a cycle-specific Monte Carlo analysis is performed based on established uncertainties in critical model input parameters to determine a probability distribution for the safety parameter (such as minimum critical power ratio (MCPR), from which the SLMCPR can be derived statistically).

[2] See also the NRC staff's review of the response to RAI 5-3 in Appendix A.

The NRC staff therefore reviewed the application of the neutronic methods to expanded operating domain reactors. The purpose of the review is to address the application of the neutronic models to conditions at expanded operating domain conditions that may result in increased uncertainties in pin or bundle power that must be accounted for in the SAFDLs when POLCA-T is applied to these conditions.

Given the neutronic aspects of extended power uprate with a maximum extended load line limit analysis or extended power uprate with a maximum extended load line limit analysis plus (EPU/MELLLA or EPU/MELLLA+) cores relative to previous core designs, the NRC staff has performed an evaluation of the applicability of the previously approved nuclear design codes to the neutronic conditions present in such core designs.

Extended Power Uprate (EPU) cores are generally designed by flattening the radial core power shape relative to a pre-EPU core. In doing so, the highest power bundle tends to remain the most limiting bundle, while other non-limiting bundles have increased power. To sustain the higher core power level through the same cycle duration, the core must be a high energy core. A high energy core has significant reactor physics attributes that differentiate such a design from a pre-EPU, pre-extended cycle core.

High energy cores require high burnable poison loadings. The high loadings are necessary to compensate for the additional excess reactivity necessary to sustain core criticality for the same cycle duration with a higher thermal power. In addition to these high burnable poison loadings, a larger fraction of assemblies are loaded, typically, in each cycle to also increase the core cycle energy. High energy cores are typically depleted in a spectral shift manner to maintain core power while achieving the desired duration.

A combination of higher batch reload fraction and a higher loading of neutron poison tends to harden the neutron spectrum during cycle exposure. Additionally, as the average bundle power is increased, the core average void fraction tends to increase. The combination of higher inventories of thermal neutron absorbers, more fissile content, and higher void fractions may result in a hard spectrum that can result in uncertainties in important neutronic parameters over exposure that have not been previously quantified or accounted for based on operating experience in a much softer exposure-averaged neutron spectrum.

Aside from these effects at the bundle level, the increase in total core power will have an impact on the core bypass conditions. During normal operation a fraction of the fission power is released in the form of radiation, which is directly deposited in the coolant and core structures. The increase in reactor thermal power will result in an increased heat load to the core bypass region which may result in either lower bypass subcooling, or potentially the formation of significant void in the core bypass. The formation of voids in the bypass (including the inter-assembly area and water cross for SVEA fuel designs) has the effect of hardening the neutron spectrum further.

At EPU/MELLLA+ the core flow is reduced and higher void fractions are expected in the core near the MELLLA+ corner at 100 percent licensed thermal power (LTP). The NRC staff expects that the flow reduction may result in significant bundle and bypass void formation as well as a substantial radial redistribution of the coolant flow within the bundle.

Westinghouse has provided substantial qualification of the neutronic methods for application to hard spectrum exposure core designs. Specifically, Westinghouse has provided in response to the NRC staff's RAIs, the results of significant gamma scan campaigns at **[ ]** and **[ ]** The NRC staff has summarized the RAI responses in Appendix A of this SE. The NRC staff requested additional information in regards to the trend in bundle and pin power uncertainty based on plants operating with:

- Long cycle (high energy) core designs
- Extended Power Uprate (high power density) core designs
- Transition core designs

Additional descriptive details of the NRC staff's detailed review are documented in Appendix A of this SE. The NRC staff reviewed the performance of the Westinghouse nuclear design methods (PHOENIX4/POLCA7) to analyze relevant gamma scan campaigns. The gamma scan campaigns were conducted at plants that operated under challenging conditions in terms of spectral index and operating strategy. The results of the comparisons to the gamma scan campaign data do not indicate any trends or biases in predictive capability. Therefore, the NRC staff concludes that the neutronic model has been adequately qualified to support its application to currently operating BWRs with expanded operating domains, including EPU/MELLLA+, without application of a conservative penalty.

3.1.2.2 Thermal-Hydraulic Model

The POLCA-T thermal-hydraulic model is based on a two-fluid approach. The energy and mass equations are solved for both the liquid and vapor phases. The phases may be in thermal non-equilibrium. Additionally POLCA-T includes optional models for **[ ]** POLCA-T is based on a five-equation, one-dimensional thermal-hydraulic approach.

The two mass conservation equations are coupled through phase change rates (evaporation or condensation), and the energy equations are coupled through interfacial heat transfer models. The momentum equation considers pressure losses in both phases. To close the thermal-hydraulic model, several constitutive relationships and boundary conditions are required. These relationships are based on empirical correlations that model the interfacial and wall phenomena.

The NRC staff reviewed the physical basis for the two-fluid model and the applicability of the key closure relationships to determine the applicability of the POLCA-T code and to assess the efficacy of the code to predict physical phenomena using those closure relationships. Specifically, the NRC staff reviewed the applicability of the pressure drop correlations, the void quality (drift flux) assumption, the counter current flow correlation, the heat transfer correlations, and the component models that describe various components in the flow path.

For the current application, the NRC staff did not review the **[ ]** These models are not required for the modeling of BWR control rod drop accidents or for BWR time domain stability evaluations. Similarly, the NRC staff did not review the application of POLCA-T to post dryout conditions. These phenomena are relevant to the application of POLCA-T to anticipated transient without SCRAM (ATWS) or loss-of-coolant accident (LOCA) analysis, and additional information regarding the qualification of these models will be provided with the application for these specific analyses. Approval of POLCA-T for control rod drop analysis and stability evaluation does not constitute review and generic approval of the POLCA-T **[ ]** or post dryout thermal modeling.

3.1.2.2.1 Conservation Field Equations

The NRC staff reviewed the basis of the generic applicability of the basic conservation equations for BWR transient analysis. These are the mass, momentum, and energy conservation equations. The basic conservation equations are based on a five-equation representation of the fluid. The mass and energy equations are solved using one equation for each phase, while the momentum equation is solved for the fluid mixture. This formulation requires empirical relationships for closure, however, it enables modeling of non-thermal equilibrium in the fluid.

3.1.2.2.1.1 Momentum

The momentum equation is based on a single-fluid (mixture) treatment of the fluid. The theoretical basis is described by Equation 7-25 in Reference 1. The finite difference formulation of the one-dimensional momentum equation is provided in Section 7.2.1 of Reference 1. The NRC staff reviewed each term of the equation. These terms include temporal acceleration, pressure gradient, gravitation, unrecoverable pressure losses, and spatial acceleration.

The TR describes the discretization of the momentum equation for vertical flows. The NRC staff, therefore, requested additional information regarding the momentum equation in RAI 8-6 and RAI 8-7. The response to RAI 8-6 is provided in Reference 51.

RAI 8-6 includes 12 sub-parts. These sub-parts request information regarding specific terms and phenomena represented in the momentum equation. The NRC staff's detailed review of the response to RAI 8-6 is documented in Appendix A of this SE.

Generally, the NRC staff found that the information provided in the response to RAI 8-6 is sufficient to describe the application of the momentum equation to: bi-directional vertical flow, generalized geometries, and very high void fractions. The NRC staff has reviewed the information and found that the modeling approach is acceptable.

Particular terms in the momentum equation rely on correlations for particular pressure losses. These correlations were reviewed by the NRC staff and are documented separately in Section 3.1.2.2.2 of this SE.

The NRC staff requested additional information regarding the performance of the momentum equation in RAI 8-7 based on the number of questions that the NRC staff raised regarding the use of the equation for generalized reactor system geometries, including non-vertical flows, changing flow areas, plena, and elbows. The NRC staff, specifically, requested that Westinghouse test the robustness of the equation by analyzing a test case to demonstrate that the momentum equation did not generate residual momentum sources. **[**

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Based on the response to RAI 8-7S1, the NRC staff is reasonably assured that the implementation of the momentum equation is appropriate and acceptable (see Appendix A of this SE) (Reference 54).

3.1.2.2.1.2 Mass and Energy

The NRC staff reviewed the energy and mass conservation equations. The mass conservation is treated with two equations, one for each phase. The NRC staff reviewed the basis and finite difference form of the equation. The mass conservation equations account for evaporation rates that allow for coupling of the two equations. The NRC staff confirmed that the POLCA-T formulation is consistent with the theoretical basis. Therefore, the NRC staff finds that the mass conservation representation for the two phases is acceptable.

The energy conservation equation theoretical basis is provided in Equation 7-11 of Reference 1. The energy conservation equation in POLCA-T is detailed in that it explicitly tracks the phasic kinetic and potential energy terms. The NRC staff reviewed the integration and the finite difference formulation of the two-phase energy equations.

The energy equation includes only one assumption, which is to neglect the heat flux due to conduction within the fluid. The NRC staff finds that this assumption is acceptable on the basis of negligible conduction heat transfer relative to the convected fluid energy and other interfacial heat transfer mechanisms. The NRC staff confirmed that the equation accurately represents energy sources and the fluid energy components. Therefore, the NRC staff finds that the POLCA-T two-phase energy equations are acceptable.

3.1.2.2.1.3 Interfacial Heat and Mass Transfer

The POLCA-T interfacial heat and mass transfer are treated in two components. The first component addresses heat and mass transfer between the phases and the heated wall boundary, and the second component addresses heat and mass transfer within the bulk of the fluid.

The wall heat transfer is modeled using heat transfer correlations. The NRC staff review of these heat transfer correlations is discussed in Section 3.1.2.2.6. Generally, heat transfer relations are applied at the wall interface as a function of the surface temperature to determine the transient heat deposition to the phases. The heat depositions are used to calculate the evaporation or condensation rates at the wall interface. These evaporation or condensation rates form one component of the interfacial mass transfer equation.

Bulk fluid interfacial heat and mass transfer relationships are also provided to describe the phenomena occurring at the phase interfaces within the fluid. The interfacial heat transfer is calculated based on weighted averages to account for differences in the heat transfer phenomena in the transition from low void fraction flows to high void fraction flows. The NRC staff has reviewed the weighting functions and found them acceptable to account for changes in the interfacial properties over the applicable range of void fractions.

At low void fractions, the interfacial heat transfer is based on models appropriate for bubbly flow conditions, where the liquid is a continuous phase. There is a slight model correction for very low void fractions **[ ]** where the vapor temperature is forced to the saturation temperature. The NRC staff finds that such a correction is acceptable for smoothing and numerical stability in the solution.

At high void fractions, the interfacial heat transfer is based on an annular flow regime where heat transfer between the phases is driven by dispersed fluid droplets and the liquid film with a continuous vapor phase. POLCA-T includes appropriate models to determine the film thickness (hence surface area) and the interfacial area of the dispersed liquid droplets. The film and droplet heat transfer coefficients are piece-wise correlated to film and droplet size and properties, respectively. The coefficients are assigned specific values when the temperature difference (phasic temperature minus saturation temperature) becomes negative.

On the high void fraction end, a slight model correction is included to force the liquid temperature to the saturation temperature. The NRC staff finds that this model correction is similar to the low void correction and equally acceptable.

The weighting of these two regimes is acceptable for modeling the transition flow regimes between bubbly and annular (slug, churn-turbulent). The NRC staff finds that the models are appropriate and include sufficient resolution of the phasic behavior in these regimes to adequately model the phase interface in terms of heat transfer. Additional details of the NRC staff review regarding the flow regime map are provided in the NRC staff review of the response to RAI 8-3 provided in Appendix A of this SE.

Therefore, the NRC staff finds that the interfacial heat and mass transfer equations are acceptable in terms of their robustness and capability to cover the anticipated important phenomena for transient BWR applications.

3.1.2.2.2 Pressure Drop Correlations

Pressure drops are calculated for each phase in the two-fluid model. Therefore, the total pressure drop across a given node is the sum of the pressure lost by the vapor and liquid phases. The wall friction is calculated according to correlations that depend on the single-phase Reynolds number. The wall friction factors are determined according to previously approved correlations. In particular, the Colebrook correlation is used for wall friction when the surface roughness is known.

Irreversible local loss coefficients can be input by the user, or sudden expansion and contraction automated calculations can be performed. The NRC staff has previously reviewed and approved the use of the sudden expansion and contraction loss coefficient correlations.

The NRC staff has previously accepted the use of these correlations, they are consistent with industry practice, and allow flexibility to input measured loss coefficients for specific components when test or plant data is available. Therefore, the NRC staff finds that the pressure drop correlations are acceptable.

3.1.2.2.3 Critical Power

The NRC staff requested additional information regarding the dryout correlation library in RAI 8-1. In particular the NRC staff requested that Westinghouse specify the correlations in the library, the applicable fuel design, and reference to the experimental data used to develop the correlation. The NRC staff requested additional information regarding the sensitivity of the critical power ratio to burnups and power distribution uncertainties in RAI 5-4 and RAI 5-5.

The response to RAI 8-1 provides the requested information for Westinghouse fuel designs currently operated in the U.S. These include the applicable correlations for SVEA-96, SVEA-96+, and SVEA-96 Optima2 (Reference 52).

The response states that internal Westinghouse requirements assure that the use of NRC approved correlations for licensing analyses specify the correlation used, refer to the NRC approved documentation, and explains how the correlation is used within the approval.

The NRC staff finds that the response is sufficient in specifying how Westinghouse treats critical power evaluations for Westinghouse fuel designs. However, the response does not provide details regarding the mixed core application.

The subject review of POLCA-T to stability and CRDA analyses, however, does not require evaluation of the critical power ratio. The NRC staff notes that the stability Appendix does not seek approval for POLCA-T to develop delta critical power ratio versus oscillation magnitude (DIVOM) slopes. Therefore, for the subject review, the calculation of the critical power ratio is ancillary to the analyses.

The NRC staff, therefore, defers the review of the subject of the applicability of the dryout correlations to review of POLCA-T for application to either transients or to generate a DIVOM curve. In these subsequent reviews the NRC staff will address the application of POLCA-T to mixed core evaluations.

Westinghouse provided responses to RAI 5-4 and RAI 5-5 in Reference 48. The response states that the current TR submittal requests approval of POLCA-T for CRDA and stability evaluations. As these analyses are limited in their scope, the NRC staff finds that the calculation of the minimum critical power ratio (MCPR) during the transient is not required to determine the respective figures of merit for licensing analyses. Therefore, the NRC staff does not require the information requested in RAI 5-4 and RAI 5-5 to complete its subject review.

The NRC staff notes that this information would be required if approval of POLCA-T is sought to calculate the DIVOM slope for stability licensing analyses. However, this application is not covered within the current TR Appendix B.

The NRC staff interprets the responses provided to RAI 5-4 and RAI 5-5 as a commitment to provide the requested information in future submittals requesting approval of POLCA-T to either transients or to DIVOM analyses. Therefore, the NRC staff's review of POLCA-T should not be construed as acceptance of the responses to RAI 5-4 and RAI 5-5.

Similarly, as the response to RAI 8-1 addresses only Westinghouse fuel designs, the NRC staff's acceptance of this RAI response for the current applications does not constitute the NRC staff's acceptance of this RAI response regarding critical power correlations for transient or DIVOM analyses where approval for mixed core applications is likely to be sought by Westinghouse.

3.1.2.2.4 Void Quality Correlation

POLCA-T fully incorporates two drift flux models from the thermal-hydraulics code (GOBLIN). The correlations, DF01 and DF02, are used to determine the velocity slip between the phases. The slip ratio is used to relate the flow quality to the nodal void fraction.

3.1.2.2.4.1 Experimental Qualification

The NRC staff has reviewed information provided in response to RAI 2-1. The response included detailed information regarding the testing apparatus and the results of detailed comparisons of the DF02 void quality correlation to data. Measurements were performed at the FRIGG test facility **[**

] These results indicate very good agreement over a wide range of void conditions.

Tests were performed using detailed tomography with a rotating gamma source and collimated detector. The results provide for a detailed scan of each axial level, and therefore, do not require any correction for shadowing effects between heated pins. **[**

] These tests are performed concurrent with critical power tests.

Qualification data for the DF01 and DF02 correlations were also supplied in response to RAI 3-5 (see Appendix A of this SE). The NRC staff found that, overall, the qualification database for the DF02 correlation includes: modern fuel bundle geometries, test conditions up to the point of critical power, and axial power shapes that are similar to those shapes expected during operation.

3.1.2.2.4.2 High Pressure Extension and Qualification

The NRC staff requested additional information regarding the qualification of the void quality correlations to higher pressures and higher void fractions. In particular, the NRC staff requested that Westinghouse justify the application of the correlation to these higher pressures (9 MPa or higher) and voids (90 percent) that may be encountered under transient or accident conditions in RAI 8-5. In particular, events such as main steam isolation valve (MSIV) closure without position SCRAM may result in high pressures and high void fractions.

The response is based on comparisons to other void quality correlations (Reference 52). In the review of WCAP-16606-P-A, the NRC staff reviewed the application of the AA78 correlation in BISON to simulate transient thermal-hydraulic conditions at these higher void and higher pressure conditions typical of ATWS scenarios. As ATWS evaluations consider vessel pressurization without SCRAM and the subsequent recirculation pump trip, these events constitute a reasonable basis to establish the highest pressures and void fractions for which the correlations are used.

In WCAP-16606-P-A, Westinghouse describes a methodology for using the Electric Power Research Institute (EPRI) void quality correlation (Chexal-Lellouche) to characterize the anticipated trends in void fraction error (if any) at higher pressure and void fractions (Reference 17).

The response to RAI 8-5 includes detailed comparisons between the Chexal-Lellouche and the DF01 and DF02 correlations for various void fractions and pressures. The response indicates consistent trends with the AA78 correlation when taken to higher void fractions and pressures consistent with the range specified in WCAP-16606-P-A. Therefore, the NRC staff finds that the robustness of the DF01 and DF02 correlations is substantially similar to the AA78 correlation as applied over the range anticipated during transient and accident conditions. On the same basis, for the NRC staff's acceptance of the extension of the AA78 correlation in BISON, the NRC staff finds extension of the DF01 and DF02 correlations is equally acceptable.

3.1.2.2.5 Countercurrent Flow Limit Correlation

The countercurrent flow limit (CCFL) correlation is adopted from the GOBLIN code for POLCA-T. The CCFL correlation is based on the Wallis formulation with a geometry correction factor based on the Holmes formulation. The CCFL is used to calculate the mass drift flux. The methodology for performing this calculation is the same in POLCA-T as in GOBLIN (Reference 45). The CCFL correlation was previously reviewed by the NRC staff and approved for modeling of BWR emergency core cooling phenomena. The basis for the approval was a demonstrated 25 percent liquid flow drainage conservatism relative to experimental results gathered at the QUAD+ countercurrent flow test facility (Reference 45). This conservatism is observed for lower liquid fluxes while the correlation more closely matches the data for very high liquid fluxes; however, the effects become inconsequential in regions of very high downward liquid fluxes, and the correlation remains conservative in the most limiting scenario of high upward steam flux.

However, the formulation for the CCFL correlation was revised (Reference 46) to support application of the BWR Emergency Core Cooling System (ECCS) evaluation model to SVEA-96 Optima2 fuel. The formulation changed the definition of the hydraulic diameter **[** **]** The modification results in a CCFL correlation that envelops and bounds all of the measurement data, and is therefore acceptable. In RAI 8-4, the NRC staff requested that Westinghouse revise the model in POLCA-T to reflect the most recently approved model to extend the application of POLCA-T to SVEA-96 Optima2.

In its response to this RAI (Reference 51), Westinghouse provided a commitment to revise the model and provide this revision to the NRC with the Appendix D POLCA-T application to ATWS. As counter-current flow is not expected to occur for CRDA or during thermal-hydraulic density wave oscillation instability events, the NRC staff finds that the response is sufficient for the NRC staff to complete its review of the subject TR. The NRC staff will impose a condition that the CCFL correlation be revised to be consistent with the model submitted to address potential non-conservatisms for SVEA-96 Optima2 prior to POLCA-T application to transient analyses where countercurrent flow may occur.

> <u>Countercurrent Flow Limit Condition</u>[3]
> The CCFL correlation shall be revised to be consistent with the model submitted to address potential non-conservatisms for SVEA-96 Optima2 by Reference 46 prior to POLCA-T application to transient analyses where counter current flow may occur.

Furthermore, the NRC staff notes that the CCFL correlation is based on axial flow data, and therefore, cannot be applied to horizontal stratified flows.

The NRC staff notes that there are some unaccounted uncertainties in the pressure dependency of the correlating parameters, but based on the large conservatism demonstrated for prototypic BWR fuel geometries, the NRC staff finds that the CCFL correlation will

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[3] See also the NRC staff's review of the response to RAI 8-4 in Appendix A.

conservatively predict the total flow rate of liquid water into the core region and lower vessel and is therefore acceptable for evaluation of ECCS initiated transients or ATWS scenarios once corrected as stated above.

3.1.2.2.6 Heat Transfer Correlations

The NRC staff reviewed the basis for the heat transfer correlations in POLCA-T. The thermal hydraulic-model in POLCA-T is based on the RIGEL code, which the NRC staff has not previously reviewed. However, the RIGEL code shares many heat transfer correlations with the previously approved GOBLIN code. Where applicable, the NRC staff identified those regimes where the NRC staff has previously reviewed particular heat transfer coefficients. In particular cases, the NRC staff identified some differences between the POLCA-T models and the GOBLIN models (see RAI 8-2). The NRC staff has found that these differences, however, are attributable to the more detailed two-fluid representation of the vapor phase in the POLCA-T methodology.

The NRC staff identified that the correlations and their bases were consistent with current industry practices in heat transfer modeling for BWR transient analyses, however, the NRC staff could not complete its review without additional information regarding the usage of the correlations in particular heat transfer regimes. The response to RAI 8-3 provides the details of the heat transfer regime maps. These maps specify where the different heat transfer correlations are applied and how they are interpolated (Reference 51). The NRC staff has reviewed these maps and found that the correlations are applied consistent with their validation ranges and that the interpolation schemes are acceptable.

The NRC staff requested additional information in RAI 3-5, however, to demonstrate integral qualification of the heat transfer models. In response to RAI 3-5, Westinghouse provided comparisons of POLCA-T calculated cladding temperatures to measurements made during the **[ ]** The **[ ]** simulate pressurization transients in BWRs. These tests cover the heat transfer regimes typical of **[ ]** and it is reasonable to conclude that these tests cover the anticipated range of application for steady state, transient, and accident conditions. Therefore, the NRC staff finds that the response to the RAI provides adequate qualification of the heat transfer correlation usage over the full range of anticipated usage.

The NRC staff‘s detailed review of the response is documented in Appendix A of this SE. The NRC staff found that the transient cladding temperature prediction was accurate **[ ]** adding confidence in the code‘s capability to predict cladding temperature over a wide range of conditions. Therefore, the NRC staff is reasonably assured that the heat transfer correlations are acceptable and are appropriately interpolated and utilized in the analysis method.

3.1.2.3 Component Models

POLCA-T includes several models for reactor system components. These models include specific component models for pumps and separators. The NRC staff has reviewed the POLCA-T formulation for these component models. Additionally, approval is sought for the hybrid use of certain historical models, in particular, the steam line code (PARA) steam line model. The NRC staff has similarly reviewed the use of these historical component models with POLCA-T.

3.1.2.3.1 Turbo Pumps

The turbo pump model refers to those models relating the torque and shaft speed. These models are generally coupled with motor models (described in Section 3.1.2.3.5). The NRC staff reviewed the turbo pump model described in Section 18.1.1 of Reference 1. The pump model is based on inputting homologous curves. This approach has been approved by the NRC staff for similar purposes and is a common approach in the industry for calculating pump transient behavior. The NRC staff reviewed the documentation in the TR and found that the description of the model is consistent with the previously approved pump model described in Reference 45. The NRC staff likewise finds that the integral qualification data for off-normal conditions provides reasonable assurance that the model adequately captures the phenomena necessary to calculate the recirculation flow for various pump speeds and reactor powers. Therefore, the NRC staff finds that this model is acceptable.

3.1.2.3.2 Jet Pumps

The NRC staff reviewed the jet pump model described in Section 18.1.2 of Reference 1. The jet pump model is consistent with the previously approved model described in Reference 45. The original model was qualified against the 1/6th scale Idaho National Laboratory (INEL) jet pump tests. During an onsite audit at the Westinghouse Energy Center, the NRC staff confirmed that the POLCA-T code test suite included comparison of the POLCA-T model against the qualification data (Reference 47). The NRC staff finds that the model remains applicable and acceptable and is reasonably assured based on the integral qualification against full reactor models in Appendix B of the TR, as well as its audit of the POLCA-T test suite (Reference 47), that the model is appropriately incorporated in POLCA-T. Therefore, the NRC staff finds that the jet pump model is acceptable.

3.1.2.3.3 PARA Steam Line Model

POLCA-T includes a generalized nodal thermal-hydraulic solution technique and, therefore, does not include specific component models for the main steam line. However, compatibility in POLCA-T was maintained to adopt PARA steam line models. The NRC staff requested additional information regarding the use of PARA steam line models with POLCA-T in RAI 8-8.

The response states that approval of the historical model is only sought so that previously developed PARA models may be used for licensing evaluations. The response states that new PARA models of the steam line will not be used in POLCA-T licensing (Reference 51). As the historical use of the model is requested only for use with previously developed models, the NRC staff finds that the PARA conditions need not be applied to future POLCA-T licensing calculations that are performed with explicit POLCA-T modeling of the steam line. Similarly, the NRC staff finds that the PARA models were developed consistent with the approved methodology, and therefore, have intrinsically met the conditions imposed by the NRC staff on the use of PARA. The NRC staff will impose a condition that only PARA models that have been previously developed in accordance with the approved methodology and consistent with the NRC staff's conditions and limitations may be used with POLCA-T.

This condition has been captured by the general —Applicability of Conditions and Limitations on Encompassed Codes" condition (see Section 3.1.2.1).

3.1.2.3.4 Steam Separator Model

The NRC staff reviewed the steam separator model described in Section 18.2 of Reference 1. The model is based on a mechanistic representation of the carryover and carryunder phenomena based on the design of typical steam separator equipment. The NRC staff reviewed the mechanistic basis. The model is based on establishing the **[** **]** The NRC staff requested additional information regarding the performance of the model to ensure that the **[** **]** approach was acceptable for steam separators. The NRC staff requested this information in RAI 8-8. The response to RAI 8-8 included qualification of the POLCA-T steam separator model to full scale qualification data collected for the AS16 and AS01 steam separator designs (Reference 51).

The NRC staff finds that the **[** **]** The NRC staff finds that qualification indicates acceptable, reasonable performance of the model **[** **]**

For the purpose of performing CRDA analyses or stability evaluations, the transient results are not expected to be sensitive to uncertainties in the steam separator modeling of the magnitude depicted in the response to RAI 8-8. However, at EPU/MELLLA+ conditions where the core outlet quality may exceed 20 percent under transient conditions, and fouling may degrade the steam separator performance, the NRC staff is not sufficiently reasonably assured in the performance of the model to evaluate it for transient applications. Therefore, the NRC staff finds that the use of this model is acceptable for its current application. The NRC staff expects that with the submittal of the AOO application in Appendix C, that Westinghouse will address steam separator model performance for transient evaluations including any effect from EPU/MELLLA+ operation or fouling.

3.1.2.3.5 Asynchronous Drive Motor

POLCA-T includes a model for relating the pump drive shaft motor torque to the convertor frequency and voltage. The NRC staff has reviewed the basis for the model and has found that the model is a reasonable projection of motor torque based on convertor input. The NRC staff, therefore, finds that the model is acceptable.

3.1.2.3.6 Internal Recirculation Pumps (Advanced Boiling-Water Reactor)

The POLCA-T pump models are sufficiently versatile to model internal recirculation pumps similar to the ABWR design. However, the POLCA-T TR does not provide sufficient information regarding the models for recirculation internal pumps (RIPs) for the NRC staff to evaluate the performance of these models under transient or accident conditions. In particular, the NRC staff does not have sufficient information regarding the modeling of the anti-reverse rotation device or the associated motor-generator inertia. Therefore, while the NRC staff finds that POLCA-T has sufficient capabilities to model RIPs in plant system models, the NRC staff has not reviewed the transient application of these models for AOO or ATWS events. Transient RIP modeling is not required for the current application.

The NRC staff expects Westinghouse to include additional information regarding the transient RIP model in AOO application (to be submitted as supplemental Appendix C to this TR). The NRC staff's approval of POLCA-T for CRDA and stability analyses does not constitute approval of the pump model for transient RIP modeling.

3.1.2.4 Control System Model

The POLCA-T methodology includes models for the control system. The control system models are required to simulate plant response to transient parameters. The NRC staff reviewed the information provided in Reference 1 but determined that a more detailed description was necessary for the NRC staff to review the control system model.

The responses to RAI 10-1 and RAI 10-2 provide the basis for the POLCA-T control system models and describe the interactions between different controllers to simulate complex control systems (Reference 52). The NRC staff reviewed the responses as documented in Appendix A of this SE. The NRC staff found that for the predominance of reactor system control systems, the modeling approach in POLCA-T is sufficiently robust to capture the dynamic control system and plant response.

The NRC staff, however, notes that certain controllers may not be adequately modeled by the POLCA-T models. For example, **[**

] Therefore, it may be the case for particularly complex control systems that the POLCA-T stand-alone control system models are inadequate to fully model all plant control systems. For these scenarios, POLCA-T has implemented the SAFIR control system simulation code currently submitted as a separate TR for the NRC's review and approval. Therefore, upon approval of the SAFIR control system TR, the NRC staff finds that retaining compatibility with SAFIR allows licensees referencing the subject TR to adequately model necessary control systems based on the plant-specific analysis.

On the basis of the more detailed control system descriptions in the responses to RAI 10-1 and RAI 10-2, as well as upon completion of the NRC staff's review of the SAFIR TR, the NRC staff is reasonably assured that POLCA-T is sufficient in its capability to model plant control systems for transient and accident licensing analysis purposes.

3.1.2.5 Heat Conduction Models

The slab and cylindrical heat conduction models are based on a time-discretized analytical solution to the transient heat conduction equation. The implicitness factor is set to 0.5, also referred to as the Crank-Nicolson method. The NRC staff reviewed the derivation of the discretized equations for either geometry and found the equations to be accurate. The topical report verifies that volumetric power generation considers fission, direct, and metal-water reaction heat sources throughout the heat structures.

The heat conduction models are solved within the POLCA-T code by balancing the heat conduction and heat transfer (including radiation) at the interface between the coolant and heat structure. The NRC staff finds this approach acceptable.

3.1.2.6 Fuel Rod Model

The fuel rod model for the POLCA-T code was evaluated under regulatory guidance for the review of fuel system designs and adherence to applicable General Design Criteria (GDC). This is provided in Section 4.2, “Fuel System Design” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP 4.2) (Reference 26). In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- the fuel system is not damaged as a result of normal operation and anticipated operational occurrences,
- fuel system damage is never so severe as to prevent control rod insertion when it is required,
- the number of fuel rod failures is not underestimated for postulated accidents, and
- coolability is always maintained.

The NRC staff's review of the POLCA-T fuel rod model was done with respect to the above guidance to ensure that the fuel model parameters which are used as inputs for POLCA-T, or other applicable POLCA-T related computer codes, are compatible with the applicable regulatory requirements identified in SRP Section 4.2.

The POLCA-T fuel rod model is based on the STAV7.2 thermal-mechanical methodology. STAV7.2 and its predecessor, STAV6.2 have been reviewed and approved by the NRC staff separately (References 22 and 4, respectively). Therefore, the NRC staff focused its review on the accurate translation of the approved methods to the transient application and on any identified differences between the previously approved methods and the POLCA-T models described in the subject TR.

The NRC staff identified 15 RAIs regarding the fuel rod model in POLCA-T. The NRC staff‘s review of the responses to these RAIs is documented in Appendix A of this SE under RAI 11-1 through RAI 11-15. The NRC staff found that the RAI responses were acceptable to demonstrate consistency with the previously approved models and to justify the applicability of the POLCA-T fuel rod models to transient analyses.

The NRC staff requested information regarding the fuel thermal and mechanical modeling methodology in RAIs 11-1, 11-2, 11-3, 11-4, 11-5, 11-6, 11-7, 11-8, 11-9, 11-10, 11-11, 11-14 and 11-15. Based on the review of the RAI responses, the NRC staff concludes that the approved STAV models have been acceptably translated to the POLCA-T method consistent with their previous NRC approval.

The NRC staff found that the fuel thermal and mechanical models (as used for domestic licensing evaluations) are consistent with the STAV7.2 models. Other models are included as optional or default alternatives to the STAV7.2 based model. Consistent with the response to RAI 11-15, the NRC staff requires that licensing evaluations be performed using the STAV7.2-based models (Reference 67).

> <u>Use of STAV7.2-based Models</u>[4]
> To be consistent with WCAP-15836-P-A, the STAV7.2 fuel thermal conductivity model and pellet relocation model provided in TR Sections 14.2.1 and 14.2.2.3, respectively, will be used in POLCA-T when performing licensing calculations.

The NRC staff‘s approval of these models, however, was subject to five conditions reported in the NRC staff‘s safety evaluation. The STAV7.2 TR and SE are provided in Reference 22.

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[4] See also the NRC staff‘s review of the response to RAI 11-15 in Appendix A.

These conditions specify the scope of the NRC staff's approval of the methods, including the range of specific fuel parameters where the models are approved. These limitations are equally applicable to the models presented in the POLCA-T TR with one exception. The response to RAI 11-15 provides a detailed justification [ ] The NRC staff reviewed the detailed justification and finds it acceptable, as documented in greater detail in Appendix A of this SE. Therefore, [ ] as stated in the responses to RAI 11-15, the NRC staff imposes these same conditions on all dynamic applications.

This condition has been captured by the general —Applicability of Conditions and Limitations on Encompassed Codes" condition (see Section 3.1.2.1).

The NRC staff has imposed historical conditions and limitations from both POLCA7 and STAV7.2. To clarify the NRC staff's approval in terms of gadolinia concentration, the NRC staff provides the following condition.

> Gadolinia Concentration Limitation[5]
> POLCA-T is only applicable to the analysis of cores loaded with gadolinia bearing fuel within the minimum-approved-maximum-gadolinia-concentration of either STAV7.2 or PHOENIX4/POLCA7 as documented in WCAP-15836-P-A and CENPD-390-P-A, respectively, or in subsequently approved submittals.

The NRC staff had requested information regarding the cladding rupture model in RAI 11-13. The cladding rupture model is retained from the BWR ECCS Evaluation Model described in Reference 5. The NRC staff requested information to ensure that the model in POLCA-T is consistent with the NRC staff's approval of the model for use in ECCS calculations. The NRC staff notes that the ECCS Evaluation Model was revised to account for cladding rupture due to rod-to-rod contact in Reference 21. In the response to RAI 11-13S1, Westinghouse provided confirmation that the cladding rupture model is consistent with the previously approved model. The detailed NRC staff's review of the RAI response is provided in Appendix A of this SE. The RAI response also provides corrections to the model description in the TR to bring the POLCA-T documentation of the cladding rupture model into alignment with the previously approved cladding rupture model (Reference 66). On the basis of the consistency of the POLCA-T model with the previously approved model, the NRC staff finds that it is acceptable.

In the particular case of the exothermic metal-water reaction model, the NRC staff found that additional information was required for the NRC staff to complete its review. Additional information regarding this model was requested in RAI 9-1 and RAI 11-12. However, the NRC staff notes that for CRDA analyses and stability evaluations, significant fuel cladding heat-up does not occur. The NRC staff notes that significant cladding heat-up is not expected during simulations of oscillations indicating core stability, or in the analysis of transients indicating margin to boiling transition. For the analysis of the control rod drop accident the gross core heatup is sufficiently limited (analyzed at cold conditions) that any additional heat provided to coolant from the potential for metal-water oxidation is sufficiently small that the use of either the Baker-Just or Cathcart-Pawel models is acceptable. These models are widely referenced for this purpose. Therefore, the NRC staff does not require specific details of the metal-water reaction model for the subject TR review. The NRC staff defers detailed review of the

[5] See also the NRC staff's review of the response to RAI 11-4 in Appendix A.

exothermic metal-water reaction models to review of the POLCA-T methods for ATWS evaluations (Appendix D) or for other applications where significant fuel heat may occur. The NRC staff's approval of POLCA-T for CRDA and stability analyses does not constitute generic approval of the exothermic metal-water reaction models.

The NRC staff expects that the fuel thermal mechanical models will be revised as new data becomes available and as Westinghouse introduces advanced fuel designs into the operating reactor fleet. For example, STAV7.2 incorporates updated versions of the STAV6.2 fuel thermal-mechanical models based on more recent test data. Additionally, the NRC staff is aware of planned fuel design and analysis method innovations, such as advanced doped pellet technology (ADOPT) additive fuel and AXIOM™ cladding, and STAV code upgrades (References 63 and 64). ADOPT and AXIOM™ will require specific NRC review before existing or updated thermal-mechanical methods are applied (Reference 22). On this basis, the NRC staff expects that the fuel rod models listed in Section 14 of the TR will become obsolete. Therefore, the NRC staff imposes the condition that these models be updated with the advent of new fuel thermal-mechanical performance models.

It is the intention of this condition to supersede the aforementioned conditions: —Use of STAV7.2-based Models" when a new fuel thermal-mechanical methodology is reviewed and approved by the NRC staff. It is also intended to be consistent with the response to RAI 11-15 (Reference 67).

The NRC staff intends to review the applicability of updated fuel thermal-mechanical models to the transient and accident analyses performed using the POLCA-T methodology during its review of subsequent, updated stand-alone fuel thermal-mechanical performance codes. The NRC staff approval is likely to be based on a consideration of the range of applicability of any revised models. Therefore, the NRC staff imposes the condition that the application of POLCA-T with revised fuel thermal-mechanical models be consistent with the NRC staff's approval of those revised models.

It is the intention of this condition to supersede the specific codes listed in the aforementioned condition: —Applicability of Conditions and Limitations on Encompassed Codes" when a new fuel thermal-mechanical methodology is reviewed and approved by the NRC staff.

The NRC staff has considered the specific case of updates to the thermal-mechanical code, but imposes a more general condition to those previously approved codes that comprise the POLCA-T methodology.

> <u>Encompassed Code Updates Condition</u>
> If a new NRC approved code takes the place of an existing POLCA-T code listed in —Applicability of Conditions and Limitations on Encompassed Codes" (see Section 3.1.2.1), licensees will need to verify that the downstream effects on POLCA-T result in conservative or essentially the same calculational results. Essentially the same results are within the margin of error for the type of analysis being performed. The implementation of the new code will also be in compliance with the —Applicability of Conditions and Limitations on Encompassed Codes" condition.

The NRC staff considers POLCA-T model updates of this kind to constitute a change from a method (or element of a method) in the safety analysis to a different method (or element of a method) that has been approved by the NRC for the intended application. Therefore, this specific type of model revision may be made without specific NRC review and approval of the specific change(s) made within the POLCA-T methodology. However, the NRC staff requires that any such changes be recorded in an auditable manner to meet the QA requirements of 10 CFR 50 Appendix B.

3.1.2.7 Calculation of Transient Power

The NRC staff requested additional information regarding the POLCA-T calculation of the reactor power in RAI 9-1. The response states that following a reactor SCRAM, the power generation includes sources from transient fission power (during the rod insertion and from delayed neutrons), fission product decay, actinide decay, decay of structural activation products, heat transfer from vessel internals, and exothermic energy release from metal-water reactions (Reference 51).

The response states that the POLCA7 neutronic code is used to calculate the fission power. The fission power is divided into two parts, that part deposited directly in the coolant (direct moderator heat) and the heat deposited in the fuel rods. The code calculates the prompt fission and delayed fission. During its audit of POLCA-T, the NRC staff reviewed the documentation and found that the POLCA-T neutron kinetics solver is based on a **[** **]** (Reference 47). **[** **]** are widely used and have previously been approved by the NRC staff in similar applications. The NRC staff finds this model to also be acceptable.

The decay power is calculated from the American Nuclear Society (ANS) Standard 5.1 or by user-supplied data. The default option in POLCA-T is the 1979 ANS Standard. The ANS standard is widely used for this application, and the NRC staff finds that its use is acceptable.

The stored energy is calculated according to the solution to the heat transfer and conduction equations for each heat structure included in the core model. Each thermal mass is assigned a heat structure to determine its transient variation in temperature and stored energy. The NRC staff finds, therefore, that the POLCA-T solution technique explicitly accounts for the stored energy.

The NRC staff notes that significant cladding heat up is not expected during simulations of oscillations indicating core stability, or in the analysis of transients indicating margin to boiling transition. For the analysis of the control rod drop accident, the gross core heat-up is sufficiently limited (analyzed at cold conditions) that any additional heat provided to the coolant from the potential for metal-water oxidation is sufficiently small, such that the use of either the Baker-Just or Cathcart-Pawel models is acceptable. These models are widely referenced for this purpose. Therefore, the NRC staff does not require specific details of the metal-water reaction model for the subject TR review.

For evaluation of ATWS, the NRC staff will require more specific details of the standard production technique and the selection of the appropriate correlation for the safety analysis.

3.1.2.8 Solution Technique

The basic solution technique for POLCA-T is based on a **[** **]** The state variable vector concept tracks the thermal-hydraulic nodal parameters (pressure, void fraction, liquid temperature, gas temperature, partial pressure of non-condensable gas, boron concentration, average liquid velocity, and average vapor velocity). These parameters can be used with the coupled hydrodynamic field equations and neutronic models to determine rates of change in the vector quantities for a series of volume cells. The flow paths (or volume cell interfaces) are described by the cross vapor and liquid velocities.

The NRC staff has reviewed the volume cell state variable vector contents and the flow path primary variables and, based on the field equations and coupled models, has determined that these provide a sufficient basis for nodal description to allow for iterative solution of the transient neutronic and hydrodynamic state.

The steady state and transient calculations are performed in a similar manner. The steady state solution technique is a special case of the transient solution where the time derivative terms for the state variables are zero. The POLCA-T code linearizes the time domain response for each state variable. The field equations, closure relationships, and associated models provide the basis for determining the rates of change in variable quantities. POLCA-T includes options for both implicit and semi-implicit time integration methods for calculating the transient response.

Based on the particular problem, a particular time integration technique may be preferred for the desired computational accuracy. For example, a stability evaluation may require the use of the semi-implicit time integration technique to avert numerical damping of the transient results. While the default option in POLCA-T is to use first order implicit time integration, there is an option to use a more accurate second degree implicit time integration technique. The TR states that the second order implicit time integration technique is generally used in practice when performing licensing analyses.

Furthermore, time step controls are necessary to ensure validity of the linearization technique. POLCA-T includes specific limits for time step control as described in Section 17 of Reference 1. Time step controls are established based on Courant limit checks, material properties out of the desired range, and on state variable derivatives. Fast disturbances result in reduced time steps to allow for accurate following of the transient behavior in POLCA-T.

When an automated time step control option is used, the time step decreases if the accuracy within successive iterations is outside a user defined allowable value. Furthermore, accuracy is ensured by controlling time step and number of iterations through convergence criteria. The NRC staff has reviewed the default values for POLCA-T. A value of **[ ]** typically is selected. This is consistent with most state-of-the-art BWR transient codes. The NRC staff finds this acceptable. There is a separate default convergence criterion on POLCA-T based on a rod surface temperature difference of **[ ]** The NRC staff finds that this value is sufficiently small that time step and iteration controls will ensure that rapid transients affecting the fuel rod thermal characteristics will be acceptably tracked in accordance with the accuracy of the physical models.

Therefore, the NRC staff finds that the solution technique is based on a sufficient set of key state variables to track the hydrodynamic and neutronic behaviors during transient analyses.

There are sufficient time integration options and controls on the iterative nature of the solution to ensure that the calculations are performed within the accuracy established by the limits and qualification of the physical models and field equations. Therefore, the time integration options and controls are acceptable.

3.1.3 Accident Scenario Identification Process

The current application for POLCA-T considers application of the POLCA-T method to control rod drop accident analysis (CRDA) and core stability evaluation. However, Westinghouse has requested generic review of the POLCA-T method to perform several transient analyses and will supplement the POLCA-T TR with appendices for each application. Westinghouse has identified the following five types of analyses for potential application of POLCA-T:

- Anticipated Operational Transients
- Stability Evaluation
- Reactivity Initiated Accidents (specifically CRDA)
- ATWS
- Loss-of-Coolant Accident (LOCA) Without Core Uncover

3.1.4 Code Assessment and Uncertainty Determination

Code assessment for POLCA-T is provided on an application specific basis. The code assessments provided in Reference 1 are for CRDA and stability evaluation. While qualification studies performed for these applications may be referenced in future submittals, the NRC staff has limited its review to those models exercised by the cases in the qualification studies provided in Reference 1.

The application specific assessment cases provide the basis for the uncertainty in calculational results specific to each application. Therefore, any acceptance criteria for licensing evaluations are based on the application specific assessment. The NRC staff‘s reviews of the assessment of POLCA-T for CRDA analysis and stability evaluation are described in Section 3.2 and Section 3.3 respectively.

3.1.5 Quality Assurance Plan

SRP 15.0.2 states that the code must be maintained under a quality assurance program that meets the requirements of 10 CFR 50 Appendix B. POLCA-T will be implemented in accordance with Westinghouse‘s Quality Management System (QMS) program, which has been reviewed and approved by the NRC staff. Westinghouse‘s Quality Management System includes computer software related requirements pertaining to software development, change control and testing, and code maintenance revisions or updates. The NRC staff audited the QMS implementation for POLCA-T and documented its findings in an audit report (Reference 47). The NRC staff found that the implementation met the requirements of 10 CFR 50 Appendix B.

As documented in the NRC staff‘s audit report, several revisions were made to the POLCA-T test suite for code maintenance and revision. These updates, specifically, include the addition of stability and complex transient test cases to test the applicability of code changes in the neutronic solution for the scope of POLCA-T‘s application (Reference 47). The NRC staff imposes a condition on the QA program as audited by the NRC staff and requires that the modified test suite be incorporated in the approved program.

> <u>Quality Assurance for POLCA-T</u>
> Future release candidates of the POLCA-T code must be tested using a software test matrix that includes the revisions audited by the NRC staff as documented in Section 3.3 of Reference 47.

3.1.6 Applicability to Boiling-Water Reactor Designs

The subject TR requests approval of POLCA-T for BWR analyses. However, the NRC staff does not find that the application is complete for categorical application to all BWR designs. The NRC staff, however, has found that the qualification database for the constitutive models and codes and the present qualification support the application of POLCA-T to BWR/2-6 and the ABWR.

3.1.6.1 Boiling-Water Reactor/3-6 Plant Designs

The NRC staff considered the application of the POLCA-T code to operating reactors with expanded operating domains. The supplemental qualification data provided in response to the NRC staff's RAIs is sufficient to demonstrate continued applicability of the code uncertainties to BWR/3-6 expanded operating domains, including increased core flow (ICF), extended load line limit analysis (ELLLA), maximum ELLLA (MELLLA), stretch power uprate (SPU), maximum extended operating domain (MEOD), extended power uprate with MELLLA (EPU/MELLLA), and EPU with maximum extended load line limit analysis-plus (EPU/MELLLA+).

3.1.6.2 Boiling-Water Reactor/2 Plant Designs

The NRC staff reviewed POLCA-T model methods specifically used in the analysis of BWR/2 plants. The BWR/2 plant designs incorporate isolation condensers. In BWR/2 plants, the Isolation Condenser System (ICS) is a standby, high pressure system for removal of fission product decay heat when the reactor vessel is isolated from the Main Condenser. The system prevents overheating of the reactor fuel, controls the reactor pressure rise, and limits the loss of reactor coolant through the relief valves.

Analyses accounting for ICS performance require models to account for condensation heat transfer in the tube bundles of the ICS. The NRC staff reviewed the basis for the condensation heat transfer correlation and found that the model basis (horizontal tube data) is applicable to the current fleet ICS designs (namely **[ ]** As noted in Section 3.1.6.4 simplified BWR designs incorporate significant design differences in the ICS relative to the BWR/2 design. Therefore, the NRC staff finds that while the model is applicable to the current operating fleet, additional qualification or justification would be required in order for the NRC staff to approve POLCA-T to model events requiring analysis of simplified BWR ICS performance.

The supplemental qualification data provided in response to the NRC staff's RAIs is sufficient to demonstrate continued applicability of the code uncertainties to hypothetical BWR/2 expanded operating domains, including ICF, ELLLA, MELLLA, SPU, MEOD, EPU/MELLLA, and EPU/MELLLA+.

3.1.6.3 Advanced Boiling-Water Reactor

The NRC staff conducted a review of the applicability of the POLCA-T methods to model features of the ABWR. The NRC staff conducted its review consistent with the review of the GE methods applicability to the ABWR documented in Chapter 15 of the NRC staff's final safety evaluation report (FSER) for the ABWR design certification (Reference 35). In its previous reviews, the NRC staff focused on the modeling features in the ODYNA and REDYA codes to model the recirculation system as well as modeling of subcooled liquid flow in the upper plenum. The NRC staff's review was guided by those features unique to the ABWR design.

Unique features of the ABWR design include the internal recirculation pumps, fine-motion control rod drives (FMCRDs), microprocessor-based digital control and logic systems, the core flooder design, and digital safety systems. Of these features, those impacting the approval of a transient analysis methodology for application are the internal recirculation pumps (RIPs), FMCRDs, and the high and low pressure core flooders (HPCF and LPCF, respectively).

The NRC staff has found that Westinghouse has extensively qualified the use of their methods to model internal recirculation pump designs. Many of the ASEA-Brown-Boveri (ABB) designed BWRs include internal recirculation pumps. The POLCA-T TR includes the **[** **]** plant in the stability modeling qualification. **[** **]** is an ABB designed BWR 75 with eight RIPs. Therefore, the NRC staff has found that the POLCA-T code is acceptable for modeling BWRs with internal recirculation pumps. However, as noted in Section 3.1.2.3.6, the POLCA-T TR does not provide sufficient details of the transient RIP model for the NRC staff to review the applicability of this model for AOO or ATWS evaluations of the ABWR. The NRC staff expects this to be addressed in the AOO application (to be submitted as supplemental Appendix C to this TR).

The POLCA7 code includes model features that allow accurate nodal calculations for partially controlled nodes, as would be present with a plant with FMCRDs. The NRC staff has previously reviewed and approved the POLCA7 axial homogenization model, which uses a one-dimensional diffusion solver to determine axial discontinuity factors to calculate nodal parameters with varying axial geometry (such as control blade insertion).

Lastly, the NRC staff considered the ABWR ECCS design. The ECCS includes the HPCF and LPCF systems that inject coolant above the core. In terms of modeling capabilities, the five-equation model will allow POLCA-T to simulate non-equilibrium between the vapor in the upper plenum and the injection. Secondly, POLCA-T will include a qualified and approved CCFL correlation to model the mass flux of coolant into the reactor from above the top of active fuel (TAF) if updated as described in the response to RAI 8-4 (Reference 51). The formulation of the thermal-hydraulic model is sufficiently flexible to model the two-fluid behavior for the injection of subcooled liquid into the upper plenum volume by allowing nodal conditions with concurrent subcooled liquid and superheated vapor. Therefore, while such modeling capability is typically required for analysis of ECCS performance during LOCA analyses, the NRC staff finds that the POLCA-T code (once updated according to the response to RAI 8-4) is sufficiently robust in its modeling representation to model these features for the ABWR ECCS for events such as inadvertent HPCF or LPCF initiation.

3.1.6.4 Simplified Boiling Water Reactors

Application of POLCA-T to the simplified boiling-water reactor (SBWR) or economic simplified boiling-water reactor (ESBWR) will require that Westinghouse submit qualification of POLCA-T to perform calculations for unique design features, such as (but not limited to): chimneys, gravity driven core cooling, standby liquid control injection into the core bypass, and modern isolation condenser designs. Therefore, the NRC staff does not find the application of the POLCA-T method to SBWR or ESBWR acceptable. This is a restriction on the POLCA-T method.

> <u>Simplified Boiling-Water Reactor Restriction</u>
> The NRC staff's approval of POLCA-T is limited to BWR/2-6 and the ABWR plant designs.

3.1.7 Code Stewardship

The POLCA-T methodology is comprised of a series of models, a solution technique, an implementation, and its associated assessment for applicability. The NRC staff notes that this method is maintained as a computer code. The NRC staff understands that several changes and updates are made to codes for various purposes and that many of these changes do not have an impact on the code's execution of the methodology it embodies.

However, some changes to the code have the potential to change the methodology. Therefore, the NRC staff imposes conditions on the stewardship of a code within its associated quality assurance procedures to ensure that the methodology for performing safety analysis does not adversely depart from the NRC approved method.

The NRC staff notes that changes that potentially affect a method for performing an evaluation in establishing the design basis of in the safety analysis for a plant require that the change be assessed using the criteria of 10 CFR 50.59 to determine if NRC review and approval are required prior to the implementation of the change.

Code changes to certain numerical methods to improve convergence, changes to enhance input or output features, changes to facilitate compilation on alternative platforms, or changes to include auxiliary functions are examples of code changes that are unlikely to have any impact on the code's execution of the approved methodology. However, other changes may constitute a departure from the methodology. Therefore, the NRC staff imposes the following code change limitation to provide clarification of the provisions of 10 CFR 50.59.

> <u>Code Change Limitation</u>
> Any changes to the POLCA-T solution techniques (i.e., calculational framework), as described in the application-specific appendices to the subject TR, that would yield inconsistency with the NRC staff approved documentation is considered by the NRC staff to constitute a departure from an element of the methodology in the safety analysis.

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

3.2.1 Introduction

The NRC staff has previously reviewed and approved RAMONA-3B to perform BWR control rod drop licensing calculations (Reference 3). RAMONA-3B is a three-dimensional coupled thermal-hydraulic and neutronic code. In the current application, Westinghouse submitted the POLCA-T code for review and approval to replace RAMONA-3B for control rod drop analysis.

3.2.2 Applicable Regulatory Bases

GDC 28 requires that the reactivity control system be designed in such a way as to limit the consequences of a control rod dropout. GDC 28 states:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability

to cool the core. These postulated accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Westinghouse has submitted the POLCA-T control rod drop accident analysis methodology for NRC's review. The purpose of the POLCA-T CRDA method is to determine the transient reactor behavior under limiting initial conditions and assess damage to both the fuel structures, supports, and the reactor coolant system (RCS) to determine whether GDC 28 is met.

The analysis method also considers fuel failures and the ability to meet the criteria in 10 CFR 100 considering multiple local fuel failures. The scope of the subject TR, however, does not include the radiological assessment methodology.

3.2.3 Technical Evaluation Regarding Control Rod Drop Accident

3.2.3.1 Overview of Control Rod Drop Accidents

CRDA is a reactivity insertion accident whereby it is postulated that during any point in the operation of the reactor, a control blade becomes stuck in the fully inserted position and becomes decoupled from the associated drive mechanism. At a later point in time the drive is withdrawn leaving the control blade in the fully inserted position. A reactivity insertion accident occurs when the control blade is postulated to become free and drop to the position of the decoupled drive.

Analysis of the control rod drop accident must consider all possible control rod configurations and operating conditions to determine the consequences from a limiting control blade drop.

Typically, the consequences of a CRDA are greatest under cold zero power conditions. Under these conditions, the control blade incremental worth is highest, the core is loosely coupled, and the RPV inventory is predominantly liquid water and potentially sub-cooled such that moderator voiding does not contribute negative reactivity feedback. The control rod drop occurs such that the dropped rod falls to the last position of the drive mechanism at a rate determined by the design of the velocity limiter at the bottom of the blade.

When a control rod drops under cold conditions, the local power around the control rod increases rapidly and dramatically, typically on the order of a decade every 25 msec. The rapid power increase results in an increase in the fuel temperature, which results in a negative reactivity addition due to the Doppler effect. The Doppler reactivity limits the peak transient power, and the event is terminated by a 120 percent average power range monitor (APRM) SCRAM.

During the CRDA, there is the potential for the local power to increase substantially and result in the formation of voids around the fuel pins in nucleation locations. The formation of these voids, while generally not credited in CRDA analyses, provides additional negative reactivity feedback to help limit the peak and integrated local power prior to the SCRAM. The Phenomena Identification and Ranking Table (PIRT) ranks this phenomenon as medium based on the small amount of negative reactivity.

The NRC staff has previously reviewed CRDA analysis methodologies and has found that the fuel rod heat transfer is sensitive to the void production at the rod surface, the specific surface conditions (including unflooded nucleation sites) and the subcooling history. The void formation rate at the rod surface is sensitive to the surface conditions and subcooling history, and the relationship between wall void growth and bulk void formation under these rapid transient conditions is not easily modeled.

The power increase from the reactivity addition is terminated by prompt negative feedback from the Doppler effect and the heat-up of the fuel surrounding the dropped blade. The nuclear dynamic response and the thermal-hydraulic models are used to determine the energy deposition in the fuel during the power increase in the early phase of the transient and through the termination after SCRAM to compare against the fuel enthalpy limits provided in SRP 4.2 (Reference 26).

The limiting CRDA is determined on a plant-specific basis considering the particular plant hardware and technical specifications. For banked position withdrawal sequence (BPWS) plants, the rod worth minimizer (RWM) issues rod blocks to limit the incremental reactivity worth of any potential dropped rod. Additionally, the analysis must account for the minimum SCRAM times based on allowable limits in the plant Technical Specifications (TS).

In the determination of the limiting control rod consideration is given to the maximum rod worth based on achievable rod motion deviations from the BPWS allowed by the plant TS, plant hardware (including the ability to bypass rod blocks issued by the RWM), and the worst single failure or operator error.

3.2.3.2 Scope of the Review

Section 3.1 of this SE documents the NRC staff's review of the basic models and formulae that comprise the POLCA-T basic code system. The scope of the NRC staff's review in terms of their application to CRDA analysis is limited to review of those models specifically related to the important phenomena affecting CRDA analysis, the exercise of the code, and the assessment of the uncertainties in developing acceptance criteria.

3.2.3.3 Phenomenology Important to Control Rod Drop Accidents

Phenomena important to the modeling of CRDAs include:

- Acceptable transient modeling capabilities for fast prompt critical transient power prediction
- Acceptable fuel rod temperature models to capture the Doppler effects during fast transients
- Acceptable fuel rod models to determine energy deposition to the fuel during the accident
- Acceptable cold core neutronic modeling to determine control blade worth
- Sufficient spatial and temporal resolution to determine the transient power shapes and transient reactivity

The NRC staff reviewed the PIRT included in Table A.2-1 of Reference 1 and found that it captures these phenomena with appropriate importance rankings. The PIRT is consistent with the PWR rod ejection accident PIRT endorsed by the NRC in Reference 41.

3.2.3.4 Control Rod Drop Accident Methodology

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In RAI 7-11, the NRC staff requested additional clarification regarding the analysis procedure in terms of explicit control rod modeling, or if capabilities were maintained in POLCA-T to analyze effective center control rods. The response to RAI 7-11 clarifies [ ]

3.2.3.4.1 Determination of Limiting Initial Conditions and Candidate Rods

While the control rod worth is a typical indicator for limiting CRDAs, other factors affect the peak fuel enthalpy for a postulated CRDA aside from the total control rod worth. [ ]

The POLCA-T methodology accounts for both of these considerations in the plant and cycle-specific determination of the limiting control rod, unless specific dynamic analyses can be shown to conservatively bound particular control rod drop sequences.

In RAI 7-5, the NRC staff requested additional information regarding the determination of the limiting initial conditions, particularly in regards to the potential for a mid-cycle shutdown and rapid restart where core exposure may affect those parameters to which the peak fuel enthalpy is sensitive, namely the axial power shape, Doppler coefficient, and delayed neutron fraction in RAI 7-5. The response to RAI 7-5 states that [ ] (Reference 51). The procedure is described in the sample analysis provided in Section A.4.6 of the TR. The NRC staff finds that this approach is acceptable to ensure that potentially limiting control rods are identified and analyzed appropriately.

In RAI 7-13, the NRC staff requested that Westinghouse evaluate the potential for a high inlet subcooling scenario initiated from a core power level above the cold zero power condition to be more limiting considering that the Doppler coefficient decreases in magnitude with increasing fuel temperature. The response to RAI 7-13 justifies the conservatism in the selection of the cold initial conditions. The cold initial conditions are selected as:

- cold conditions with sufficient subcooling to prevent coolant saturation ensure that the cladding heat transfer coefficient remains lower than under conditions of nucleate boiling,
- cold conditions with sufficient subcooling to prevent coolant saturation ensure that the power pulse is not retarded by negative reactivity insertion from void formation, and
- at cold zero power conditions, the control blade worth is maximized due to spectral softening.

<u>Initial Conditions Condition</u>
Consistent with TR Section A.5.4 Item 1, POLCA-T CRDA cases must assume **[**
]

In RAI 7-15, the NRC staff requested that Westinghouse provide additional details regarding the determination of the limiting initial condition in regards to the precise process for determining the single worst operator failure in terms of bypassing control rods during the startup procedure. The response states that the single worst operator error and worst-case credible equipment malfunction are explicitly accounted for in the analysis. These assumptions are not fuel type dependent and therefore applicable to all fuel designs. The methodology quoted in the response relies on **[**
] (Reference 51). This approach is fully consistent with the approach approved by the NRC staff for the RAMONA-3B methodology for CRDA analysis. The NRC staff finds that the proposed TR revision and the description of the methodology in the RAI response are adequate and acceptable.

3.2.3.4.2 Fuel Rod Environment

The analysis considers the “hot” rod in terms of the predicted fuel enthalpy rise. **[**

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The NRC staff has reviewed this approach and determined that it is an acceptable methodology for determining the nodal average Doppler feedback. This is a key factor in determining that the nodal power history and the average nodal environment are adequately treated by POLCA-T. The “hot” rod calculation is also an acceptable means for evaluating the limiting rod enthalpy rise that accounts for the specific local conditions.

3.2.3.4.3 Consequence Assessment

The POLCA-T calculated fuel enthalpy is compared against appropriate criteria to determine fuel cladding failure and fuel melting and coolability limits. Section A.2.4 of Reference 1 specifies these limits. In RAI 7-26, the NRC staff requested that Westinghouse address the interim acceptance criteria for application to new reactor plants – in particular the ABWR. The response to RAI 7-26 states that the interim acceptance criteria in Reference 26 have been adopted for new reactors (Reference 66). The NRC staff finds this acceptable.

Until final criteria are published, the acceptance criteria for existing plants are consistent with the criteria presented in SRP 15.4.9.A (Reference 25). The number of fuel rods damaged is determined by comparing the calculated fuel enthalpy to a reduced threshold to account for calculational uncertainties. The NRC staff review of the uncertainty determination is described in Section 3.2.3.5.6.

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] The NRC staff has previously reviewed proposed alternate reactivity insertion accident fuel and core coolability criteria and has not endorsed these alternative limits. The NRC has not yet endorsed or published final design basis acceptance criteria for CRDA analyses. According to the TR Section A.2.4, once the NRC has finalized revised SRP design basis acceptance criteria that these will be adopted by Westinghouse (Reference 1). On these bases, the NRC staff has imposed the following condition on CRDA acceptance criteria.

> <u>Control Rod Drop Accident Acceptance Criteria Condition</u>
> Until final acceptance criteria are published by the NRC, the POLCA-T methodology will determine the extent of fuel damage using the interim acceptance criteria in Standard Review Plan Section 4.2 Revision 3 Appendix B for new reactor applications.
>
> Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt these criteria for all CRDA analyses.

In order to determine the number of failed fuel rods due to pellet cladding mechanical interaction (PCMI) using the acceptance criteria provided in SRP 4.2 Appendix B (Reference 26), the analysis must consider the hydrogen content of the cladding. The NRC staff requires that the hydrogen content be evaluated in these cases using an approved BWR correlation. The NRC staff has reviewed a hydrogen pickup model for this purpose previously in Reference 22. In its review, the NRC staff determined that the hydrogen pickup model is acceptable to determine the cladding hydrogen content at the onset of postulated transients such as BWR control rod drop.

> <u>Hydrogen Pickup Model Condition</u>
> When utilizing hydrogen content dependent PCMI cladding failure limits from Figure B-2 of SRP 4.2, the hydrogen content must be determined using the NRC approved hydrogen pickup model described in Reference 22 or a subsequently NRC approved model.
>
> Once final acceptance criteria are published by the NRC, the POLCA-T methodology for determining the hydrogen content (if applicable based on final acceptance criteria) will be determined using an NRC approved method.

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The NRC staff notes, however, for new reactor applications, such as for application of the POLCA-T CRDA methodology to the ABWR, the radiological consequences must be evaluated, and the source must include the transient fission gas release. Therefore, the NRC staff imposes a condition on the assessment of the radiological consequences for new reactor applications.

Radiological Consequences for New Reactors Condition[6]
When determining the radiological fission product inventory, the traditional source method contained in Regulatory Guide (RG) 1.195 (Reference 59) and alternative source method contained in RG 1.183 (Reference 58) must include an increased inventory to account for transient fission gas release (FGR) for new reactor applications. The transient FGR must be calculated according to the following correlation from Appendix B of SRP 4.2:

$$\text{Transient FGR} = \{(0.2286\Delta H) - 7.1419\}$$

Where:
FGR = Fission gas release, % (must be > 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt any relevant transient FGR requirements for all CRDA analyses.

3.2.3.5 Qualification Basis

The qualification basis for the CRDA application of POLCA-T is based on the previously reviewed qualification of the PHOENIX4/POLCA7 neutronic model, briefly summarized in Section 3.1.2.1, and several integral effects calculations. POLCA-T calculational results were compared against a Nuclear Energy Agency (NEA) computational benchmark problem for PWR rod ejection accidents (NEACRP PWR REA). Two integral effects tests were included in the qualification, namely the Peach Bottom Unit 2 end of Cycle 2 turbine trip test (PB2 EOC2 TT) and the special power excursion tests (SPERT) performed in the 1960s with the SPERT III E core. Finally, a computational qualification study was performed using the previously reviewed and approved RAMONA-3B code.

3.2.3.5.1 Code Benchmark Comparison

POLCA-T was used to model a PWR rod ejection accident, specifically the NEACRP-L-335 computational benchmark case. The purpose of the benchmark comparison is to demonstrate assurance that the physical models are (1) sufficient to model important physical phenomena to predict behavior that is consistent with other state-of-the-art transient codes, and (2) provide

[6] See also the NRC staff's review of the response to RAI 7-25 in Appendix A.

reasonable assurance that the model interfaces are performing as designed by showing transient predictions that are consistent with an independently established benchmark.

While the benchmark calculation is for a PWR rod ejection accident, essentially the same key physical processes drive the accident progression for BWR rod drop accidents. These phenomena include fuel heat-up, Doppler reactivity feedback, and control rod worth determination.

Comparisons between the PANTHER 4X4 reference solution, POLCA-T, and an independent NRC-approved PWR rod ejection methodology (SPNOVA/VIPRE) indicate that the relative performance between these methodologies is consistent, and transient calculational predictions are very similar. The benchmark qualification, therefore, provides reasonable assurance that the software is solving the coupled thermal-hydraulic and neutronic equations in a manner that is consistent with the methodology description.

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Therefore, the NRC staff finds that the benchmark qualification supports the conclusion that the POLCA-T methodology addresses physical phenomena in sufficient detail to model reactivity insertion accidents with the same degree of accuracy as other state-of-the-art transient codes.

3.2.3.5.2 Fast Transients

POLCA-T calculational results were compared against two experiments performed at Peach Bottom Unit 2 (PB2). While these qualification calculations are performed for a turbine trip event, the purpose of the qualification is to demonstrate that the neutronic and thermal-hydraulic model coupling is sufficient to model the transient behavior during a very rapid increase in neutron flux, and reactor power. The turbine trip transient response is driven predominantly by the rapid reactivity insertion associated with void collapse due to back pressure following the turbine trip.

While the PB2 EOC2 TT tests are included in the CRDA qualification for POLCA-T to illustrate the efficacy of the coupled solution technique, the analytical results of the study were reviewed by the NRC staff in terms of qualification of the void reactivity feedback modeling in order to support the qualification of the stability methodology.

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For the NRC staff to have confidence in the pedigree of standard production calculations performed using the POLCA-T methodology, those model options activated in the qualification must be similarly employed in the standard production process for licensing evaluations. To this end the NRC staff has imposed the following condition.

> <u>Standard Production Condition</u>
> Standard production CRDA calculations using POLCA-T must use modeling options and features that are consistent with those options and features used in the qualification calculations provided in Appendix A of the TR.

When the standard production condition is met the NRC staff has reasonable assurance that the uncertainty analysis based on the qualification calculations remains applicable to licensing calculations.

The NRC staff compared the results of the POLCA-T predicted axial power shape to that predicted using the NRC staff's independent TRACE-PARCS code. The results are provided in Figure 3.2.3.5.2.1. The TRACE and POLCA-T **[**

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The NRC staff finds that the qualification against steady state PB2 measurements provides adequate bases for acceptance of the POLCA-T **[**

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The turbine trip tests were explicitly modeled using POLCA-T. The transient response in core power was compared against measured core power**.** The comparison figures indicate that the transient response peak power is accurately modeled. Further comparison of the local power range monitor (LPRM) measurements for particular strings indicates that the transient behavior in core power following the initiation of core void collapse is adequately modeled.

Figure A.3-9 of Reference 1 provides transient LPRM calculated and measured responses.
[

] the code shows general agreement within established nodal uncertainties with the measurements and overall produces accurate predictions of the total core behavior.

The qualification analyses demonstrate acceptable coupling between the thermal-hydraulic and neutronic models to determine core reactivity, transient flux distribution, and local heat flux for transients that occur on a very short time scale (on the order of seconds); which is similar to the time scale for control rod drop accidents.

3.2.3.5.3 Super Prompt Critical Transients

Comparisons between POLCA-T predictions and experimental results at the SPERT facility were provided in the qualification. The purpose of these comparisons is to provide a basis for the POLCA-T control rod worth determination and prompt criticality transient modeling capability. The accurate modeling of these phenomena is essential in predicting the initial transient character of the power pulse. The comparisons of POLCA-T predictions to the power peak, and time of the peak power provide a basis for the qualification of the Doppler reactivity modeling.

The qualification of POLCA-T to evaluate super prompt critical reactivity transients included comparisons to experimental data collected at the SPERT facility during the 1960s. A total of 80 non-fuel damaging power excursion tests were performed with the SPERT III E core. The POLCA-T qualification has examined three of these tests, namely tests 18, 43, and 49. These tests were performed at **[ ]** The pressure was **[ ]** and the coolant temperature ranged between **[ ]**

The experimental uncertainties for the SPERT III E core were somewhat large; however, direct comparisons to POLCA-T calculations show that POLCA-T predicts transient core power that agrees with the experimental results. The time to peak power, integrated energy release, and peak power as predicted by POLCA-T are slightly and consistently conservative while still remaining within the range of experimental uncertainty.

The NRC staff notes that the POLCA-T calculations show a greater degree of agreement with the experimental results than those predicted using the approved RAMONA-3B code as shown in Reference 3. However, the NRC staff requested that Westinghouse provide a greater degree of detail to ensure that the improved overall agreement is not a result of competing effects related to the modeling techniques. The NRC staff requested additional information in RAI 7-18. The response to RAI 7-18 (Reference 54) provides the qualification against the SPERT III E test case 18 and the results of Doppler coefficient sensitivity analyses. A detailed evaluation of the response is provided in Appendix A of this SE. To summarize, the SPERT III E comparisons indicate acceptably accurate agreement with the experimental results. The sensitivity studies indicate consistent sensitivity to the Doppler coefficient between the approved RAMONA-3B method and POLCA-T, thus indicating consistency in the importance of the Doppler effect between the two codes. The consistency provides the NRC staff with assurance that the code system predicts sensitivities that are consistent with the NRC staff's expectations based on the PIRT and previous analyses.

The NRC staff compared the POLCA-T qualification analysis for the SPERT III E test case 43 against a calculation performed using the NRC staff's independent TRACE-PARCS code (Reference 62). The results of both calculations are compared to the experimental data in Figure 3.2.3.5.3.1. The results demonstrate that POLCA-T predicts the peak power and time of peak power with greater accuracy than TRACE-PARCS.

The POLCA-T qualification against the SPERT III E experiments provides reasonable assurance that the POLCA-T neutronic transient model can accurately predict changes in gross core power with reactivity insertion events exceeding one dollar in total worth. Comparisons to the SPERT III E power shapes also confirm the calculational robustness of the POLCA-T code to converge on transient core power shapes. The transient core power shape is particularly difficult to calculate for the SPERT III E core given the very high degree of neutron leakage, and therefore provides a high degree of assurance that the neutronic power distribution modeling is acceptable.

3.2.3.5.4 Code-to-Code Comparisons (RAMONA-3B)

Appendix A of the TR includes code-to-code comparisons between POLCA-T and RAMONA-3B for an ASEA-ATOM designed internal recirculation pump BWR. These comparisons were performed for an equilibrium core of SVEA-96 Optima2 and are performed on a consistent basis using the standard production CRDA analysis procedure.

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3.2.3.5.5 Discussion of the Qualification Relative to the High Importance Phenomena Identification and Ranking Table items

3.2.3.5.5.1 Calculation of the Power History

3.2.3.5.5.1.1 Control Rod Worth

The SPERT III E qualification provides insight into the transient reactivity insertion calculation and the determination of the individual control rod worth. The predictions of the initial transient response for the core power for tests 43 and 49 illustrate that the POLCA-T neutronic solver can adequately determine the reactivity insertion to predict the rate of increase in core power during the transient. However, experimental uncertainties for the SPERT III E experimental tests are too great to provide a basis for the determination of the uncertainty in control rod worth. However, core follow analyses with initial criticality provide a basis for the determination of the POLCA7 neutronic model to determine control rod worth under cold conditions. Additionally, the exercise of the POLCA-T code to model the SPERT III E tests provides a reasonable degree of assurance that consistent biases do not exist in the determination of the peak fuel enthalpy during reactivity insertion accidents, and therefore, in RAI 7-6 the NRC staff requested that Westinghouse use cold critical core follow data to establish an uncertainty in control rod worth and compare this value to the **[ ]** assumed in the uncertainty analysis.

The NRC staff reviewed the response to RAI 7-6. The response provides quantification of the control blade worth uncertainty based on various cold critical data. The response provides

[
] This value is **[** **]** analysis. Therefore, the NRC staff finds that the **[** **]** in the analysis is a conservative estimate of the 95 percent confidence limit.

3.2.3.5.5.1.2 Rate of Reactivity Insertion

The rate of reactivity insertion has been generically evaluated for GE BWR designs BWR/2-6 in CENPD-284-P-A (Reference 3). The value for the limiting drop speed is 0.948 meters per second (m/sec). The NRC staff finds that this is a generically applicable (and previously approved) drop speed for the current operating fleet of BWR/2-6 designs. The approved velocity is the maximum velocity conservatively accounting for geometry tolerances to three standard deviations (References 12, 13, and 14).

Sensitivity studies were performed in Reference 3 using the RAMONA-3B methodology that demonstrate the relative insensitivity of the peak fuel enthalpy to the dropped rod speed. The TR reports **[** **]** in the peak fuel enthalpy when the drop speed is increased from 0.95 m/sec to 1.53 m/sec. **[**
] Therefore, the NRC staff concurs that over a limited range, the consequences of the CRDA are **[** **]** On this basis, the NRC staff agrees with the PIRT ranking of medium for the drop speed for currently operating GE BWR designs (BWR/2-6). The NRC staff likewise concurs that the drop velocity of 0.948 m/sec (3.11 ft/sec) is appropriate and acceptable for BWR/2-6 designs.

However, the ABWR incorporates significant design changes in the control blade mechanical design. In particular, the ABWR control blade design does not incorporate a velocity limiter. Figures 3.2.3.5.5.1.2.1 and 3.2.3.5.5.1.2.2 illustrate the differences between the designs. Therefore, the NRC staff expects that the drop velocity for a control blade for the ABWR would exceed the 0.948 m/sec (3.11 ft/sec) velocity specified in Section A.2.5.2.7 of the subject TR. Calculations performed by Brookhaven National Laboratory (BNL) for the NRC in Reference 44 predict an increase in predicted peak fuel enthalpy on the order of 30 percent when the drop velocity is increased from 5 ft/sec to 15 ft/sec. The NRC staff submits that 15 ft/sec is much greater than the maximum velocity for a dropped rod with a limiter, but provides this reference to demonstrate that over a greater range of velocities that the analytic results are expected to become notably sensitive to the drop velocity. Therefore, the NRC staff cannot conclude that the drop speed is appropriate for use in ABWR calculations.

The NRC staff has previously analyzed the consequences of a postulated control rod drop accident as documented in Reference 35. SRP Section 15.4.9 states that a specific calculation of the radiological consequences for this accident is not necessary unless unusual plant or site features are present, or the applicant's calculation shows an unusually large amount of fuel damage (Reference 24). However, the NRC staff specifically evaluated this accident because it is the first application involving the ABWR standard design with hypothetical site boundaries. The intent of the NRC staff evaluation was to establish a reference for comparison of future applications incorporating the ABWR design. For the reference ABWR fuel design and control rod design, the NRC staff estimated that 6 fuel rods would melt and that 770 would become perforated (Reference 35). These results are likely to be sensitive to the core design, particularly the bundle lattice (8X8 or 10X10) and the worth of the control blade.

In the NRC staff's safety evaluation report for the ABWR, the NRC staff refers to guidance in Regulatory Guide (RG) 1.77 for the rod ejection accident for pressurized water reactors, in particular Appendix A part 2 (Reference 57), as appropriate guidance for developing input for the CRDA analysis. Appendix A part 2 of Reference 57 states that: ―the rate of ejection [for pressurized water reactor rod ejection accidents] should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction."

The NRC staff notes that future applicants referencing the ABWR design are expected to calculate the degree of fuel damage resulting from postulated CRDA events to compare to the NRC staff's calculation. Therefore, the NRC staff imposes an analogous condition to RG 1.77 Appendix A part 2 on the application of POLCA-T to analyze a postulated CRDA for the ABWR. The rod drop velocity shall be established based on the maximum velocity for the control blade accounting for the most conservative blade weight and geometry based on manufacturing tolerances or a conservative value.

> Advanced Boiling-Water Reactor Control Rod Drop Accident Analysis Condition 1
> The POLCA-T application to the analysis of the consequences of the ABWR CRDA requires that the rod drop velocity be determined based on the specific ABWR control rod design including sufficient conservatism to account for manufacturing uncertainties.

The ABWR control rod drop velocity will be dependent on the control rod design selected for the ABWR and should be reported to the NRC for review and approval as either part of a generic TR for a control blade design applied to the ABWR or in a plant specific ABWR fuel transition license amendment request.

Section 3.2.3.5.6.2 of this SE describes the review approach for ABWR CRDA analysis reviews based on the guidance of SRP 15.4.9 (Reference 24). The NRC staff intends to review the rod drop velocity assumed in the analysis for the first ABWR plant-specific application of POLCA-T to CRDA analyses. The NRC staff has previously endorsed the methodology in Appendix A of NEDO-10527 (Reference 14) for determining the rod drop velocity for conventional control blade designs. An analogous and parallel approach would likewise be acceptable for ABWR type control blades once a design has been established by Westinghouse.

3.2.3.5.5.1.3 Delayed Neutron Fraction

In RAI 7-8, the NRC staff requested that Westinghouse compare the value for the delayed neutron fraction using the current version of PHOENIX4/POLCA7 to those values for principle nuclides used in the previously approved CRDA transient methodology based on RAMONA-3B. The response states that the delayed neutron fraction libraries are based on various sources and that erroneous data in the ENDF/B-VI library were replaced with more reliable data (Reference 51). The NRC staff finds that the delayed neutron libraries are acceptable for CRDA analyses and notes that the uncertainty in the delayed neutron fraction is appropriately accounted for as described in the response to RAI 7-4 (Reference 52). The detailed NRC staff review of the RAI response is documented in Appendix A of this SE.

3.2.3.5.5.1.4 Fuel Temperature Feedback

In RAI 7-14, the NRC staff requested that the applicant describe any models that account for the thermal expansion of the fuel pellet and its effect on the Doppler coefficient. Additionally, the NRC staff asked if the Doppler coefficient was based on the increased resonance absorption in nuclides other than the major actinides, if so, to provide a confirmatory PHOENIX4 analysis that demonstrates that no volatile nuclides contribute significantly to the overall negative Doppler feedback. The detailed NRC staff's review of the RAI response is documented in Appendix A of this SE.

The response provides a methodology for quantifying **[** **]** expected to occur during the transient. The methodology is based on using a simplified Nordheim-Fuchs reactivity insertion model with an adiabatic thermal solution. The NRC staff finds that this simplified approach is acceptable **[**

] the NRC staff finds that POLCA-T application to entire cores of fresh fuel is limited. Therefore, the NRC staff imposes the condition that the bias be applied to all CRDA analyses.

> **[** **]** Condition[7]
> POLCA-T analysis of the CRDA requires that the final calculated fuel enthalpy be adjusted to account for the **[** **]** Equation (2) in Section A.5.3.2 of the subject TR and the response to RAI 7-14 (Reference 52) must be used for calculating the bias for all core exposures.

On a cycle-specific basis, if a licensee seeks to credit the conservatism in POLCA-T for the rare situation of a beginning of cycle (BOC) calculation for a fresh startup core, then the basis shall be submitted to the NRC for review and approval. The NRC staff makes the clarification that even if a relaxation of the condition is sought for a fresh core, that mid-cycle evaluations must incorporate the bias regardless of the core average exposure at BOC.

The response provides additional information regarding the other nuclides in the fuel and provides the NRC staff with reasonable assurance that the modeling assumptions regarding volatile nuclide migration will not have a significant impact on the predicted temperature/reactivity feedback during CRDA analysis.

3.2.3.5.5.2 Calculation of Pin Fuel Enthalpy Increase

3.2.3.5.5.2.1 Fuel Heat Capacity

The primary phenomenon driving the fuel enthalpy increase is the integral power during the CRDA. The integral power is predominantly a function of the neutron power response to the reactivity insertion and the Doppler reactivity feedback. The fuel heat capacity is a parameter affecting the CRDA analysis only insofar as it is used to predict the average fuel temperature during the transient evaluation. The average nodal temperature is used to determine the Doppler reactivity feedback.

The heat capacity will affect the final fuel enthalpy, however, given the relatively fast nature of the transient, the calculation of the transient heat conduction through the pellet, gas gap, and cladding will have a second order effect on the peak enthalpy increase.

Therefore, the NRC staff considered the uncertainty in the heat capacity in the overall uncertainty analysis. The NRC staff reviewed the heat capacity model in STAV7.2 (incorporated by reference in POLCA-T through References 22 and 4). The standard error in the thermal conductivity was evaluated by the NRC staff and their contractors and was found to

[7] See also the NRC staff's review of the response to RAI 7-14 in Appendix A.

be acceptable for thermal-mechanical design methods. The reported error in the fuel heat capacity is 3 J/kg-K. For the temperature range of interest for CRDA analysis, this is approximately 1 percent of the heat capacity.

An error of this magnitude was compared to the error assumed for the Doppler reactivity coefficient reported in Section A.5.3.2. The NRC staff finds that the primary sensitivity of the enthalpy to the heat capacity is through the evaluation of the nodal Doppler reactivity. An error of 1 percent in the heat capacity will translate to roughly a 1 percent error in the predicted nodal average temperature change. Based on the dependency of the Doppler worth ($T^{-1/2}$) on fuel temperature, the expected impact on the nodal Doppler reactivity will be approximately one half of the error in the temperature (see Appendix B: Doppler Reactivity Uncertainty Analysis of this SE). An error of 0.5 percent is negligible compared to the 5 percent assumed in the macroscopic cross section and will not impact the numerical results of the uncertainty analysis. Therefore, the NRC staff concludes that explicit consideration of the uncertainty in the heat capacity is not required for the CRDA uncertainty analysis.

3.2.3.5.5.2.2 Gas Gap Conductance

In RAI 7-12, the NRC staff requested that Westinghouse evaluate the conservatism of the gas gap conductance model. Specifically, the NRC staff notes that decreased gas gap conductance results in higher fuel temperatures and increased negative Doppler feedback, and may result in a compensating effect in terms of integrated energy deposition during the transient depending on the specific sensitivity to the Doppler coefficient. Additionally, the NRC staff requested that Westinghouse evaluate the adequacy and conservatism of the model considering that the hot pin in a bundle may have different gas gap conductance behavior than the average pin in that bundle or axial node.

Westinghouse provided a response to RAI 7-12 in Reference 54. The detailed NRC staff evaluation of the response is documented in Appendix A of this SE. The NRC staff found that STAV7.2 predicts a conservative **[ ]** gas gap heat transfer coefficient. When considered with historical sensitivity studies performed with RAMONA-3B as documented in Reference 3, the NRC staff has reasonable assurance that the **[ ]** is conservative for the evaluation of the consequences of a CRDA event. Therefore, the NRC staff agrees with the Westinghouse treatment of the gas gap conductance for CRDA analysis insofar as the gas gap conductance assumed in the analysis is conservative.

3.2.3.5.5.2.3 Pin Peaking Factors

Strong spatial neutron flux peaking is known to occur for loosely coupled cores during rapid control rod motion. Such peaking is expected across the lattice during CRDAs at low power or cold conditions. **[ ]** the NRC staff requested that Westinghouse provide qualification of the PHOENIX4/POLCA7 pin power reconstruction model under controlled cold conditions in RAI 7-10. The response to RAI 7-10 contains comparative analyses between the POLCA7 and PHOENIX4 predicted local power distributions (Reference 54). The detailed NRC staff review of the comparative analyses is documented in Appendix A of this SE. The NRC staff reviewed the comparisons and found that **[ ]** Therefore, the NRC staff finds that the rod power uncertainties documented in CENPD-390-P-A (Reference 9) are adequately justified for the radial power shapes encountered during CRDAs.

In RAI 7-9, the NRC staff requested that Westinghouse discuss the iterative solution technique and describe any controls on the time step to ensure that the transient pin power distribution is calculated in successive thermal-hydraulic nuclear iterative loops to adequately characterize the total integrated energy deposition. In response to RAI 7-9, Westinghouse provided sensitivity analyses with various maximum time steps to demonstrate that the transient power solution was adequately converged (Reference 52). The detailed review of the response is documented in Appendix A of this SE. Based on the original RAI response, the NRC staff could not conclude with reasonable assurance that the sensitivity analyses demonstrated that the transient power solution was saturated. Therefore, the NRC staff issued a supplemental request for additional information (RAI 7-9S1). The NRC staff requested that additional cases be considered with smaller time steps.

The NRC staff reviewed the response to RAI 7-9S1 provided in Reference 54. The detailed NRC staff review of the information is documented in Appendix A of this SE. The conclusion of the NRC staff evaluation is that the **[**

] Therefore, the NRC staff finds that the time step size is acceptable to calculate the power distribution evolution during CRDA analysis.

3.2.3.5.6 Uncertainty Assessment and Acceptance Criteria

3.2.3.5.6.1 Boiling-Water Reactor/2-6

An uncertainty analysis is used to account for the effects of input and calculated parameters on the peak fuel enthalpy for those items identified in the PIRT for which bounding values have not been used in the analysis. These uncertainties are determined and addressed in a manner that is consistent with the approved RAMONA-3B method (Reference 3). These uncertainties include the total control rod worth, power peaking factors, coolant density, the nodal and local peaking factors, the gap heat transfer coefficient, the Doppler effect, and the delayed neutron fraction.

[**]** In RAI 7-6, the NRC staff requested that Westinghouse evaluate the POLCA7 cold eigenvalue qualification database to determine the POLCA7 cold eigenvalue uncertainty and compare this to the assumed control rod worth uncertainty. The response to RAI 7-6 provides qualification of the nuclear methods against local cold critical eigenvalue measurements (Reference 51). The NRC staff reviewed the response and found that the qualification data are sufficient **[**

] The NRC staff finds that this assumption will adequately bound any uncertainty in the SCRAM worth for the limiting CRDA scenario (during reactor startup). Should Westinghouse seek a relaxation of this conservatism in the future, the changes in the methodology and uncertainty analysis will require the NRC‘s review and approval.

Conservative SCRAM Reactivity Insertion Limitation
A relaxation of the conservative SCRAM worth assumption described in TR Section A.5.4 Item 4 in the CRDA analysis is considered by the NRC staff to constitute a change in an element of the methodology in the safety analysis. Relaxation of the SCRAM worth assumption will generate analysis results that are non-conservative relative to the approved method.

The NRC staff reviewed qualification data of the POLCA7 based nuclear methods against relevant gamma scan data, and reviewed qualification of the pin power reconstruction methods against detailed lattice transport calculational methods as documented in Appendix A of this SE. On the basis of the detailed qualification, the NRC staff concludes that it is appropriate to use the power distribution uncertainties from CENPD-390-P-A (Reference 9) in the POLCA-T CRDA uncertainty analysis.

There are **[**

] Therefore, the NRC staff finds that this treatment of the coolant density uncertainty is conservative and bounding.

The **[**

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The Doppler coefficient uncertainty is assumed to be the same in POLCA-T as established for RAMONA-3B. While both methods rely on PHOENIX4/POLCA7 methods, the NRC staff requested in RAI 7-1 that Westinghouse demonstrate the computational efficacy of the nuclear design code suite to predict Doppler worth for modern fuel designs, operating strategies, and fuel burnup. The response to RAI 7-1 provides several comparative analyses using higher order and Monte Carlo methods. The response also provides comparisons of PHOENIX to international benchmarks (Reference 51). The detailed review of the RAI response is documented in Appendix A of this SE. The response provides the NRC staff with reasonable assurance that the predictive capability of PHOENIX is retained at a similar degree of accuracy for modern BWR fuel designs and operating strategies, and therefore, the use of the historical uncertainty in the analysis is reasonable and acceptable.

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A simple point kinetics model of a reactivity insertion accident predicts that the energy deposition in the fuel (which is related to the fuel enthalpy under essentially adiabatic conditions) is inversely proportional to the fuel Doppler coefficient and heat capacity and proportional to the difference between the control rod reactivity and the delayed neutron fraction. Previous studies have shown the fuel heat deposition, and consequently the fuel enthalpy to be sensitive to the delayed neutron fraction (References 42 and 43).

The NRC staff requested additional information regarding the sensitivity of POLCA-T to the delayed neutron fraction in RAI 7-4. The detailed NRC staff review of the response is provided in Appendix A of this SE. [

] The NRC staff finds that the revised sensitivity and uncertainty analyses are acceptable.

Sensitivity studies were performed for the control rod drop speed, SCRAM delay, and SCRAM time. The NRC staff found that the uncertainty analysis is valid based on bounding assumptions regarding the SCRAM delay and SCRAM time. In RAI 7-7, the NRC staff requested that Westinghouse provide additional details regarding the assumptions pertaining to the negative reactivity insertion rate during a SCRAM. The response states that the SCRAM speeds are based on the TS requirements for SCRAM speed (Reference 51). The NRC staff finds that this approach is acceptable.

In RAI 7-19, the NRC staff requested that Westinghouse consider a mass flow rate sensitivity using a base case critical rod pattern at the nominal flow rate. [

]

The response to RAI 7-19 provides the results of sensitivity analyses performed for CRDAs over a wide range of core flow rates (Reference 54). The results demonstrate that the figure of merit is [ ] to the core mass flow rate, thus justifying the PIRT ranking and its exclusion from the uncertainty analysis.

The overall uncertainty is then established by convoluting these individual contributors and establishing the 95 percent confidence limit. The NRC staff finds that including these uncertainty contributions in the fuel damage threshold relative to the limits for CRDA analysis is adequate when combined with the **[ ]** to provide reasonable assurance that licensing analyses performed using the POLCA-T CRDA methodology will be acceptable in demonstrating compliance with GDC 28.

3.2.3.5.6.2 Advanced Boiling-Water Reactor

The NRC staff has reviewed the analytic methodology for assessing fuel damage resulting from a postulated CRDA using the POLCA-T method. The NRC staff has likewise reviewed the basis for the uncertainty parameters used in assessing the acceptance criteria relative to the limits specified in SRP 4.2. The NRC staff's evaluation documented in Section 3.2.3.5.6.1 regarding the uncertainty assessment is applicable to the operating fleet of BWRs and the ABWR, except for the rod drop velocity for the latter. On these bases, the NRC staff finds application of the methodology to the ABWR plant design acceptable when the analysis is performed assuming an appropriate rod drop velocity.

However, SRP 15.4.9 (Reference 24) specifically differentiates between BWR/2-6 plants and the ABWR. The SRP directs the NRC staff to review ABWR applications against the analysis performed by the NRC staff that is documented in the ABWR FSER (Reference 35). Specifically, for ABWR reviews, the reviewer is directed to compare the applicant's safety analysis report to the NRC staff's assumptions for computing rod drop accident doses and to the radiological consequences. Thus, the reviewer confirms that the applicant's design would produce similar results or note significant differences. The review also must evaluate the applicant's ability to satisfy the coolability criteria (See SRP 4.2, Reference 26).

The NRC staff requested in RAI 7-25 that Westinghouse provide the methodology to assess the dose consequences of a postulated CRDA. The response to RAI 7-25 states that if radiological consequences must be evaluated, these consequences will not be evaluated using POLCA-T. POLCA-T in conjunction with the established acceptance criteria based on SRP 4.2 and the associated uncertainties is used to determine the extent of fuel damage. The radiological consequences will be evaluated using either RG 1.183 (alternate source term, Reference 58) or RG 1.195 (traditional method, Reference 59). The NRC staff finds that this approach is acceptable so long as the transient FGR is calculated and added to the fission product inventory for the dose assessment.

Therefore, ABWR applications require that licensees or applicants referencing the POLCA-T CRDA analysis methodology determine the dose consequences using a combination of POLCA-T calculations (to determine fuel damage) and RG 1.183 or RG 1.195 to compare directly the consequences determined by the NRC staff's reference analysis in Reference 35. The analyses will be evaluated by the NRC staff on the basis of: (1) the appropriateness of the analysis assumptions relative to applicable guidance in RG 1.77, (2) demonstrated margin to the dose limits reported in 10 CFR 100.11, and (3) demonstrated compliance with the coolability criteria specified in SRP 4.2.

Advanced Boiling-Water Reactor Control Rod Drop Accident Analysis Condition 2
Application of POLCA-T to evaluate CRDA for the ABWR requires submittal of:

1. The basis of the rod drop velocity for review,

2. An evaluation of the dose, based on the NRC staff guidance in RG 1.183 or RG 1.195 and the transient FGR correlation in Appendix B of SRP 4.2 and the relevant radiological consequence analysis assumptions provided in Table 15.2 of the FSER for the ABWR, and

3. An evaluation of the coolability against the criteria in Appendix B of SRP 4.2.

On a plant-specific basis, bounding CRDA calculations may be referenced for the ABWR where similar screening methods to the operating fleet methods are employed to demonstrate compliance with the aforementioned acceptance criteria on a cycle-specific basis. The acceptability of this approach is contingent upon the NRC staff review and acceptance of the control rod drop velocity assumed in the analysis.

3.2.4 Conditions, Limitations, and Restrictions

This section of the SE provides a comprehensive listing of those conditions, limitations, and restrictions that are applicable to the use of POLCA-T for CRDA analyses.

3.2.4.1 Initial Conditions Condition (Section 3.2.3.4.1)

Consistent with TR Section A.5.4 Item 1, POLCA-T CRDA cases must assume **[**
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3.2.4.2 Control Rod Drop Accident Acceptance Criteria Condition (Section 3.2.3.4.3)

Until final acceptance criteria are published by the NRC, the POLCA-T methodology will determine the extent of fuel damage using the interim acceptance criteria in Appendix B of SRP 4.2, Revision 3, for new reactor applications.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt these criteria for all CRDA analyses.

3.2.4.3 Hydrogen Pickup Model Condition (Section 3.2.3.4.3)

When utilizing hydrogen content dependent PCMI cladding failure limits from Figure B-2 of SRP 4.2, the hydrogen content must be determined using the NRC approved hydrogen pickup model described in Reference 22 or a subsequently NRC approved model.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology for determining the hydrogen content (if applicable based on final acceptance criteria) will be determined using an NRC approved method.

3.2.4.4 Radiological Consequences for New Reactors Condition (Section 3.2.3.4.3)

When determining the radiological fission product inventory, the traditional (RG 1.195) and alternative (RG 1.183) source methods must include an increased inventory to account for transient fission gas release for new reactor applications. The transient fission gas release (FGR) must be calculated according to the following correlation from Appendix B of SRP 4.2:

Transient FGR = $\{(0.2286\Delta H) - 7.1419\}$

Where:

FGR = Fission gas release, % (must be > 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt any relevant transient FGR requirements for all CRDA analyses.

3.2.4.5 Standard Production Condition (Section 3.2.3.5.2)

Standard production CRDA calculations using POLCA-T must use modeling options and features that are consistent with those options and features used in the qualification calculations provided in Appendix A of the TR.

3.2.4.6 Advanced Boiling-Water Reactor Control Rod Drop Accident Analysis Condition 1 (Section 3.2.3.5.5.1.2)

The POLCA-T application to the analysis of the consequences of the ABWR CRDA requires that the rod drop velocity be determined based on the specific ABWR control rod design including sufficient conservatism to account for manufacturing uncertainties.

3.2.4.7 **[ ]** Condition (Section 3.2.3.5.5.1.4)

POLCA-T analysis of the CRDA requires that the final calculated fuel enthalpy be adjusted to account for the **[ ]** Equation (2) in Section A.5.3.2 of the subject TR and the response to RAI 7-14 (Reference 52) must be used for calculating the bias for <u>all</u> core exposures.

3.2.4.8 Conservative SCRAM Reactivity Insertion Limitation (Section 3.2.3.5.6.1)

A relaxation of the conservative SCRAM worth assumption described in TR Section A.5.4 Item 4 in the CRDA analysis is considered by the NRC staff to constitute a change in an element of the methodology in the safety analysis. Relaxation of the SCRAM worth assumption will generate analysis results that are non-conservative relative to the approved method.

3.2.4.9 Advanced Boiling-Water Reactor Control Rod Drop Accident Analysis Condition 2 (Section 3.2.3.5.6.2)

Application of POLCA-T to evaluate CRDA for the ABWR requires submittal of:

1. The basis of the rod drop velocity for review,

2. An evaluation of the dose, based on the NRC staff guidance in RG 1.183 or RG 1.195 and the transient FGR correlation in Appendix B of SRP 4.2 and the relevant radiological consequence analysis assumptions provided in Table 15.2 of the FSER for the ABWR, and

3. An evaluation of the coolability against the criteria in Appendix B of SRP 4.2.

3.2.5 Conclusions Regarding Control Rod Drop Accident Analysis

The NRC staff has reviewed the POLCA-T general model description, the qualification basis for the CRDA application, and the combination of uncertainties used to establish appropriate acceptance criteria for CRDA analysis. The NRC staff found that the capabilities of the POLCA-T code were appropriate to model the important phenomena dictating plant transient behavior under the conditions of CRDA.

The NRC staff also reviewed the qualification basis and found that it was sufficient to demonstrate the applicable capabilities of the POLCA-T methodology as they are exercised in the conduct of CRDA analysis.

In the course of its review, the NRC staff identified particular aspects of the CRDA analysis methodology that required special treatment for application to the ABWR. In these cases, the NRC staff identified particular conditions on the POLCA-T application to the ABWR. These conditions arise predominantly due to two aspects that are unique to the ABWR relative to the operating fleet of BWR/2-6 plant designs. The first of which is the control blade design. Since the ABWR control blade design lacks a velocity limiter, the NRC staff found that the generically applicable value used in the operating plant analyses was not sufficiently justified for use in ABWR calculations. The NRC staff imposed the condition that application of POLCA-T to the ABWR requires justification of the analytical rod drop velocity. Additionally, the acceptance criteria for new plant designs in terms of the fuel enthalpy have been updated. Therefore, the NRC staff imposed conditions that the ABWR analysis be performed using these revised acceptance criteria consistent with the updated SRP and the ABWR FSER.

The NRC staff has otherwise concluded that for application to the operating fleet of reactors that the qualification basis and uncertainty analysis is adequate to justify the POLCA-T acceptance criteria. However, certain conditions were identified as key aspects of the POLCA-T methodology for performing these calculations to ensure that the calculation remains within the accuracy demonstrated as part of its qualification. The applicable conditions, limitations, and restrictions are documented throughout the SE and are provided in a comprehensive listing in Section 3.2.4. When exercised with the conditions, limitations, and restrictions listed in Section 3.2.4 of this SE, the NRC staff finds that the POLCA-T CRDA analysis methodology is acceptable.

3.3 POLCA-T for Stability Assessment and Acceptance Criteria

3.3.1 Introduction

The NRC staff has previously reviewed and approved RAMONA-3B to perform time domain stability evaluations (Reference 7). RAMONA-3B is a three-dimensional coupled thermal hydraulic and neutronic code. In the current application, Westinghouse submitted the POLCA-T code for review and approval to replace RAMONA-3B for stability evaluations.

RAMONA-3B is a three-dimensional best-estimate time domain transient BWR code. The neutronic solver is based on a one-and-a-half group diffusion approximation with fast flux extrapolation length and thermal flux albedo boundary conditions. A distinguishing feature of the RAMONA-3B code is the integral momentum equation, which allows for accurate modeling of BWR stability phenomena with high computational performance. Additionally, RAMONA-3B has the capability of modeling the reactor vessel internals and balance of plant with explicit numerical integration techniques. This capability allows RAMONA to account for thermal-hydraulic phenomena within the entire flow loop without introducing significant numerical damping (Reference 6).

RAMONA-3B was reviewed and approved by the NRC staff in 1996 (References 6 and 7). The NRC staff concluded that RAMONA-3B could estimate the channel, core-wide, and regional oscillatory mode **[ ]** for realistic BWR operating conditions based on qualification of RAMONA-3B against out-of-pile channel instability threshold test data, **[ ]** core wide oscillation tests, and **[ ]** regional mode oscillation measurements.

The RAMONA-3B application for reload evaluations for the BWR Owners' Group long-term stability solution methods was generically approved given the qualification, description of the precise methodology, and established acceptance criteria depending on the oscillation mode. In general, regional mode oscillations for stable reactors are harder to excite and only a limited number of regional mode oscillations have been included in previous qualification studies.

The NRC staff review and approval was contingent upon several technical limitations as incorporated by reference in the NRC staff's SE. These include the following requirements when using RAMONA-3B for reload stability evaluations (References 6 and 7):

- Each thermal-hydraulic channel shall be modeled with at least 24 axial nodes
- The code model radial nodalization must be such that
    - No single region of can be associated with **[ ]** of the overall core power
    - The core model must include at least three regions for each bundle type that accounts for a significant fraction of the power generation
    - The model must include a hot-channel for each significant bundle type with the actual conditions of the hot channel
- Each of the thermal-hydraulic regions must have its own axial power shape to account for the three-dimensional effects.
- For regional mode oscillations, a full core model is recommended
- A review must be performed to confirm that the perturbation actually excites each mode of the oscillation.

In general, one limitation of the time domain methods is that this can only be used to predict the decay ratio of the dominant oscillation mode. In most stable cases, the dominant oscillation mode is core-wide as opposed to regional. In these cases, the RAMONA-3B methodology could not necessarily establish the regional mode decay ratio unless the core was already in an unstable configuration. For these cases, the NRC staff accepted the approach whereby an instability threshold was established by increasing the reactor power in the model until a regional mode oscillation could be excited.

Specifically, the NRC staff approved the out-of-phase instability threshold power calculation when the acceptance criterion is set to either (Reference 7):

- The actual threshold power for out-of-phase instabilities calculated by RAMONA-3B minus an uncertainty margin that is calculated as the power required to reduce by 0.2 the core wide decay ratio under those operating conditions, or
- The power at which the code-wide decay ratio is 1.0 (i.e., 20 percent higher than the core-wide acceptance criteria) if out-of-phase instabilities are not observed in the transient response following an appropriate out-of-phase perturbation.

These acceptance criteria were approved by the NRC staff based on the extensive benchmarking of the RAMONA-3B methodology. However, the NRC staff notes that benchmarking and qualification do not require the same control procedures for input determination. The NRC staff found that time domain stability evaluations are highly sensitive to the careful determination of appropriate input parameters, including nodalization, time step control, and physical input parameters such as loss coefficients. Therefore, while the benchmarking studies performed using RAMONA-3B indicate a predictive capability for the decay ratio with an uncertainty of roughly **[ ]** the NRC staff required **[ ]** acceptance criteria (References 6 and 7).

3.3.2 Applicable Regulatory Bases

GDC 12 requires that unstable oscillations either be prevented or detected and suppressed before fuel design limits are exceeded. GDC 12 states:

> 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.

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.

Westinghouse has submitted a methodology that will predict the likelihood for the onset of an instability event. The methodology uses the decay ratio to determine whether a reactor instability is highly unlikely, and thus may meet the requirements of GDC 10 under these conditions without particular regard to the potentiality of an instability event.

3.3.3 Technical Evaluation Regarding Stability Evaluation

3.3.3.1 Basics of Boiling-Water Reactor Instability

There are multiple mechanisms that may cause a boiling water reactor system to experience oscillations. These are:

- Oscillations due to Flow-Regime Changes (Bi-Modal)
- Control System Instabilities
- Loop Oscillations
- Density Wave Oscillations

Oscillations due to changes in flow regime (Bi-Modal) can occur during startup. As the power increases, the flow will transition into different flow regimes, i.e., slug flow to annular to churn-turbulent. Under some conditions, the flow may oscillate back and forth between flow regimes. This will occur for each channel separately such that the core will not be oscillating as a whole. This results in increased noise in the core flow. Since the flow regime is well defined for steady state full power operations, this type of instability is not expected at full-power conditions.

Control system instabilities are caused by some external controller rather than power/flow mismatch. This may be due to a control system algorithm that causes a pump or valve to oscillate under certain conditions, or spurious control blade motion. Control system instabilities are not a part of the application of POLCA-T, since they are not neutronic/thermal-hydraulic driven instabilities.

Loop oscillations are often seen in a heated channel with a riser in natural circulation. If a pocket of steam with a higher void fraction flows up the riser, this would cause an increase in buoyancy and an increase in flow in the channel. This increase in flow causes a decrease in void production in the heated channel, which would cause the buoyancy to decrease and the flow to decrease. The decrease in flow causes an increase in void production, and the oscillation cycle starts over again. This would not occur in a critical nuclear reactor, since the average core void fraction will remain nearly constant due to strong void reactivity feedback.

Density wave oscillations are the main focus in BWR stability analysis. There have been numerous instability events in BWRs world-wide, most arising from density wave oscillations. A few of the events that have occurred in U.S. operating BWRs, each of which had a 2-3 second period, are:

- LaSalle 2 on March 9, 1988 — power oscillation of 25-60 percent
- Washington Nuclear Project 2 on August 15, 1992 — power oscillation of 23-43 percent
- Nine Mile Point 2 on July 24, 2003 — power oscillation and oscillation power range monitor (OPRM) SCRAM
- Perry on December 23, 2004 — power oscillation and OPRM SCRAM

The dynamic aspects of nuclear reactors have been studied and described in general for decades, (Reference 28). The dynamics of the BWR have been studied in extensive detail due to the greater concern for neutronic-thermal-hydraulic stability resulting from operation in the presence of moderator voids (References 29, 30, 31, 32, 33, and 34). Most of the more than two dozen instability events that have occurred world-wide in BWRs are the result of special stability tests. However, some have occurred during normal operation resulting in reactor shutdown. The basic cause of this type of instability is a change in reactivity caused by void fraction fluctuations.

Three modes of density wave oscillation are considered for analysis using the POLCA-T thermal-hydraulic computer code: core-wide (or in-phase) mode, regional (out-of-phase) mode, and channel mode. In the case of the core-wide instability mode, the power and flow of the entire core oscillate in-phase. On the other hand, in the case of the regional instability, the power and flow in one half of the core oscillate out-of-phase with the power and flow in the other half of the core. In the case of the channel instability, the flow oscillates in one channel independently of the remainder of the core.

Extensive discussion has been provided in Reference 34 regarding the complex nature of boiling water reactor dynamics. In summary of that discussion, the oscillatory response of the BWR depends on the movement of density waves through the core, coupled with neutronic feedback. The density wave causes a delay in the local pressure drop due to a change in inlet flow. The sum of all local pressure drops may then result in a local drop that is out-of-phase with the inlet flow.

The neutronic feedback is a function of the neutron dynamics, the fuel dynamics, the local thermal-hydraulics, and the reactivity feedback dynamics.

Prediction of power-flow oscillation, or instability, thus necessitates the ability to accurately characterize the thermal-hydraulic dynamics, especially voiding and two-phase flow, along with an accurate characterization of the neutronic dynamics. Previous NRC staff safety evaluation reports, References 9 and 2 provide the details of reviews of the nuclear design methodology and the thermal-hydraulic basis. The current neutronic – thermal-hydraulic dynamic methodology is based on these approved methods. Reference 7 provides the details of previously approved stability methods. The present review must bring the two together and determine the adequacy of POLCA-T to predict the stability of modern BWRs.

Previous stability modeling efforts, such as the LAPUR methodology developed by the NRC adopt a frequency domain (as opposed to a time domain) solution methodology to determine the decay ratio associated with a particular plant configuration. The LAPUR method combines neutron kinetic and thermal-hydraulic models in the frequency domain to determine the open loop transfer function (OLTF). The OLTF is a feedback loop modeled to include all of the dynamic feedback mechanisms arising from density wave propagation through the reactor.

In a case where a system is truly unstable, the OLTF is able to propagate the excitation via feedback mechanisms once the initial perturbation is removed. According to the Nyquist theorem, in a frequency domain, the response of the OLTF can be used to determine if a system is unstable by observing if this response passes through or encircles the negative unity point on the real axis (Reference 37).

In essence, the OLTF is a characteristic function of the reactor system given its configuration and conditions. As the point of interest is the negative unity point on the real axis, the actual mechanism of the perturbation to the system is moot, since the system will self-sustain an excitation at this point. Typically, the transfer functions analyzed by the LAPUR code are OLTF for the core natural frequency and the decay ratio, as well as the reactivity-power closed loop transfer function (CLTF). The code outputs the Nyquist plot for each OLTF (Reference 34).

In many cases, the OLTF response for a real reactor system will not pass through the negative unity point for its operating conditions. The decay ratio is then used as a measure of the damping of oscillations for situations where oscillations are not self-excited by the system. The decay ratio is calculated by determining the distance in the frequency domain between the negative unity point on the real axis and the nearest point on the OLTF response locus; thereby establishing the margin to the onset of instability.

The decay ratio, therefore, is a parameter that characterizes the system given its conditions, and is a measure of the system's margin to instability.

In the time domain, a decay ratio of unity indicates an oscillatory mode that is exactly self-excited, and the oscillation would continue without an external mechanism driving it. A limit cycle oscillation of this type is the oscillation for a given system that once it is excited, will return to the same oscillation if additional higher modes are excited and allowed to decay. A limit cycle oscillation with a decay ratio of unity would therefore be sinusoidal after sufficient time has passed for higher order damped modes to decay.

Decay ratios less than unity indicate that the reactor system is stable as responses to a perturbation in these cases self dissipate or dampen. While an initial perturbation may excite several different modes of oscillation, the primary interest is that mode which is nearest to the limit cycle oscillation as the others will decay rapidly after the external driving mechanism is removed. The decay ratio is most easily inferred from the transient response by taking the ratio of a peak in the oscillatory response to a previous peak.

Westinghouse proposes to use the POLCA-T time domain code to perform stability analyses. In the time domain, the OLTF is not calculated directly. Instead, an artificial perturbation (i.e., an external driving mechanism) is applied to the system. In these cases, the steady state solution is changed by varying the reactor control state (typically). The external driving force is then removed by returning the reactor to its steady state control state, and the code predicts the transient behavior of the system.

The decay ratio is then inferred by observing the transient response. While this means for determining the decay ratio is less direct than calculating the OLTF in the frequency domain, and in many ways more computationally expensive, the decay ratio remains a characteristic parameter of the core for its given conditions. The decay ratio is determined based on the ratio of subsequent positive peaks in the transient response trace. Given that the response is not a perfect analytical solution, this ratio should be calculated while the positive peaks are relatively large to prevent masking from numerical noise.

The larger peaks tend to occur early in the transient before significant damping has occurred. However, in the very earliest stages of the transient, higher order modes of oscillation (many points on the OLTF locus) may be excited. However, these rapidly decay compared to the fundamental mode - which is nearest to the limit cycle. Therefore, a balance in each calculation must be preserved to ensure that the selected peaks in the transient response are not too early in the transient, yet not too late in the transient.

3.3.3.2 Scope of the Review

Section 3.1.2 of this safety evaluation report documents the NRC staff's review of the basic models and formulae that comprise the POLCA-T basic code system. The scope of the NRC staff's review in terms of their application to stability is limited to review of those models specifically related to the important phenomena affecting reactor stability, the exercise of the code to evaluate stability, and the assessment of the uncertainties in developing acceptance criteria.

3.3.3.3 Phenomenology Important to Boiling-Water Reactor Stability

Section 3.3.3.1 provides an overview of the basic phenomena driving reactor instability; therefore, the NRC staff review is focused on assessing the acceptability of the POLCA-T methodology in regards to the following important phenomena for stability evaluation:

- Acceptable neutronic and thermal-hydraulic coupling to account for the void reactivity feedback mechanisms
- Acceptable fluid flow modeling to characterize density wave propagation in the core
- Acceptable fluid flow models to capture transient flow regime transition
- Appropriate nodalization, iteration, and time step size control to adequately model instability phenomena without artificial numerical damping or artificial numerical-induced oscillations.
- Acceptable pressure drop models to adequately capture the phase lags associated with single and two-phase pressure drops.
- Acceptable fuel rod modeling to determine the fuel time constant
- Acceptable methods for determining the first harmonic mode flux shape in order to ensure that regional mode perturbations excite those oscillations nearest to the limit cycle.

The NRC staff requested that Westinghouse provide PIRT for stability analyses in RAI 6-33. In response to RAI 6-33, Westinghouse provided a stability PIRT (Reference 54). The NRC staff reviewed the contents of the PIRT for completeness and reviewed the rankings for the individual phenomena to ensure that the highly important phenomena were identified. The detailed NRC staff review is documented in Appendix A of this SE. The NRC staff found that the PIRT reflects those phenomena identified by the NRC staff as having a significant influence on the evaluation of channel, regional, and core-wide stability.

The NRC staff reviews the PIRT against the modeling capabilities of the methodology to ensure that all important phenomena are captured in the evaluation model. In the case of the POLCA-T application to stability analyses, the **[**
] Therefore, the NRC staff did not consider the PIRT directly in its review of the uncertainty assessment. Rather, the NRC staff used the PIRT and the general code information and component model qualification supplied in RAI responses to assess the capabilities of POLCA-T. As documented in greater detail in Appendix A of this SE, the NRC staff found that the highly ranked PIRT phenomena were represented by acceptable, qualified models in the POLCA-T code system.

3.3.3.4 Stability Methodology

For core-wide and regional stability analysis, the figure of merit is the decay ratio. Generally, POLCA-T is used to infer the decay ratio by performing transient calculations. These
[

] The process for performing these calculations is described in Section B.3 of the TR (Reference 1). This calculational process is substantially similar to the RAMONA-3B process previously approved by the NRC (References 6 and 7). Since the backup stability protection analysis method is predicated on analyzing the decay of an oscillation, the method may only be applied for initial steady state conditions and not during a transient.

> <u>Steady State Evaluation Condition</u>
> The POLCA-T core stability methodology is only approved for determining the decay ratio by perturbing otherwise steady state conditions.

While the NRC staff has found the overall methodology acceptable, time domain stability analysis techniques are highly sensitive to the manner in which the calculations are performed. In particular, the propagation of density waves is a key phenomenon in the acceptable modeling of BWR stability performance. When inappropriate time step controls, node sizes, or time integration techniques are employed, the transient solution may be susceptible to artificial numerical instabilities or numerical damping. The approval of a time domain stability methodology for BWRs, therefore, requires the NRC staff's review of the methods for developing acceptable nodalization schemes and review of those methods for assuring that transient core models are developed adequately to ensure artificial numerical effects do not adversely impact the predicted transient core behavior.

In particular, simulated oscillations may be numerically enhanced or damped depending on the input options selected for the: time integration technique, node size, time step size, iteration scheme, perturbation method, or other input options in the codes used to generate the nuclear data. A key example of the sensitivity of a stability analysis to an analysis input is the selection of a time integration scheme. If fully-implicit time integration is used to analyze the oscillations, the calculated oscillations are artificially damped. This artificial damping would result in a non-conservative prediction of the decay ratio. Therefore, the NRC staff requested detailed information regarding the stability methodology to ensure that the predicted decay ratios are accurate. This information was requested in several RAIs.

Another important part of the overall stability methodology is the selection of the limiting initial conditions for the transient calculation. The stability of the reactor is a strong function of particular plant parameters that may vary with cycle exposure. The NRC staff conducted a review of the determination of the limiting initial conditions as well.

3.3.3.4.1 Time Integration

In RAI 6-24, the NRC staff requested additional information regarding the degree of semi-implicitness in the time integration scheme for the POLCA-T stability analyses. In the response to RAI 6-24, Westinghouse provided **[**
] The POLCA-T results were compared to the analytical solution to determine the degree of numerical damping. The results indicate that a **[**
] insignificant numerical damping (Reference 50).

In RAI 6-19, the NRC staff requested that Westinghouse perform a sensitivity analysis to demonstrate that the standard production analysis inputs did not result in numerical damping. This analysis was predicated on the propagation of a temperature oscillation through a channel using the semi-implicit time integration technique. The results were provided to the NRC staff in Reference 50. The NRC staff reviewed the inlet and outlet temperature oscillations and confirmed that no numerical damping was introduced using the standard analysis inputs.

The NRC staff has reviewed the sensitivity analyses and concurs with Westinghouse's assessment that: (1) **[ ]** and (2) that the **[ ]** stability analyses. The NRC staff imposes the condition that the **[ ]**

<u>Semi-Implicitness Condition</u>[8]
[ ]

3.3.3.4.2 Nodalization

In RAI 6-3, the NRC staff requested that Westinghouse describe the process for generating an appropriate nodalization for models used in stability analysis. The response to RAI 6-3 states that the **[ ]** (Reference 48).

As stated in Section 3.3.1, the NRC staff has previously imposed nodalization conditions on the RAMONA-3B time domain stability methodology. These conditions specified requirements for the axial and radial nodalization. In the POLCA-T methodology, the **[ ]** Therefore, the NRC staff does not explicitly impose any nodalization conditions for the core model in the current SE noting that the analysis method inherently elects an appropriate nodalization.

However, when using **[ ]** through the POLCA-T core model. The response to RAI 6-3 provides the description of the standard production process for RCS nodalization, which is consistent with the POLCA-T qualification calculations. This standard process is a reasonable basis for determining nodalization for the RCS. Therefore, the NRC staff finds that the nodalization methodology is acceptable.

3.3.3.4.3 Time Step

During an on-site audit, the NRC staff identified an open item (Open Item 2) and issued RAI 4-11 requesting that Westinghouse address the time step size for stability calculations (Reference 47). Westinghouse provided this information as a supplement to the response to RAI 4-8 (Reference 54). The response states that the Westinghouse standard production

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[8] See also the NRC staff's review of the response to RAI 6-24 in Appendix A.

process **[** **]** The response states that the **[** **]** and was selected based on several parametric studies. The NRC staff reviewed the response and finds that: (1) **[** **]** and (2) the uncertainties developed from the qualification analyses are consistent with the standard production techniques.

On the basis that the **[** **]** the POLCA-T model (and hence numerical damping will be minimized), the qualification analyses were performed using an acceptable methodology. Secondly, since the qualification analyses are consistent with the standard production techniques, the uncertainties developed for the decay ratio are applicable to the POLCA-T stability methodology used for licensing calculations. The NRC staff **[** **]**

> Time Step Condition[9]
> POLCA-T decay ratio analyses will be performed **[** **]** To use a different time step, POLCA-T must be re-qualified against a representative sample of benchmark cases.

The re-qualification shall consider a representative sample of the cases provided in the TR Appendix B. The purpose of the re-qualification is to establish if analyses performed using a revised time step result in analysis results that are either conservative or essentially the same. Here, essentially the same means no significant change in the prediction uncertainty or bias. If the re-qualification indicates that the results are essentially the same or conservative, Westinghouse may adopt the new time step without review and approval by the NRC. However, use of a revised time step must be documented in the Westinghouse Reload Safety Evaluation (WRSE) provided to the licensee for their retention.

3.3.3.4.4 Iteration Scheme

In RAI 6-23, the NRC staff requested additional information regarding the iteration scheme for stability analyses. In particular, the NRC staff requested that Westinghouse confirm that sufficient nuclear iterations are performed between thermal-hydraulic iterations to ensure adequate convergence of the transient power and flow response during simulated oscillations. The response to RAI 6-23 states that the nuclear and thermal-hydraulic calculations are both performed during each outer-loop iteration (Reference 50). The NRC staff reviewed this information and agrees that this ensures that an adequate number of nuclear iterations are performed to model the moderator feedback phenomena for stability evaluations.

3.3.3.4.5 Perturbation (Disturbance) Method

In order to **[** **]**

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[9] See also the NRC staff's review of the response to RAI 4-8 in Appendix A.

[

]

The NRC staff reviewed the [
] In RAI 6-14, the NRC staff requested additional information regarding the sensitivity of the time domain results to the type of disturbance applied. Westinghouse provided sensitivity studies in response to RAI 6-14 (Reference 48). The detailed NRC staff review of this information is provided in Appendix A of this SE. The NRC staff found that the variation in the decay ratio with each alternative perturbation was insignificant compared to the decay ratio uncertainty. Good agreement between the predicted decay ratios provides reasonable assurance that the time domain systems model is appropriately modeling the same feedback mechanisms one would utilize in a frequency domain transfer function model.

However, the NRC staff requested additional information in RAI 6-21. It is possible, and likely, that the initial perturbation may excite several instability modes. For instance, if a sufficiently large reactivity perturbation is applied, it may be large enough to excite several higher harmonic modes depending on the eigenvalue separation of those modes. The higher modes will decay at a significantly higher rate than the first excited mode. Therefore, when inferring the decay ratio from the transient response, appropriate methods must be applied to ensure that the decay ratio is not underestimated by crediting the rapid decay of higher modes.

In response to RAI 6-21, Westinghouse provided additional descriptive details of the methodology for inferring the decay ratio (Reference 48). The response states that [

] The NRC staff finds that this methodology allows for the accurate prediction of the decay ratio of the first excited mode, and is therefore, acceptable.

On these bases, the NRC staff finds that the disturbance methodology described in the subject TR for all three stability modes (channel, core-wide, and regional) is appropriate and acceptable.

3.3.3.4.6 Determination of the Limiting Initial Conditions

Section B.9 of the TR describes the use of POLCA-T as a replacement for RAMONA-3B in the stability analysis framework described in the approved RAMONA-3B TRs (References 6 and 7). Specifically, the TR references step 3 in the overall Westinghouse stability methodology: —Establishes the process for identifying the limiting plant conditions to be evaluated.” This process is described in greater detail in the referenced TR (CENPD-295-P-A, Reference 1 in the subject TR, Reference 7 in this SE).

Section 5.1.3 of Reference 7 describes the method by which the limiting initial conditions are established. The process relies on comparison of key parameters affecting the reactor stability. These parameters are provided in Table 5-3 of the same TR. The [

]

Section 5.1.3 of Reference 7 states that several exposure points over cycle depletion are considered. These exposure points are used with the parameters in the table to determine the limiting initial conditions. In cases where several points are potentially limiting, the TR specifies that several exposure points must be considered to establish the limiting initial conditions.

The NRC staff has previously approved this methodology for selecting the limiting initial conditions. Additionally, Table 5-3 of Reference 7 correctly identifies those parameters having the greatest influence on the plant stability. Therefore, the NRC staff finds that the rationale behind the selection process remains justified, even considering modern operating strategies. The methodology is also robust in terms of explicit evaluation of all potentially limiting conditions. Therefore, the NRC staff is reasonably assured that the initial conditions identified by the stability analysis methodology will represent the actual limiting plant conditions for the cycle analyzed.

3.3.3.4.7 Methodology Aspects Unique to Regional Mode Oscillations

The NRC staff identified three aspects of the stability evaluation methodology that are unique to regional mode oscillations. These include: (1) the potential for asymmetric bypass void formation, (2) the methodology for determining the regional symmetry plane, and (3) the efficacy of the methodology for evaluating steep radial flux gradients.

3.3.3.4.7.1 Bypass Void Formation Under Natural Circulation Conditions

Bypass void formation is the subject of RAI 6-6 and RAI 6-33S1. In the stability PIRT, Westinghouse identified bypass void formation as an important phenomenon in the simulation of regional mode oscillations (Reference 54). Therefore, the NRC staff considered the POLCA-T methodology for taking into account the reactivity feedback mechanisms associated with bypass void formation.

The response to RAI 6-6 provides a description of the POLCA7 method for **[ ]** This is **[ ]** Nuclear parameters are calculated **[ ]** (Reference 48). The NRC staff reviewed this methodology and found that it captures the first order effect of the void formation on the neutron spectrum. Since the bypass void fractions tend to be small, even under conditions of natural circulation, the NRC staff finds that the methodology will be acceptably accurate to capture the phenomenon.

However, the NRC staff requested additional information in RAI 6-33S1. Several options exist in POLCA-T to model the core bypass. **[ ]** Under conditions of regional oscillations, the total core power and total core pressure drop do not vary. Therefore, while local bypass conditions may change based on the transient reactor power response during a regional oscillation, **[ ]** (see Section 3.3.3.4.7.2)). The NRC staff requested that Westinghouse justify the POLCA-T bypass nodalization for regional mode analysis in RAI 6-33S1.

The response describes the methodology employed by Westinghouse to determine the effects of local dynamic bypass void formation on the regional mode oscillation (Reference 66). The NRC staff has reviewed this method as documented in Appendix A of this SE. The NRC staff found that the methodology is appropriate and acceptable. The NRC staff, however, imposes the condition that this method be used to determine the impact (if any) of local dynamic bypass void formation on the regional mode oscillation decay ratio.

Bypass Channel Modeling for Regional Mode Oscillations Condition[10]

> When performing regional mode stability analyses, the significance and impact of dynamic bypass void formation must be assessed using the methodology described in the response to RAI 6-33S1 (Reference 66).

On a cycle-specific basis, compliance with the Bypass Channel Modeling for Regional Mode Oscillations Condition must be documented in the WRSE for retention by the licensee.

3.3.3.4.7.2 Determination of the First Harmonic Symmetry Plane

The NRC staff requested additional clarifying information in RAI 6-18 regarding the regional stability analysis methodology. Without specific analyses, it is not possible to infer the regional mode symmetry plane. The shape of the first harmonic flux determines the location of the regional mode power peaks. Westinghouse provided additional information in RAI 6-18 regarding the methodology for regional mode analyses. There are **[**

] (Reference 50).

The NRC staff has reviewed **[**

] (Reference 65).

[**]** the NRC staff requested additional information in RAI 6-7. The subject of RAI 6-7 is the capability of POLCA-T to predict the onset of an oscillation if unstable reactor conditions are present. The response to RAI 6-7 provides assurance that POLCA-T **[**
] (Reference 48). When considered in tandem with the SVEA-96 Optima2 stability tests and analyses in Section B.6.1 of the TR, the NRC staff finds that there is reasonable assurance that POLCA-T will calculate density wave oscillations when the appropriate conditions are encountered. Therefore, the NRC staff finds that there is reasonable assurance that the **[**
]

The response to RAI 6-29 also refers to a method for demonstrating that regional mode oscillations **[**
] (Reference 65). This **[**
]
Therefore, the NRC staff finds that the approach is acceptable. In its review of this approach, the NRC staff states specific requirements for the usage of this method in its SE

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[10] See also the NRC staff's review of the response to RAI 6-33 in Appendix A.

attached to CENPD-295-P-A (Reference 7). The NRC staff finds that these conditions for the instability-threshold power calculations are likewise applicable to POLCA-T.

Therefore, the NRC staff imposes these conditions on the POLCA-T **[ ]** methodology.

[ ]

The NRC staff likewise **[ ]** Similarly, during a regional mode oscillation, the power on one side of the core will increase while it will decrease in the symmetric side of the core. **[ ]**

On the basis of these methods, the NRC staff finds that the POLCA-T methodology has sufficient capability to determine the regional mode symmetry plane. Once the symmetry plane is inferred, an appropriate regional disturbance can be applied to excite the regional mode oscillation. Therefore, the NRC staff finds that the methodology is acceptable.

3.3.3.4.7.3 Radial Flux Gradient Modeling

Modeling of regional mode instabilities requires that the POLCA-T neutronic engine reliably predict radial flux gradients that are generally greater than those experienced during normal operation. This capability is required to capture the power peaking that occurs at the maxima of the radial flux higher harmonic shape. The NRC staff requested additional information in RAI 6-17 and RAI 6-22 regarding the uncertainty analysis for the regional mode oscillation decay ratio. In RAI 6-17, the NRC staff notes that radial flux gradients are expected to be greater, and that the body of qualification data does not necessarily capture the expected radial power shapes encountered during regional mode oscillations. Westinghouse provided additional information in References 48, 50, and 52.

As described in greater detail in Appendix A of this SE, the NRC staff reviewed the available qualification data justifying the applicability of the POLCA7-based kinetics engine to model sharp radial flux tilts as well as the qualification of the POLCA7 predicted **[ ]**

[ ] the NRC staff finds that the available data to qualify the methodology are sufficient to justify equivalent capability of POLCA-T to model regional and core-wide oscillations.

3.3.3.4.8 Methodology Aspects Unique to Channel Oscillations

When determining the thermal-hydraulic stability of particular channels, the channel decay ratio is calculated without neutronic feedback (Reference 1). The assumption in the analysis is that [ ]

In order to determine the decay ratio, the transient channel flow rate is used instead of power indications. The NRC staff reviewed this methodology and found it appropriate, as this is a direct measure of the degree to which the flow oscillation is damped.

Another feature of the channel oscillation methodology is that the reactor core pressure drop must be fixed because it is controlled by the rest of the core, which does not oscillate. This calculation is easily performed using the same POLCA-T model as was used for the core-wide or regional mode decay ratio calculations. As the core is comprised of many fuel bundles, however, a process must be established to determine the most unstable fuel bundle in order to evaluate the channel oscillation stability margin. The NRC staff has previously approved a process in CENPD-295-P-A to determine the potentially limiting fuel bundles (Reference 7). The [ ] The selection process must consider the [ ] to ensure that all of the potentially limiting fuel channels have been selected for specific evaluation of the channel decay ratio. Therefore, the NRC staff imposes a condition on the channel stability evaluation process.

> <u>Selection of Potentially Limiting Bundles for Channel Decay Ratio Calculations</u>
> The method for identifying the potentially limiting fuel bundles for specific decay ratio evaluation on a cycle specific basis must conform to steps (3) and (4) of Section 5.1.4.2 of CENPD-295-P-A (Reference 7).

On a cycle-specific basis compliance with the Selection of Potentially Limiting Bundles for Channel Decay Ratio Calculations Condition must be documented in the WRSE for retention by the licensee.

3.3.3.5 Qualification Basis

Many stability measurements were included in the qualification. These include both reactor stability tests and loop experiments. For the cores considered in the core-wide oscillation qualification, the [ ] Channel oscillations were benchmarked against stability testing of the SVEA-96 Optima2 bundle design at the FRIGG test loop. [ ]

[
]

In response to RAI 6-12 and RAI 6-35, Westinghouse provided additional descriptive details of the plant conditions present during the stability tests referenced in the TR (References 48 and 52).

3.3.3.5.1 Core-Wide Oscillations

The core-wide oscillation modeling capability of POLCA-T was **[** **]** The auto-regressive moving average technique (ARMA) was used to analyze signal-to-noise data during testing to determine the decay ratio.

The NRC staff compared the decay ratio prediction error to the measured decay ratio and plotted the data in Figure 3.3.3.5.1.1. The **[** **]** In RAI 6-8, the NRC staff requested that Westinghouse provide additional details regarding the capabilities of POLCA-T to predict decay ratios for reactor cores that are highly stable. In response to RAI 6-8, Westinghouse stated that the **[** **]** Therefore, given the intended application, the **[** **]** (References 48).

The NRC staff notes, POLCA-T generates **[** **]** This result is important since a time domain method **[** **]** The initial perturbation to a steady state condition may excite many oscillatory modes above the mode closest to the limit cycle. If these higher modes are excited by the initial perturbation, then they will decay rapidly in the early transient response and may lead to a non-conservative calculation for the decay ratio if the earliest part of the transient response is used to determine the decay ratio. The precise methodology for inferring the decay ratio **[** **]**

[**]** The NRC staff **[** **]** Using the **[** **]** the NRC staff plotted the measured and calculated decay ratios as a function of the power-to-flow ratio. Increasing power-to-flow ratios typically results in a small boiling length in the core. The smaller boiling length typically means a higher two-phase to single-phase pressure drop ratio, and subsequently higher decay ratios. The single-phase pressure drop is always in-phase with the core flow, while the two-phase pressure drop is out-of-phase; as the ratio of two-phase to single-phase drop ratio increases the core becomes more susceptible to core-wide oscillations because of the phase lag for the two-phase pressure drop.

The measured decay ratio data is plotted in Figure 3.3.3.5.1.2; the calculated decay ratio is plotted in Figure 3.3.3.5.1.3. The [ ] This is best illustrated for the [ ] where the calculated decay ratios are closely correlated with a linear trend line with the power-to-flow ratio. The NRC staff also found that the calculated [ ] whereas the trend lines for the [ ]

The [ ] cores were mostly comprised of [ ] The qualification dataset supports the determination that the POLCA-T stability evaluation methodology captures the effects of important physical processes on instability margin based on density wave phenomena, however, the qualification dataset provides an indication that plant and cycle conditions may [ ] The NRC staff provided another plot in Figure 3.3.3.5.1.4 demonstrating the trend in [ ] When these subset of data are compared, the trend lines are in good agreement with the calculated values and trends in the decay ratio provided in Figure 3.3.3.5.1.3. The NRC staff notes that [ ] Therefore, the [ ] However, the NRC staff further notes that the methodology is applied to analyze those conditions where the reactor is [ ]

The NRC staff requested additional information in RAI 6-35 in order to [ ] on the same basis. As described in greater detail in Appendix A of this SE, the NRC staff used the data from the response to construct [ ] (Figures A.6.1 through A.6.3). The results indicate the [ ] The NRC staff similarly provided a plot of the ratio of the calculated-to-measured decay ratio as a function of the power-to-flow ratio multiplied by the nodal peaking factor. The [ ] Therefore, the NRC staff finds that the qualification [ ] in the POLCA-T methodology.

The NRC staff compared analytic results predicted using POLCA-T to calculations performed using the NRC staff's independent TRACE-PARCS code system. The NRC staff compared the [ ] calculated decay ratio and frequencies to equivalent predictions performed using both codes with consistent perturbations. The NRC staff decay ratio calculations were performed using the TRACE v50rc3eplm version of the TRACE code (Reference 61). The comparison is provided in Table 3.3.3.5.1.1. The results indicate that the POLCA-T calculations are in [ ] with the measurements both in terms of the decay ratio and the frequency.

Calculated and measured decay ratios for the [ ] were plotted simultaneously in Figure B.7-2 of the TR. The NRC staff reviewed the figure and found that POLCA-T does not appear to indicate any [ ]

[ ] in the predicted decay ratio based on fuel type, core size, reactor power, operating domain, or plant design. The comparisons indicate that the POLCA-T stability methodology is particularly robust based on demonstrated performance over various reactor conditions ranging from stable conditions to nearly unstable conditions. Therefore, the NRC staff concludes that inferring a model uncertainty from these data is applicable to a wide range of reactor conditions.

3.3.3.5.2 Channel Oscillations

The capabilities of POLCA-T to predict channel instability phenomena were demonstrated in the TR through comparisons against stability tests performed for a SVEA-96 Optima2 sub-bundle. POLCA-T was used to predict the bundle power for the onset of thermal-hydraulic channel flow oscillations. The [.] The NRC staff finds that the qualification provides reasonable assurance that: (1) POLCA-T has the capability to predict the onset of thermal-hydraulic instability, and (2) that the density wave capabilities are sufficient to accurately predict the oscillatory phenomena for channel analysis.

In RAI 6-11, the NRC staff requested additional information regarding the applicability of the methodology to legacy fuel. The response provides the results of sensitivity analyses that bound the uncertainty in loss coefficient factors. The sensitivity analysis results [ ] The response also states that the [ ] treated in the analysis methodology based on information provided by the utility for legacy fuel (Reference 48). Additional descriptive details of the sensitivity analyses and the NRC staff review are

contained in Appendix A of this SE. On these bases, the NRC staff finds that the application of the POLCA-T methodology to legacy fuel will not have an adverse effect on the efficacy of the code to predict the channel stability characteristics.

3.3.3.5.3 Regional Oscillations

The NRC staff requested additional information in RAI 6-32 regarding the regional mode oscillation test performed at [ ] Specifically, the [ ] performed during the test was done at conditions where the reactor was [ ] The NRC staff requested that Westinghouse perform qualification analyses and provide detailed POLCA-T analysis results and to also compare the results to the previously approved RAMONA-3B code.

The response to RAI 6-32 provides a description of the calculation and also provides plots of key variables, namely [ ] (Reference 65). The NRC staff reviewed the qualification information provided and determined that POLCA-T accurately predicts the regional mode oscillation [ ] When compared against the RAMONA-3B calculations, the POLCA-T results [ ]

The NRC staff finds that the [ ] a comprehensive integral basis for the qualification of the code to [ ] The NRC staff found the performance of the POLCA-T methodology to replicate the [ ] when compared to the previously approved method. Therefore, the NRC staff is reasonably assured that POLCA-T is capable of regional mode decay ratio calculations.

3.3.3.5.4 Uncertainty Assessment and Acceptance Criteria

The code qualification includes full scale integral tests. These tests include regional mode oscillation tests performed at **[ ]**, core-wide oscillation tests performed at **[ ]** and channel stability tests performed for SVEA-96 Optima2. The qualification data are used to determine the uncertainty in the POLCA-T predicted decay ratio for the purpose of determining an acceptance criterion for the decay ratio, below which, the likelihood of an unstable oscillation to develop is sufficiently small that one is reasonably assured that they will not occur.

[ ] In RAI 6-5, the NRC staff requested that Westinghouse provid additional details regarding the determination of the measurement uncertainty. Westinghouse provided an acceptable response describing how the measurement uncertainty was determined (References 50 and 52). In RAI 6-17, the NRC staff requested that Westinghouse **[ ]** In the responses to RAI 6-17 and RAI 6-22, Westinghouse provided an adequate justification **[ ]** (References 48, 52, and 50).

The NRC staff finds that the core-wide oscillation assessment database provides an adequate basis to determine that there are **[ ]** The NRC staff finds that **[ ]** The NRC staff furthermore concludes that determining the **[ ]** the uncertainty in the POLCA-T methodology.

The decay ratio acceptance criterion is defined in terms of the prediction uncertainty and any design margin. The NRC staff requested additional information regarding the adequacy of the decay ratio acceptance criterion in RAI 6-16. In response to RAI 6-16, **[ ]** (References 48 and 65). The NRC staff reiterates its position in terms of the decay ratio acceptance criterion as a condition.

> <u>Decay Ratio Acceptance Criterion</u>[11]
> The decay ratio acceptance criterion shall be **[ ]**

Westinghouse has revised the decay ratio acceptance criterion to be consistent with the NRC staff's condition. Therefore, the NRC staff finds that the revised decay ratio acceptance criterion provides sufficient margin to provide reasonable assurance to protect against the onset of reactor instabilities.

3.3.4 Long Term Stability Solutions

The NRC staff has reviewed the POLCA-T calculational methodology insofar as it is used to predict the decay ratio for various oscillatory modes and reactor conditions. The NRC staff requested additional information based on Section B.9 of the subject TR regarding the use of

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[11] See also the NRC staff's review of the response to RAI 6-16 in Appendix A.

POLCA-T within the reload licensing framework for the various approved long term stability solutions (LTSs). In particular, the subject TR Appendix B references CENPD-295-P-A (Reference 7) in terms of the reload licensing methodology.

Reference 7 provides specific information for the reload licensing approach for plants implementing the following BWROG interim and LTSs: Interim Corrective Actions (ICA)[12], Option Enhanced I-A (Option EIA), Option I-D, Option II, and Option III. These solutions and the applicable analysis methodologies are provided in References 10, 11, 15, 16, 55, and 56.

The NRC staff requested additional information in RAI 6-4 and RAI 6-25 regarding the integration of POLCA-T in the LTS licensing methodology. The response to RAI 6-4 states that POLCA-T is used only to predict the decay ratio **[**

] (Reference 50). The particular selection of reactor conditions was evaluated by the NRC staff in Section 3.3.3.4.6. The response to RAI 6-4 states that certain licensing analyses relevant to establishing cycle-specific LTS parameters, **[**

]

[ ]

The subset of analyses in the host of LTS options where decay ratio calculations are performed is limited to those options that specify an exclusion or a controlled region on the power/flow operating map or require evaluation of regional, core, and channel decay ratio. These solutions include Option EIA, Option I-D, Option II, and back-up stability protection (BSP) for Option III plants. The specific implementation of POLCA-T for each of these LTSs was provided for the NRC staff's review in response to RAI 6-25 (Reference 65). The detailed NRC staff's review of the response is provided in Appendix A of this SE; the NRC staff found that the implementation was acceptable.

For several LTS options, it has historically been the practice to evaluate the likelihood of developing unstable regional mode oscillations based on a correlation of the core and channel decay ratios (see Figure 3-3 of Reference 7). These historical approaches were developed to simplify the evaluation the regional mode directly with one-dimensional methods. As POLCA-T includes the three-dimensional capability implicitly, the NRC staff imposes a condition requiring explicit regional mode assessment. This condition is consistent with analytical approach illustrated in Figure 3-2 of Reference 7.

> <u>Regional Oscillation Likelihood Assessment Condition</u>
> LTS evaluations of the likelihood of regional mode oscillations require explicit evaluation of the regional mode decay ratio. Calculation of the regional mode decay ratio requires that analyses be performed using a full core model.

The NRC staff notes that the decay ratio analysis is **[**

] Therefore, the NRC staff imposes the condition that the use of POLCA-T for LTS options requiring the determination of decay ratios requires that the analysis be performed on a cycle-specific basis.

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[12] ICAs were implemented in plant licensing bases prior to development and widespread implementation of the BWROG LTSs. They are mentioned here for completeness based on the contents of CENPD-295-P-A.

<u>Cycle Specific Long Term Stability Solution Analysis Condition</u>
The use of POLCA-T to determine decay ratios requires that the analysis be performed on a cycle-specific basis.

The NRC staff has not reviewed the use of POLCA-T for relevant stability analyses other than the prediction of the core-wide, regional, and channel decay ratios. Where applicable in the licensing methodology, the NRC staff finds it acceptable to utilize POLCA-T and the associated acceptance criteria for decay ratio calculations within the approved LTS licensing frameworks. Other analyses such as the calculation of the Delta Critical Power Ratio versus Oscillation Magnitude (DIVOM) curve slope using POLCA-T, have not been reviewed by the NRC staff. Therefore, these analyses must be performed using the historically approved methods, unless Westinghouse submits additional information in the form of POLCA-T TR appendices to justify the use of POLCA-T for these alternative purposes for NRC review and approval.

The NRC staff's review of POLCA-T is limited to the calculation of decay ratios for use within the BWROG LTSs relying on the calculation of exclusion regions. The BWROG LTSs and the associated procedures for establishing the appropriate exclusion regions are described in References 15, 16, and 11. Any deviation from these BWROG LTS procedures requires justification prior to implementation.

<u>Long Term Stability SolutionProcedures Limitation</u>
The NRC staff's review of POLCA-T does not include any modification to the BWROG LTSs. Therefore, any deviation from the approved BWROG LTS procedures requires justification prior to implementation.

The NRC staff has not performed a generic review of the applicability of the BWROG LTSs to all expanded operating domains mentioned in Section 3.1.6. The NRC staff's review of the POLCA-T methodology to predict the decay ratio for plants with expanded operating domains does not herein constitute the NRC staff's approval of the BWROG LTSs for these domains.

3.3.5 Conditions, Limitations, and Restrictions

This section of the SE provides a comprehensive listing of those conditions, limitations, and restrictions that are applicable to the use of POLCA-T for stability evaluations.

3.3.5.1 Steady State Evaluation Condition (Section 3.3.3.4)

The POLCA-T stability methodology is only approved for determining the decay ratio by perturbing otherwise steady state conditions.

3.3.5.2 Semi-Implicitness Condition (Section 3.3.3.4.1)

[ ]

3.3.5.3 Time Step Condition (Section 3.3.3.4.3)

POLCA-T decay ratio analyses will be performed **[**

] To use a different time step, POLCA-T must be re-qualified against a representative sample of benchmark cases.

3.3.5.4 Bypass Channel Modeling for Regional Mode Oscillations Condition (Section 3.3.3.4.7.1)

When performing regional mode stability analyses; the significance and impact of dynamic bypass void formation must be assessed using the methodology described in the response to RAI 6-33S1 (Reference 66).

3.3.5.5 **[ ]** (Section 3.3.3.4.7.2)

[

]

3.3.5.6 Selection of Potentially Limiting Bundles for Channel Decay Ratio Calculations (Section 3.3.3.4.8)

The method for identifying the potentially limiting fuel bundles for a specific decay ratio evaluation on a cycle-specific basis must conform to steps (3) and (4) of Section 5.1.4.2 of CENPD-295-P-A (Reference 7).

3.3.5.7 Decay Ratio Acceptance Criterion (Section 3.3.3.5.4)

The decay ratio acceptance criterion shall be **[**
]

3.3.5.8 Regional Oscillation Likelihood Assessment Condition (Section 3.3.4)

LTS evaluations of the likelihood of regional mode oscillations require explicit evaluation of the regional mode decay ratio. Calculation of the regional mode decay ratio requires that analyses be performed using a full core model.

3.3.5.9 Cycle-Specific Long Term Stability Solution Analysis Condition (Section 3.3.4)

The use of POLCA-T to determine decay ratios requires that the analysis be performed on a cycle-specific basis.

3.3.5.10 Long Term Stability Solution Procedures Limitation (Section 3.3.4)

The NRC staff‘s review of POLCA-T does not include any modification to the BWROG LTSs. Therefore, any deviation from the approved BWROG LTS procedures requires justification prior to implementation.

3.3.6 Conclusions Regarding Stability Evaluation

The NRC staff has reviewed the POLCA-T general model description and its qualification basis for stability application. This qualification basis includes a wide variety of core decay ratio measurements for several classes of BWR plant and fuel designs. The qualification basis therefore provides support for the wide range of application of POLCA-T to model stability phenomena and to accurately predict the decay ratio. The NRC staff found that the prediction uncertainty was adequately quantified and incorporated into an appropriate licensing evaluation acceptance criterion.

In the course of its review, the NRC staff determined that the model description was insufficient to currently justify the application of POLCA-T to perform DIVOM calculations. Therefore, the NRC staff's approval of POLCA-T to determine the channel, core-wide, and regional mode decay ratios does not constitute approval, herein, for POLCA-T to determine the DIVOM slope.

Similarly, during the course of its review, the NRC staff determined aspects of the stability methodology that were key elements and important in the execution of the code during licensing analyses. These aspects were identified, and where appropriate, are associated with specific conditions, limitations, and restrictions on the methodology. The NRC staff has documented these conditions, limitations, and restrictions in Section 3.3.5. When the POLCA-T stability evaluation methodology is exercised within the conditions, limitations, and restrictions specified in Section 3.3.5, the NRC staff finds that this methodology is acceptable for determining the margin to instability.

4 SUMMARY OF CONDITIONS AND LIMITATIONS

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.

4.1 General Usage

Often a general purpose transient analysis computer program, such as POLCA-T is developed to analyze a number of different events for a wide variety of plants. These codes can constitute the major portion of an evaluation model for a particular plant and event. Generic reviews are often performed for these codes to minimize the amount of work required for plant- and event-specific reviews. To this end, the NRC staff conducted a generic review of the model description document for POLCA-T. On this basis, the NRC staff identified generic conditions, limitations, and restrictions applicable to the usage of POLCA-T. This section of the SE provides a comprehensive listing of those conditions, limitations, and restrictions identified though the body of this SE.

4.1.1 Applicability of Conditions and Limitations on Encompassed Codes (Section 3.1.2.1)

Licensees implementing POLCA-T should provide justification that STAV7.2, PHOENIX4, POLCA7, and PARA computer codes and methodology, when approved in the licensing basis for use, are utilized in a manner that is in compliance with the conditions identified in the NRC staff SEs. The exception to this is called out in the response to RAI 11-15 (Reference 67).

If a specific plant has not been licensed for the use of the computer codes and methodology that are utilized by POLCA-T, then that licensee will need to take appropriate licensing action for application of these computer codes. Licensees will need to verify that the conditions and limitations imposed on each of the NRC approved codes (STAV7.2, PHOENIX4, POLCA7, and PARA), encompassing the POLCA-T methodology will continue to be satisfied each time the POLCA-T methodology is utilized.

4.1.2 Mixed oxide Restriction (Section 3.1.2.1)

POLCA-T is not approved to analyze cores containing MOX fuel.

4.1.3 Countercurrent Flow Limitation Condition (Section 3.1.2.2.5)

The CCFL correlation shall be revised to be consistent with the model submitted to address potential non-conservatisms for SVEA-96 Optima2 by Reference 46 prior to POLCA-T application to transient analyses where countercurrent flow may occur.

4.1.4 Use of STAV7.2-based Models (Section 3.1.2.6)

To be consistent with WCAP-15836-P-A, the STAV7.2 fuel thermal conductivity model and pellet relocation model provided in TR Sections 14.2.1 and 14.2.2.3, respectively, will be used in POLCA-T when performing licensing calculations.

4.1.5 Gadolinia Concentration Limitation (Section 3.1.2.6)

POLCA-T is only applicable to the analysis of cores loaded with gadolinia bearing fuel within the minimum-approved-maximum-gadolinia-concentration of either STAV7.2 or PHOENIX4/POLCA7 as documented in WCAP-15836-P-A and CENPD-390-P-A, respectively, or in subsequently approved submittals.

4.1.6 Encompassed Code Updates Condition (Section 3.1.2.6)

If a new NRC approved code takes the place of an existing POLCA-T code listed in —Applicability of Conditions and Limitations on Encompassed Codes" (see Section 3.1.2.1), licensees will need to verify that the downstream effects on POLCA-T result in conservative or essentially the same calculational results. Essentially the same results are within the margin of error for the type of analysis being performed. The implementation of the new code will also be in compliance with the —Applicability of Conditions and Limitations on Encompassed Codes" condition.

4.1.7 Quality Assurance for POLCA-T (Section 3.1.5)

Future release candidates of the POLCA-T code must be tested using a software test matrix that includes the revisions audited by the NRC staff as documented in Section 3.3 of Reference 47.

4.1.8 Simplified Boiling-Water Reactor Restriction (Section 3.1.6.4)

The NRC staff's approval of POLCA-T is limited to BWR/2-6 and the ABWR plant designs.

4.1.9 Code Change Limitation (Section 3.1.7)

Any changes to the POLCA-T solution techniques (i.e., calculational framework), as described in the application-specific appendices to the subject TR, that would yield inconsistency with the NRC staff approved documentation are considered by the NRC staff to constitute a departure from an element of the methodology in the safety analysis.

4.2 Control Rod Design Accident Specific

This section of the SE provides a comprehensive listing of those conditions, limitations, and restrictions that are applicable to the use of POLCA-T for CRDA analyses. These are repeated from Section 3.2.4 for completeness.

4.2.1 Initial Conditions Condition (Section 3.2.3.4.1)

Consistent with TR Section A.5.4 Item 1, POLCA-T CRDA cases must assume **[**
]

4.2.2 Control Rod Design Accident Acceptance Criteria Condition (Section 3.2.3.4.3)

Until final acceptance criteria are published by the NRC, the POLCA-T methodology will determine the extent of fuel damage using the interim acceptance criteria in Standard Review Plan Section 4.2 Revision 3 Appendix B for new reactor applications.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt these criteria for all CRDA analyses.

4.2.3 Hydrogen Pickup Model Condition (Section 3.2.3.4.3)

When utilizing hydrogen content dependent PCMI cladding failure limits from Figure B-2 of SRP 4.2, the hydrogen content must be determined using the NRC approved hydrogen pickup model described in Reference 22 or a subsequently NRC approved model.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology for determining the hydrogen content (if applicable, based on final acceptance criteria) will be determined using an NRC approved method.

4.2.4 Radiological Consequences for New Reactors Condition (Section 3.2.3.4.3)

When determining the radiological fission product inventory, the traditional (RG 1.195) and alternative (RG 1.183) source methods must include an increased inventory to account for transient fission gas release for new reactor applications. The transient fission gas release (FGR) must be calculated according to the following correlation from SRP 4.2 Appendix B:

Transient FGR = $\{(0.2286\Delta H) - 7.1419\}$

Where:

FGR = Fission gas release, % (must be > 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment.

Once final acceptance criteria are published by the NRC, the POLCA-T methodology will adopt any relevant transient FGR requirements for all CRDA analyses.

4.2.5 Standard Production Condition (Section 3.2.3.5.2)

Standard production CRDA calculations using POLCA-T must use modeling options and features that are consistent with those options and features used in the qualification calculations provided in Appendix A of the TR.

4.2.6 Advanced Boiling-Water Reactor Control Rod Drop Accident Analysis Condition 1 (Section 3.2.3.5.5.1.2)

The POLCA-T application to the analysis of the consequences of the ABWR CRDA requires that the rod drop velocity be determined based on the specific ABWR control rod design including sufficient conservatism to account for manufacturing uncertainties.

4.2.7 **[ ]** (Section 3.2.3.5.5.1.4)

POLCA-T analysis of the CRDA requires that the final calculated fuel enthalpy be adjusted to account for the **[ ]** Equation (2) in Section A.5.3.2 of the subject TR and the response to RAI 7-14 (Reference 52) must be used for calculating the bias for <u>all</u> core exposures.

4.2.8 Conservative SCRAM Reactivity Insertion Limitation (Section 3.2.3.5.6.1)

A relaxation of the conservative SCRAM reactivity insertion assumption described in TR Section A.5.4 Item 4 in the CRDA analysis is considered by the NRC staff to constitute a change in the method of evaluation in the safety analysis. Relaxation of the SCRAM reactivity insertion assumption will generate analysis results that are non-conservative relative to the approved method.

4.2.9 Advanced Boiling-Water Reacto Control Rod Drop Accident Analysis Condition 2 (Section 3.2.3.5.6.2)

Application of POLCA-T to evaluate CRDA for the ABWR requires submittal of:

1. The basis of the rod drop velocity for review,

2. An evaluation of the dose based on the NRC staff guidance in RG 1.183 or RG 1.195 and the transient FGR correlation in Appendix B of SRP 4.2 and the relevant radiological consequence analysis assumptions provided in Table 15.2 of the FSER for the ABWR, and

3. An evaluation of the coolability against the criteria in Appendix B of SRP 4.2.

4.3 Stability Specific

This section of the SE provides a comprehensive listing of those conditions, limitations, and restrictions that are applicable to the use of POLCA-T for stability evaluations. These are repeated from Section 3.3.5 for completeness.

4.3.1 Steady State Evaluation Condition (Section 3.3.3.4)

The POLCA-T stability methodology is only approved for determining the decay ratio by perturbing otherwise steady state conditions.

4.3.2 Semi-Implicitness Condition (Section 3.3.3.4.1)

[ ]

4.3.3 Time Step Condition (Section 3.3.3.4.3)

POLCA-T decay ratio analyses will be performed **[ ]** To use a different time step, POLCA-T must be re-qualified against a representative sample of benchmark cases.

4.3.4 Bypass Channel Modeling for Regional Mode Oscillations Condition (Section 3.3.3.4.7.1)

When performing regional mode stability analyses; the significance and impact of dynamic bypass void formation must be assessed using the methodology described in the response to RAI 6-33S1 (Reference 66).

4.3.5 **[ ]**

4.3.6 Selection of Potentially Limiting Bundles for Channel Decay Ratio Calculations (Section 3.3.3.4.8)

The method for identifying the potentially limiting fuel bundles for specific decay ratio evaluation on a cycle-specific basis must conform to steps (3) and (4) of Section 5.1.4.2 of CENPD-295-P-A (Reference 7).

4.3.7 Decay Ratio Acceptance Criterion (Section 3.3.3.5.4)

The decay ratio acceptance criterion shall be **[ ]**

4.3.8 Regional Oscillation Likelihood Assessment Condition (Section 3.3.4)

LTS evaluations of the likelihood of regional mode oscillations require explicit evaluation of the regional mode decay ratio. Calculation of the regional mode decay ratio requires that analyses be performed using a full core model.

4.3.9 Cycle-Specific Long Term Stability Solution Analysis Condition (Section 3.3.4)

The use of POLCA-T to determine decay ratios requires that the analysis be performed on a cycle-specific basis.

4.3.10 Long Term Stability Solution Procedures Limitation (Section 3.3.4)

The NRC staff's review of POLCA-T does not include any modification to the BWROG LTSs. Therefore, any deviation from the approved BWROG LTS procedures requires justification prior to implementation.

5 CONCLUSIONS

If the NRC's criteria or regulations change so that its conclusions about the acceptability of the methods are invalidated, the licensee referencing the report (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.

Appendix A of the TR describes the POLCA-T application methodology for CRDA. The staff has reviewed the POLCA-T general model description, the qualification basis for the CRDA application, and the combination of uncertainties used to establish appropriate acceptance criteria for CRDA analysis. The staff found that the capabilities of the POLCA-T code were appropriate to model the important phenomena dictating plant transient behavior under the conditions of CRDA.

The staff also reviewed the qualification basis and found that it was sufficient to demonstrate the applicable capabilities of the POLCA-T methodology as they are exercised in the conduct of CRDA analysis.

In the course of its review, the staff identified particular aspects of the CRDA analysis methodology that required special treatment for application to the ABWR. In these cases, the staff identified particular conditions on the POLCA-T application to the ABWR. These conditions arise predominantly due to two aspects that are unique to the ABWR relative to the operating fleet of BWR/2-6 plant designs. The first of which is the control blade design. Since the ABWR control blade design does not incorporate a velocity limiter, the staff found that the generically applicable value used in the operating plant analyses was not sufficiently justified for use in ABWR calculations. The staff imposed the condition that application of POLCA-T to the ABWR requires justification of the analytical rod drop velocity. Additionally, the acceptance criteria for new plant designs in terms of the fuel enthalpy have been updated. Therefore, the staff imposed conditions that the ABWR analysis be performed using these revised acceptance criteria consistent with the updated SRP and the ABWR FSER.

The staff has otherwise concluded that for application to the operating fleet of reactors, the qualification basis and uncertainty analysis are adequate to justify the POLCA-T acceptance criteria. However, certain conditions were identified as key aspects of the POLCA-T methodology for performing these calculations to ensure that the calculation remains within the accuracy demonstrated as part of its qualification. The applicable conditions, limitations, and restrictions are documented throughout the SE and are provided in a comprehensive listing in Section 4. When exercised with the conditions, limitations, and restrictions listed in Section 4 of this SE, the staff finds that the POLCA-T CRDA analysis methodology is acceptable.

Appendix B of the TR describes the POLCA-T application methodology for stability. The staff has reviewed the POLCA-T general model description and its qualification basis for stability application. This qualification basis includes a wide variety of core decay ratio measurements over several classes of BWR plant and fuel designs. The qualification basis therefore provides support for the wide range of application of POLCA-T to model stability phenomena and to accurately predict the decay ratio. The staff found that the prediction uncertainty was adequately quantified and incorporated into an appropriate licensing evaluation acceptance criterion.

In the course of its review the staff **[** **]** Therefore, the staff's approval of POLCA-T to determine the channel, core-wide, and regional mode decay ratios does not constitute approval, **[** **]**

Similarly, during the course of its review, the staff determined aspects of the stability methodology that were key elements and important in the execution of the code during licensing analyses. These aspects were identified, and where appropriate, are associated with specific conditions, limitations, and restrictions on the methodology. The staff has documented these conditions, limitations, and restrictions in Section 4. When the POLCA-T stability evaluation methodology is exercised within the conditions, limitations, and restrictions specified in Section 4, the staff finds that this methodology is acceptable for determining the margin to instability.

The staff has reviewed the POLCA-T code, 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 conditions and limitations described in Section 4 of this report. When exercised appropriately, the methods as documented in Reference 1 and its associated Appendices A and B are acceptable for performing the prescribed safety analyses for CRDA and stability, respectively.

6 <u>REFERENCES</u>

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) Accession No. ML071370710).

2. RPB 90-93-P-A, "Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," October 1991.

3. CENPD-284-P-A, —Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996 (ADAMS Accession No. ML072250547).

4. CENPD-285-P-A, —Fuel Rod Design Methods for Boiling Water Reactors," ABB CE, July 1996 (ADAMS Accession No. ML081770446).

5. CENPD-293-P-A, —BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," ABB CE, July 1996.

6. CENPD-294-P-A, —Thermal-hydraulic Stability Methods for Boiling Water Reactors" ABB CE, July 1996.

7. CENPD-295-P-A, "Thermal-hydraulic Stability Methodology for Boiling Water Reactors" ABB CE, July 1996.

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

9. CENPD-390-P-A, —The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000 (ADAMS Accession No. ML010100268).

10. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions. Licensing Basis Methodology for Reload Applications," August 1996 (ADAMS Accession No. ML072260045).

11. NEDO-31960, "BWR Owners' Group Long Term Stability Solutions Licensing Methodology," May 1991 (ADAMS Accession No. ML072260045).

12. NEDO-10527 Supplement 1, Rod Drop Accident Analysis for Large Boiling Water Reactors: Addendum No.1 Multiple Enrichment Cores with Axial Gadolinium, General Electric, July 1972 (ADAMS Accession No. ML081140547).

13. NEDO-10527 Supplement 2, Rod Drop Accident Analysis for Large Boiling Water Reactors: Addendum No.2 Exposed Cores, General Electric, January 1973 (ADAMS Accession No. ML081140547).

14. NEDO-10527, Rod Drop Accident Analysis for Large Boiling Water Reactors, General Electric, March 1972 (ADAMS Accession No. ML081140547).

15. NEDO-31960 Supplement 1, —BWR Owners' Group Long Term Stability Solutions Licensing Methodology (Supplement 1)," March 1992.

16. NEDO-32339, —Reactor Stability Long Term Solution: Enhanced Option I-A," March 1994.

17. WCAP-16606-P-A, —Supplement 2 to BISON Topical Report RPA 90-90-P-A," Westinghouse Electric Company, January 2008 (ADAMS Accession No. ML081280718).

18. WCAP-16081-P-A, —10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2," Westinghouse Electric Company, March 2005 (ADAMS Accession No. ML051260171).

19. WCAP-10106-P-A, —A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors," Westinghouse Electric Company, January 1986 (ADAMS Accession No. ML072070198).

20. WCAP-10841-P, —Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Programs for Boiling Water Reactors," Westinghouse Electric Company, July 1985 (ADAMS Accession No. ML072070201).

21. WCAP-15682-P-A, —Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification, and Application," Westinghouse Electric Company, April 2003 (ADAMS Accession No. ML031540701).

22. WCAP-15836-P-A, —Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006 (ADAMS Accession No. ML061220485).

23. NUREG-0800, —Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 15.0.2, —Review of Transient and Accident Analysis Methods," U.S. NRC, December 2005.

24. NUREG-0800, —Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 15.4.9, Rev. 3, —Spectrum of Rod Drop Accidents (BWR)", U.S. NRC, March 2007.

25. NUREG-0800, —Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 15.4.9.A, Rev. 2, —Spectrum of Control Rod Drop Accidents (BWR)", U.S. NRC, July 1981.

26. NUREG-0800, —Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 4.2, —Fuel System Design", U.S. NRC, March 2007.

27. NUREG/CR-6696, LAPUR5.2 Verification and Users Manual, February 2001 (ADAMS Accession No. ML010520049).

28. Hetrick, D. L., Dynamics of Nuclear Reactors, University of Chicago Press, Chicago, Illinois, 1971.

29. March-Leuba, J., Dynamic Behavior of Boiling Water Reactors, PhD Dissertation, University of Tennessee, Knoxville, Tennessee, 1984.

30. March-Leuba, J. And E. D. Blakeman, A Study of Out-of-Phase Power Instabilities in Boiling Water Reactors, 1988 International Reactor Physics Conference, Jackson Hole, Wyoming, September 1988.

31. March-Leuba, J., A Reduced-Order Model of Boiling Water Reactor Linear Dynamics, Nuclear Technology, 75, 15-22, October 1986.

32. March-Leuba, J., D. G. Cacuci, and R. B. Perez, Nonlinear Dynamics and Stability of Boiling Water Reactors: Part 1 — Qualitative Analysis, Nuclear Science and Engineering, 93, 111-123, 1986.

33. March-Leuba, J., D. G. Cacuci, and R. B. Perez, Nonlinear Dynamics and Stability of Boiling Water Reactors: Part 2 — Quantitative Analysis, Nuclear Science and Engineering, 93, 124-136, 1986.

34. NUREG/CR-6003, Density-Wave Instabilities in Boiling Water Reactors, June 1992 (ADAMS Accession No. ML063620350).

35. NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, July 1994 (ADAMS Accession Nos. ML080670560, ML080710117 and ML080710127).

36. SCIE-NRC-330-97, BWR Stability Modeling with NRC Current Generation Thermal Hydraulic System Codes, October 1997.

37. Lahey, R., Moody, F., Thermal Hydraulics of a Boiling Water Nuclear Reactor, American Nuclear Society, 1977.

38. Andersen, J.G.M., Shaug, J.C., Wirth, A.L., TRACG Time Domain Analysis of Thermal Hydraulic Stability Sensitivity to Numerical Method and Comparison to Data, Paper Presented at the Stability Symposium, Idaho Falls, Idaho, August 10-11, 1989.

39. K. S. Smith, "An Analytic Nodal Method for Solving the Two-Group Multidimensional, Static and Transient Neutron Diffusion Equation," Department of Nuclear Engineering Thesis, M.I.T., Cambridge, Mass., March 1979.

40. CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors, BR 91-255-P-A, May 1991.

41. B.E. Boyack et al., —Phenomena Identification and Ranking Tables (PIRTs) for Rod Ejection Accident in Pressurized Water Reactors Containing High Burnup Fuel," NUREG/CR-6742, September 2001 (ADAMS Accession No. ML012890477).

42. Diamond, D., et al., *Estimating the Uncertainty in Reactivity Accident Neutronic Calculations*, BNL-NUREG-66230.

43. Diamond, D., et al., *A Qualitative Approach to Uncertainty Analysis for the PWR Rod Ejection Accident*, BNL-NUREG-67430.

44. Cokinos, D., et al., *Effects of Subcooling and Rod Drop Speed on the BWR Rod-Drop Accident,* BNL-NUREG-30722.

45. WCAP-11284-P-A, —Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Electric Company, October 1989 (ADAMS Accession No. ML070890510).

46. WCAP-16078-P-A, —Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 fuel," Westinghouse Electric Company, November 2004 (ADAMS Accession No. ML050390440).

47. NRC Audit Results Summary Report, —WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model," May 2010, (ADAMS Accession No. ML100840695).

48. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, —Response to NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‗POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007 (ADAMS Accession No. ML072900261).

49. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Follow-Up Response to Question 6-9 of the NRC‘s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model‘ (TAC No. MD5258) (Proprietary/Non-proprietary),” LTR-NRC-08-8, February 7, 2008 (ADAMS Accession No. ML080430051).

50. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Follow-Up Response to NRC‘s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model‘ (TAC No. MD5258) (Proprietary/Non-proprietary),” LTR-NRC-07-62, December 12, 2007 (ADAMS Accession No. ML073580493).

51. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Response to Second Round of NRC‘s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model‘ (TAC No. MD5258) (Proprietary/Non-proprietary),” LTR-NRC-08-27, June 25, 2008 (ADAMS Accession No. ML081890191).

52. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Follow-Up Response to the Second Round of NRC‘s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model‘ (TAC No. MD5258) (Proprietary/Non-proprietary),” LTR-NRC-08-36, August 22, 2008 (ADAMS Accession No. ML082520770).

53. Powers, D., Meyer, R., *Cladding Swelling and Rupture Models for LOCA Analysis*, NUREG-0630, March 1980.

54. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Further Responses to the Second Round of NRC‘s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model‘ (TAC No. MD5258) (Proprietary/Non-Proprietary),” LTR-NRC-08-59, December 16, 2008 (ADAMS Accession No. ML083660101).

55. Implementation Guidance for Stability Interim Corrective Action, Letter BWROG-92030 to BWR Owners‘ from R. D. Binz (BWROG), March 18, 1992.

56. BWR Owners‘ Group Guidelines for Stability Interim Corrective Actions, Letter BWROG-94079 to M. J. Virgilio (NRC) from L. A. England (BWROG), June 6, 1994.

57. RG 1.77, Assumptions Used for Evaluating A Control Rod Ejection Accident for Pressurized Water Reactors, May 1974.

58. RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

59. RG 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors, May 2003.

60. U.S. Advanced Boiling Water Reactor Design Control Document, Chapter 4, General Electric Nuclear Energy, Revision 4, March 1997.

61. Gehin, J., Downar, T., Ivanov, K., March Lueba, J., —Final Report: TRACE-PARCS Testing for BWR Stability Analysis," January 31, 2007.

62. Ulses, A., Assessment of the TRACE Code Against SPERT III-E Core Reactivity Insertion Accident Data, Paper Presented at the Fourth Workshop on Advanced Reactors with Innovative Fuel, Fukui, Japan, February 20-22, 2008.

63. LTR-NRC-08-9 P, —'Westinghouse Presentation on Westinghouse Fuel Performance Update Meeting' (Slide Presentation of February 20-21, 2008) and Associated Material (Proprietary)," February 13, 2008. (ADAMS Accession No. ML081280106)

64. LTR-NRC-09-7 P, —Meeting Presentation on ‗Westinghouse Fuel Performance Update Meeting,'" (Slide Presentation of February 5-6, 2009), February 5, 2009 (ADAMS Accession No. ML090680309).

65. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, —Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‗POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, March 13, 2009 (ADAMS Accession No. ML090850349).

66. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, —Fourth Set of Responses to the Second Round of NRC's Request of Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‗POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-21, April 8, 2009. (ADAMS Accession No. ML100280999).

67. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, —Fifth Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‗POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-25, May 12, 2009 (ADAMS Accession No. ML091380095).

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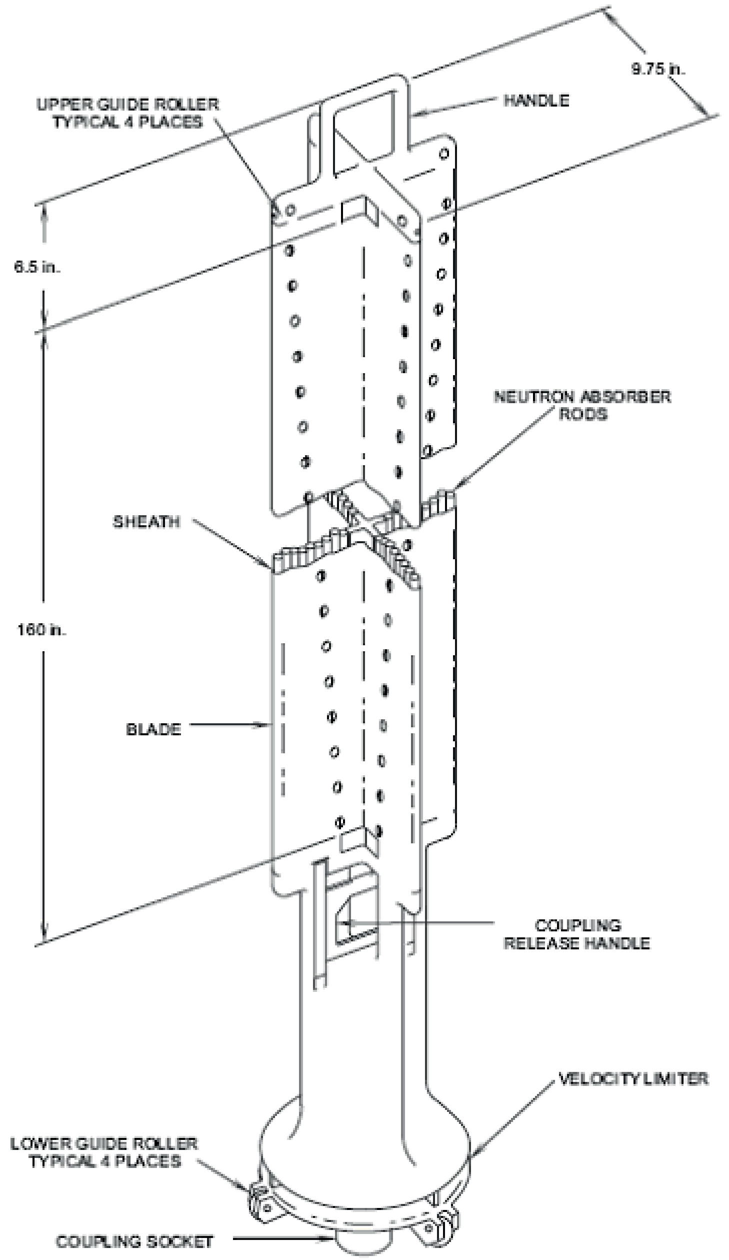

Figure 3.2.3.5.5.1.2.1: Typical BWR/2-6 Control Blade Design[13]

[13] Figure dimensions are based on the Shoreham BWR/4 FSAR.

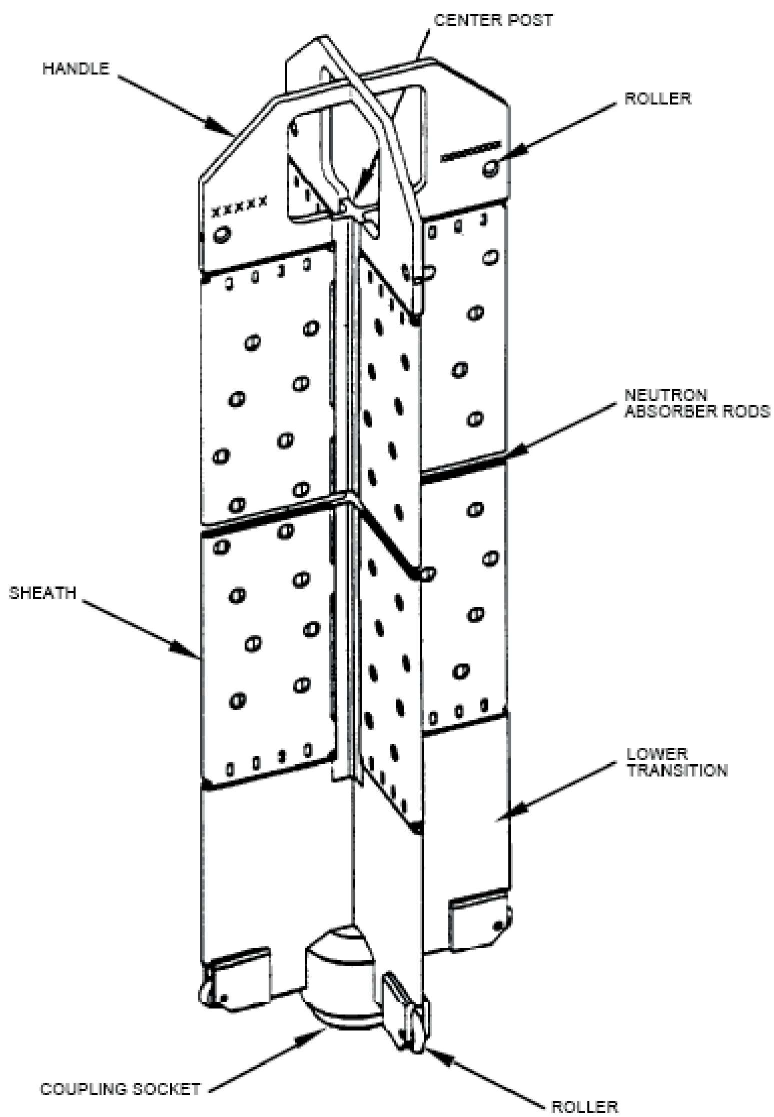

Figure 3.2.3.5.5.1.2.2: ABWR Control Blade Design (Reference 60)

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Table 3.3.3.5.1.1

[ ]		[ ]		TRACE-PARCS Calculated	
Decay Ratio	Frequency [Hz]	Decay Ratio	Frequency [Hz]	Decay Ratio	Frequency [Hz]
[ ]	[ ]	[ ]	[ ]	0.735	0.431
[ ]	[ ]	[ ]	[ ]	0.664	0.456