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framatome

ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA

ANP-10346NP-A
Revision 0

October 2019

NRC-IC-19-017
T4.12.1
Rec'd 11/11/19 HHE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 30, 2019

Mr. Gary Peters, Director
Licensing and Regulatory Affairs
Framatome Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR FRAMATOME INC. TOPICAL REPORT ANP-10346, REVISION 0, "ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA" (EPID L-2017-TOP-0067)

Dear Mr. Peters:

By letter dated December 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17355A231), Framatome, Inc. (Framatome, formerly AREVA, Inc.) submitted Topical Report (TR) ANP-10346, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. By letter dated May 10, 2019 (ADAMS Accession No. ML19099A364), an NRC draft safety evaluation (SE) regarding our approval of TR ANP-10346P, Revision 0, was provided for your review and comment. By letter dated June 20, 2019 (ADAMS Accession No. ML19175A122), Framatome provided comments on the draft SE. The NRC staff's disposition of the Framatome comments on the draft SE are discussed in the attachment (ADAMS Accession No. ML19276E207) to the final SE enclosed with this letter.

The NRC staff has found that TR ANP-10346P, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in licensing action requests, our review will ensure that the material presented applies to the specific plant involved. Requests for licensing actions that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Framatome publish approved proprietary and non-proprietary versions of TR ANP-10346P, Revision 0, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

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

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.

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

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Framatome will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Project No. 728
Docket No. 99902041

Enclosure:
Final Safety Evaluation (Proprietary)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10346P, REVISION 0,

"ATWS-I ANALYSIS METHODOLOGY FOR BWRS USING RAMONA5-FA"

FRAMATOME, INC.

PROJECT NO. 728/DOCKET NO. 99902041

1.0 INTRODUCTION

By letter dated December 15, 2017 (Ref. 1), as supplemented by letter dated March 8, 2019 (Ref. 5), Framatome, Inc. (Framatome formerly known as AREVA, Inc.) submitted a topical report (TR) which presents a methodology for the evaluation of the Anticipated Transient Without Scram with Instability (ATWS-I) for boiling water reactors (BWRs) using a updated version of the RAMONA5-FA methodology. The TR is entitled, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," and can be identified by its TR number, ANP-10346P (Ref. 2).

The RAMONA5-FA methodology was originally developed to predict the critical power response of a BWR core to regional oscillations. The methodology was approved by the U.S. Nuclear Regulatory Commission (NRC) for this purpose in TR EMF-3028P-A (Ref. 3). A number of the models from RAMONA5-FA were subsequently incorporated in the AISHA and SINANO methodologies described in ANP-3274P, which were approved for use to analyze the ATWS-I event for extended flow window (EFW) applications at Monticello Nuclear Generating Plant, Unit 1 (Monticello) (Ref. 4). Subsequently, Framatome updated the RAMONA5-FA methodology to incorporate enhancements to address important phenomena for the ATWS-I event on a generic (i.e., non-plant specific) basis. ANP-10346P provides information on the updated RAMONA5 FA methodology along with an ATWS-I specific phenomena identification and ranking table (PIRT), validation, and analysis procedure.

ANP-10346P provides: (1) a description of the models, correlations, and coupling routines within RAMONA5-FA relevant to ATWS-I modeling; (2) additional assessment of the revised RAMONA5-FA code to validate its predictions of the onset of instability and subsequent growth and mitigation of core power oscillations, as well as the thermal hydraulic response under oscillatory dryout/rewet conditions; (3) specific parameters and assumptions to be used during performance of ATWS-I analyses; (4) sensitivity studies or other technical justifications for generic analysis assumptions; and (5) sample ATWS-I problems. Since the NRC has previously approved multiple components of this methodology as part of the review of TR EMF-3028P-A and the Monticello EFW license amendment request, the primary focus of the NRC staff was on the aspects of this methodology that are novel approaches to ensure applicability on a generic basis, as well as the integration of multiple methodologies developed at different times into a single approach for generic ATWS-I analyses. However, emergent

Enclosure

issues may arise which lead to questions about prior approved methodology components within the context of this methodology.

2.0 BACKGROUND

An ATWS-I event is defined as a scenario in which an anticipated operational occurrence (AOO) occurs followed by the failure of a reactor trip to occur, which results in thermal hydraulic conditions which can allow unmitigated, unstable power oscillations to grow. Typical AOOs for which this can occur for BWRs include the turbine trip event and the two recirculation pump trip (2RPT) event, both of which cause lower core flow coupled with decreasing feedwater temperature. If no scram occurs, then the decreasing feedwater temperature will cause the core power to increase. Eventually, the combination of lower core flow and increasing core power will cross the instability boundary, and power oscillations may occur. If the oscillations grow without mitigation, they may become large enough to cause loss of adequate cooling, an increase in cladding temperature, and subsequent fuel failure leading to loss of core coolability. An acceptable means of ensuring that core coolability is maintained is to ensure that the PCT remains below 2200 °F, which is consistent with similar core coolability requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." Some further discussion of these specific events can be found in Section 3.2 of the TR.

The methodology described in ANP-10346P is intended to allow plant-specific ATWS-I analyses to be performed in order to support operation in the EFW domain on the power-flow map. The RAMONA5-FA code (Ref. 3) was originally developed to implement a long-term stability solution (LTSS), which ensures that the plant core monitoring system can detect and suppress potential power oscillations due to coupled neutronic-thermal hydraulic instabilities. An ATWS-I analysis methodology (Ref. 4) was subsequently approved for a plant-specific application at Monticello that utilized many of the constitutive models from RAMONA5-FA, but resolution of some issues required use of **[** **]** that were implemented by use of the AISHA and SINANO methodologies. The updated RAMONA5-FA methodology described in ANP-10346P provides a generic ATWS-I analysis approach that can be applied at any plant utilizing the BWR/3 through BWR/6 designs. ANP-10346P also includes additional validation necessary to demonstrate the applicability of the RAMONA5-FA methodology, as updated, to the larger oscillations (relative to LTSS conditions) and dryout/rewetting conditions that may occur during an ATWS-I event. Since the methodologies documented in References 3 and 4 will be discussed repeatedly throughout this safety evaluation (SE), for ease of comprehension in this SE, these methodologies will henceforth be referred to as the "RAMONA5-FA LTSS methodology" (Ref. 3) and the "Monticello ATWS-I methodology" (Ref. 4).

The intended purpose of the ATWS-I analysis is to demonstrate that plant specific mitigation strategies are adequate to provide reasonable assurance that core coolability is maintained. Such mitigation strategies typically include operator action to reduce the water level in the downcomer or activation of the Standby Liquid Control System (SLCS). The water level in the downcomer is reduced by terminating or reducing the flow of feedwater (FW) to the vessel. This reduction of cold FW injection causes the core inlet subcooling to decrease. Additionally, reduction of the water level in the downcomer below the level of the FW spargers causes the feedwater to fall through a steam environment and absorb additional heat, which also reduces the amount of core inlet subcooling. The reduced core inlet subcooling leads to reduction in the core power back below the instability boundary. The SLCS acts more directly to reduce core power by injection of soluble boron in the coolant.

Since the NRC review of ANP-10346P depends, in part, on the assumption that selected technical models have previously been reviewed and approved by the NRC for stability related calculations as part of the review of the RAMONA5-FA LTSS methodology and Monticello ATWS-I methodology, the SEs associated with the aforementioned documents were reviewed. In the SE for the RAMONA5-FA LTSS methodology, several limitations were outlined associated with the **[ ]** neutronic methods. However, the methodology described in ANP-10346P for use of RAMONA5-FA to analyze the ATWS-I event utilizes the ADAPKIN module, which does not have any specific limitations. No additional limitations or conditions were imposed by the NRC on the use of RAMONA5-FA for analysis of instabilities and associated power oscillations. The SE for the Monticello ATWS-I methodology differs in that it only evaluates a plant-specific methodology and thus does not contain limitations and conditions, however, no specific concerns were identified for the models and correlations that were adopted as part of the methodology described in the TR.

3.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.62 requires that the licensee/applicant provide an acceptable reduction of risk from ATWS events by inclusion of prescribed design features and demonstrating their adequacy in mitigation of the consequences of an ATWS event. Within the context of the review of ANP-10346P, the ATWS-I analyses are intended to demonstrate that the combination of automated plant functions and prescribed operator actions will be sufficient to preclude fuel failure.

The regulation in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems [(ECCS)] for light-water nuclear power reactors," is not directly applicable to the ATWS-I event because it is intended to address postulated loss-of-coolant accidents rather than ATWS events. However, this regulation does present a set of acceptance criteria for ensuring adequate cooling of fuel such that significant fuel failures do not occur.

General Design Criterion (GDC)-12, "Suppression of reactor power oscillations," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," addresses the ability of a plant to suppress power oscillations that may occur. The ATWS-I analyses show that this GDC is met by demonstrating that the prescribed operator actions are sufficient to suppress any power oscillations.

The Standard Review Plan (NUREG-0800, herein referred to as "SRP") is the primary regulatory guidance document used by the NRC staff to support review of this TR. In particular, SRP Chapter 15.8, "Anticipated Transients Without Scram" (Ref. 7), establishes acceptance criteria for ATWS events. SRP 15.8 does include additional GDCs beyond those listed above, however, they define vessel, ECCS, and containment performance requirements. This is not a significant concern for ATWS-I events, therefore, these GDCs were not considered as part of this review.

ANP-10346P describes an application of an evaluation model to perform licensing analyses for an accident. As such, additional guidance for the evaluation may be found in SRP Chapter 15.0.2, "Review of Transient and Accident Analysis Methods" (Ref. 8). This chapter includes provisions for the review of submittals related to evaluation models intended for use in analyzing postulated accidents. This guidance is intended for use with design basis accidents, and as such, is not fully applicable to the ATWS-I event. However, the guidance provides a useful framework for the NRC staff to review this TR. In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 along with the applicable acceptance criteria in SRP

Chapter 15.8 in conducting its review of the TR. In accordance with SRP Chapter 15.0.2, the review covered the areas of: (1) documentation, (2) evaluation model, (3) accident scenario identification process, (4) code assessment, (5) uncertainty analysis, and (6) quality assurance plan. To the extent possible, the NRC staff leveraged the prior review and approval of the RAMONA5-FA LTSS methodology and the Monticello ATWS-I methodology.

4.0 TECHNICAL EVALUATION

ANP-10346P describes a methodology by which the RAMONA5-FA code can be used for analysis of the ATWS-I event. The NRC staff review of ANP-10346P was performed by following the key elements of the evaluation model development and assessment process (EMDAP) outlined in Regulatory Guide (RG) 1.203 (Ref. 12) and echoed in SRP Chapter 15.0.2 (Ref. 8). While this guidance was intended mainly to address design basis accidents, the general principles can be applied to ATWS-I analysis methodologies. In summary, the areas of review were as follows:

1. Accident scenario description and phenomena identification and ranking – Framatome's break-down of the ATWS-I event and its relevant phenomena, and characterization of the consequences. The NRC staff utilized other available approved PIRTs and relevant guidance to inform their assessment of whether all the relevant phenomena are appropriately addressed in the validation basis, acceptance criteria, and/or procedure used to confirm that the acceptance criteria are met.

2. Evaluation methodology – the proposed ATWS-I analysis methodology, including initial conditions, assumptions, and approach to ensuring that the acceptance criteria are met. Since this methodology includes use of the evaluation model, by extension, this area includes the models and correlations within the RAMONA5-FA code.

3. Code assessment – the assessments performed by Framatome to validate the RAMONA5-FA performance for the thermal hydraulic and neutronics phenomena expected during ATWS-I events, particularly during unstable power oscillations and for the specific fuel designs currently used by Framatome customers.

4. Uncertainty analysis – This area is not formally required since the ATWS-I event is not a design basis event. However, the NRC staff did confirm that Framatome adequately addressed the parameters that have the most impact on the results of the analyses through conservative assumptions or sensitivity studies.

5. Documentation – The NRC reviewed Framatome's documentation of the various aspects of this analysis methodology, including ANP-10346P itself as well as various documents supporting the RAMONA5-FA code and calculational files or procedures that provide detail on the intended steps to be taken when performing ATWS-I analyses or qualifying the methodology for different plant configurations and fuel designs.

The documentation associated with ANP-10346P is captured by various calculational files, validation reports, technical references, code documentation, and ANP-10346P itself. The additional documentation reviewed by the NRC staff that were not formally submitted on the docket as References 2 or 5 are summarized in the audit report (Ref. 6). While this information was not necessary to make a safety finding, the NRC staff did confirm that the information was consistent with that presented in ANP-10346P and the request for additional information (RAI) responses. In order to resolve all key technical issues necessary to make a safety finding on

this TR, the documentation was found to include sufficient information for the NRC staff to understand the intended application and validation of the methodology described in the TR. As such, the NRC staff acceptance of the adequacy of the licensee's discussion of each area implicitly includes acceptance of the licensee documentation associated with that area.

RG 1.203 also discusses a sixth key element of the EMDAP, quality assurance (QA) processes. This aspect is not explicitly discussed in detail for this SE because the bulk of the QA processes are captured within the overall Framatome QA program, which is consistent with requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The NRC staff believes that reasonable assurance exists based on previous experience with Framatome, that the QA processes are adequate, and the documentation reviewed as part of the audit associated with ANP-10346P was consistent with a robust QA program.

Each of the first four aforementioned areas will be discussed and evaluated in the following subsections.

4.1 Accident Scenario Description and Acceptance Criteria

As per the review guidance in Chapter 15.0.2 of the SRP, the accident scenario description and phenomena identification and ranking process is intended to ensure that the dominant physical phenomena influencing the outcome of the given accident scenario are correctly identified and ranked. Once an accident scenario has been described, then figures of merit (FoMs) can be determined for use in evaluating whether acceptance criteria are met. The subsequent phenomena identification and ranking process will determine the physical phenomena affecting the FoMs and rank them by their importance. By doing so, an applicant can demonstrate that reasonable assurance exists that they are accurately capturing and modeling the dominant physical phenomena necessary for evaluation of the accident scenario in question.

Section 4.0 of ANP-10346P provides an extensive description of the various characteristics of the large coupled neutronic/thermal-hydraulic oscillations that uniquely characterize the ATWS-I event. In addition, other potential characteristics of an ATWS-I event that are potentially important are discussed, including potential prompt criticality, the possibility of boiling within bypass flow channels, and the cyclical dryout/rewetting that may be experienced by fuel. Framatome's understanding of the characteristics of the ATWS-I event were used to develop a PIRT which identifies specific physical processes and parameters that are expected to be relevant to the ATWS-I event.

The PIRT is intended to identify the dominant phenomena pertaining to ATWS-I licensing analyses. Because the RAMONA5-FA ATWS-I methodology is based on a preexisting approved methodology (the RAMONA5-FA LTSS methodology), Framatome used the ATWS-I PIRT to determine which equations and closure relations required development or enhancement in order to apply the methodology to ATWS-I. In addition to model development, Framatome also used the ATWS-I PIRT to define the types of validation and sensitivity studies that were needed to support the methodology.

An important step in the NRC staff's evaluation of the RAMONA5-FA ATWS-I methodology, therefore, was to determine whether the ATWS-I PIRT provided in ANP-10346P suitably encompassed all important phenomena for ATWS-I analyses, and whether the importance levels indicated in ANP-10346P were consistent with the NRC staff's current knowledge of the ATWS-I phenomena. In order to make this determination, the NRC staff reviewed PIRTs

developed in 2001 and 2011 under the guidance of the NRC (Ref. 9 and Section 5 of Ref. 17), more recent NRC published studies of ATWS-I scenarios (Refs. 10 and 11), and other available sources of information from open literature or internal NRC experience based on reviewing ATWS-I methodologies.

An important basis for the PIRT is identification of appropriate FoMs that correlate with the acceptance criteria for the ATWS-I evaluation. The primary acceptance criterion is the PCT, since Framatome elected to use a 2200°F upper limit on PCT to demonstrate that fuel/cladding damage sufficient to challenge core cooling will not occur. Secondary acceptance criteria are discussed in the Calculational Procedure section of this document, which are related to the timing of events in the ATWS-I accident progression (including any required mitigating actions). When appropriate FoMs are identified, the phenomena expected to affect the FoMs can be identified as well as ranked in importance.

Framatome identified three FoMs, which are evaluated by the NRC staff below:

- Oscillation inception, which is correlated with the decay ratio. Since the decay ratio describes the relative instability of a system, a higher decay ratio leads to earlier oscillation inception as well as a more rapid increase in oscillation magnitude. As such, this FoM directly affects the timing of failure to rewet, should it be predicted to occur. This is consistent with the primary FoM for the PIRT developed by the NRC staff as part of Reference 17.

- Limit cycle amplitude, which defines the worst possible oscillation that can occur for a given system and core configuration. The oscillations that arise during an ATWS-I will reach a maximum amplitude due to physical limitations on the severity of the density and power swings. Previous NRC experience (e.g., Ref. 10) indicates that the limiting amplitude is not well correlated with the decay ratio, therefore, to ensure that the worst-case power oscillations are captured, a separate FoM is necessary.

- Post-dryout, which generally encompasses the dryout and rewetting behavior. This includes cyclical dryout and rewetting, as well as periods of extended dryout due to failure to rewet. This behavior directly affects the PCT, since loss of cooling due to dryout is the primary cause of any PCT increases during the ATWS-I event that are significant enough to challenge the 2200°F limit.

Based on the NRC staff's knowledge of the ATWS-I event as correlated with the information presented in the TR, Framatome did an acceptable job of characterizing the event and the relevant phenomena. Framatome identified a key acceptance criterion, core coolability. Even though maintaining the PCT below 2200°F is not a precondition for ensuring that core coolability is maintained during ATWS-I conditions, Framatome proposes use of this acceptance criterion as a proxy for core coolability. This proxy for core coolability is already considered in NRC regulatory requirements as an acceptable way to ensure that core coolability is maintained. In addition, Framatome identified that the timing of specific transitions would be important in providing reasonable assurance that the prescribed operator actions would occur in adequate time to mitigate the consequences of the event. This information was used with the event and phenomena characterization to develop a set of high importance phenomena that must be adequately captured by the ATWS-I analysis methodology. The ranking of phenomena as described in the PIRT documented in ANP-10346P is consistent with the NRC staff's understanding of the ATWS-I event.

The NRC staff's conclusions regarding the ATWS-I PIRT were used to guide other portions of the review, particularly the NRC staff's review of ANP-10346P, namely the Evaluation Model, Code Assessment, and Uncertainty Analysis sections of this SE. The following table indicates which section of this SE is associated with each of the high and medium ranked phenomena from the PIRT documented in ANP-10346P. The low ranked phenomena are also captured or otherwise dispositioned in the analysis methodology, but are not expected to have enough of an impact on the ATWS-I evaluation results to require a high level of fidelity or sensitivity studies. While not explicitly mentioned in the below table, the integral benchmarks discussed in Sections 4.3.6 and 4.3.7 provide valuable validation of the ANP-10346P methodology's ability to conservatively predict the important phenomena affecting the FoMs associated with the ATWS-I event.

High Importance Phenomena from Table 4-1 of ANP-10346P	**NRC Evaluation (including relevant sections from this SE)**

High Importance Phenomena from Table 4-1 of ANP-10346P	NRC Evaluation (including relevant sections from this SE)

Medium Importance Phenomena from Table 4-1 of ANP-10346P	NRC Evaluation (including relevant sections from this SE)

As a result of the above discussion, the NRC staff has determined that Framatome appropriately characterized the ATWS-I scenario, identified the appropriate acceptance criteria, and constructed a PIRT that identified the most important phenomena and processes for the analysis methodology to capture. The NRC staff considerations in determining whether the analysis methodology is acceptable with respect to each phenomenon are discussed in the following sections, as described in the above table. In general, the oscillation inception FoM was addressed by examining how the given model or correlation affects the timing of oscillation onset. The limit cycle amplitude and post-dryout FoMs were generally considered through use of the PCT as a proxy, since a conservative application of these FoMs would be expected to increase the PCT. In several cases, the FoMs were not explicitly evaluated because the model or correlation was already established to accurately capture the phenomenon of interest.

4.2 Evaluation Methodology

Chapter 15.0.2 of the SRP describes the review of the evaluation model as part of the transient and accident analysis methods. The associated acceptance criteria indicate that models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. Restated in terms of the review procedures provided in Section III of Chapter 15.0.2, it must be determined if the physical modeling described in the theory manual and contained in the mathematical models is adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.

A number of models have been previously reviewed and approved by the NRC for similar purposes, so the scope of the NRC staff review was limited to that necessary to confirm the applicability of these models to the ATWS-I event. The models described in ANP-10346P are discussed in individual subsections below.

<u>Major Assumptions</u>

Section 4.16 of the TR identifies major assumptions made in RAMONA5-FA ATWS-I relative to the previously approved RAMONA5-FA LTSS methodology. These major assumptions were identified and justified primarily through engineering judgment, based on extensive application experience with the approved RAMONA5-FA methodology which is largely similar to the current methodology.

The major assumptions identified in Section 4.16 of the TR are: (1) the three dimensional (3D) nodal adaptive neutron kinetics methodology is assumed adequate for ATWS-I and (2) new water property functions are used and [ ]. The NRC staff's discussion and evaluation of these two assumptions is contained in Sections 4.2.1 and 4.2.3.11.5 of this SE, respectively.

4.2.1 Review of ANP-10346P Section 5.1 – Neutronics

The ANP-10346P methodology uses an adaptive 3D neutron kinetics solution with [ ] to determine the time evolution of the 3D neutron flux distribution during anticipated transient events. This neutronic solution methodology is identical to that used in the RAMONA5-FA LTSS methodology and the Monticello ATWS-I methodology. This adaptive 3D neutron kinetics solution is a [ ] methodology, which means that it solves for the

neutron flux level at each discretized axial level in each fuel assembly in the core, [ ]. The neutronic and thermal hydraulic solutions are coupled on a nodal level as well. Importantly, this means that the coupled neutronic/thermal-hydraulic methodology has sufficient fidelity to accurately resolve any anticipated axial and radial oscillation pattern, including core-wide and side-to-side radial mode behavior (including more complex modal interactions such as rotating modes) as well as single-channel instability. This was a key reason why the [ ].

During the large-amplitude oscillations that are characteristic of the ATWS-I event, up to and including limit cycles with dryout and failure to rewet, the neutronic solution becomes even more highly-peaked spatially and undergoes larger variations over time relative to LTSS applications with smaller oscillation amplitudes. However, based on the NRC staff's knowledge and experience with similar neutronics methodologies, this behavior is not expected to challenge the ability of the methodology to accurately represent the physical behavior under these conditions. In fact, the [ ]. Therefore, the NRC staff has concluded that the adaptive 3D neutron kinetics solution remains applicable and appropriate for this application.

Because the neutronic methodology in ANP-10346P did not change with respect to the RAMONA5-FA LTSS methodology, and this neutronic methodology remains suitable for ATWS-I applications, the NRC staff did not perform a detailed review of the neutronic methodology as a whole. However, the NRC staff did review the methodology for artificially applying noise to the neutronic solution during a transient calculation. The NRC staff's previous experience has indicated that the method of applying noise is important for correctly determining the timing of oscillation onset, which affects the ability of the methodology to predict whether operator actions occur sufficiently early to mitigate the potential public safety consequences of ATWS-I.

To assist in determining whether the implementation of artificial noise was acceptable for ATWS-I, the NRC staff asked for additional information in RAI-13. In the RAI response (Ref. 5), Framatome discussed that the noise is applied [ ] This information provides a high degree of confidence that both the in-phase and out-of-phase modes can be adequately and reliably excited in a timely fashion (i.e., shortly after one or both modes become unstable), [ ].

The NRC staff finds the implementation of [ ] in analysis of the ATWS-I event using RAMONA5-FA as described in ANP-10346P to be acceptable, with a limitation and condition to ensure that excessive "tuning" of [ ] does not occur without justification. The use of the neutron kinetics solution implemented in RAMONA5-FA was also found to be acceptable, based on previous NRC approvals and the known ability of this methodology to capture neutron kinetics responses similar to those expected during an ATWS-I event.

4.2.2 Review of ANP-10346P Section 5.2 – Fuel Thermodynamics

The following subsections discuss specific aspects of the fuel thermodynamics models that are relevant to accurately capture the temperature response in the fuel and subsequent heat transfer to the coolant during an ATWS-I event.

4.2.2.1 Review of ANP-10346P Section 5.2.1 – ATWS-I Fuel Pin Heat Conduction

The methodology described in ANP-10346P determines the time-dependent axial and radial temperature distribution in the "average rod" within each fuel assembly, as well as in the "hot rod" (peak power rod) within each assembly. The average rod temperatures and heat generation rate are used [

]

At each axial level in the assembly, a one-dimensional (1D) radial time-dependent transient temperature calculation is performed from the radial center of the fuel pin to the outer surface of the cladding, similar to the Monticello ATWS-I and RAMONA5-FA LTSS methodologies. This approach is on par with other state-of-the-art methods and provides a suitably accurate and realistic calculation for oscillatory ATWS-I conditions. Consistent with the previous methodologies, [ ], which is acceptable based on the fact that the model exhibits good agreement with experimental benchmarks, as discussed in Section 4.3 of this SE.

Unlike the [ ] in the Monticello ATWS-I methodology, the ANP-10346P methodology solves the radial temperatures [

]

Therefore, the NRC staff finds the ANP-10346P fuel rod conduction methodology to be acceptable based on its use of previously approved modeling approaches combined with state of art computational solution schemes appropriate for the intended application.

4.2.2.2 Review of ANP-10346P Section 5.2.2 – ATWS-I Heater Rod Conduction Model

A separate heat conduction model is used for calculating time-dependent axial and radial temperature distribution in heater rods representative of the KATHY facility. The NRC staff determined that the only difference between this model and the one in the Monticello ATWS-I methodology was that the latter calculated the [

]

Because the heater rod conduction model in the ANP-10346P methodology is more accurate than the previously-accepted model used in the Monticello ATWS-I methodology, and is used for the same scope and range of application – namely, to determine the heater rod temperature response during the KATHY ATWS-I experiments – the NRC staff concludes that the previous approval of the heater rod conduction model in the Monticello ATWS-I methodology is applicable to the ANP-10346P methodology, and no further review of the model was performed.

4.2.2.3 Review of ANP-10346P Section 5.2.3 – Heat Transfer Coefficient

The ability of the fluid to transfer heat from the outer surface of the clad or heater rod is strongly dependent on the phase of the fluid (liquid, vapor, or both) and the ability of the liquid phase to contact the surface. The ANP-10346P methodology calculates a wetted heat transfer coefficient (HTC) under single-phase conditions using [ ] correlation, a wetted HTC under two-phase conditions using [ ] correlation, a dry HTC using [ ], and models for transitions between these regimes.

The [ ] single-phase liquid correlation and the [ ] correlation are the same as in the RAMONA5-FA LTSS methodology and the Monticello ATWS-I methodology. [

] The NRC staff concludes that the single phase liquid and boiling heat transfer models are acceptable based on their previous validation and approved use in the RAMONA5-FA LTSS methodology, and that the regime transition and [ ] are acceptable because they are based on realistic physical principles and demonstrate good agreement with measured data in the benchmarks given in Section 6.0 which cover a wide range of conditions applicable to ATWS-I.

The dry HTC is determined using a correlation [

]

The NRC staff issued RAI-3 to address the NRC staff's concerns regarding the acceptability of using a [ ]. In particular, RAI-3 requests more information to justify [ ]. In the RAI response (Ref. 5), the licensee provided additional plots

[ ]

However, the NRC staff observed the following:

(1) the magnitude of the power oscillations following failure to rewet was [ ];

(2) the dry HTC correlation was [ ]; and

(3) Framatome only used the [ ].

Additional data points were provided by Framatome from [

] This is due partly to the fact that KATHY provides prototypical ATWS-I conditions, and the NRC staff does not expect the heat flux to go significantly beyond the KATHY data range without causing the fuel to exceed 2200 °F.

The data provided by Framatome demonstrates a clear correlation between [

] that is reasonably bounding by incorporating the conservatisms discussed above, as items (2) and (3).

[

]

4.2.2.4 Review of ANP-10346P Section 5.2.4 – Hot Fuel Pin Model

As discussed in Section 4.2.2.1 of this SE, the ANP-10346P methodology provides a separate calculation for temperature and heat transfer in the hot fuel pin as opposed to the average fuel pin, which the NRC staff finds to be an acceptable approach to both calculate the maximum cladding temperature and provide realistic coolant temperatures and reactivity feedback for the neutronics solution.

[

] The NRC staff finds this approach acceptable because it provides conservative [ ], which is expected to increase the calculated PCT values during ATWS-I analyses.

4.2.2.5 Review of ANP-10346P Section 5.2.5 – Material Properties

The ANP-10346P methodology uses fuel pellet and cladding thermophysical properties based on [ ]. The NRC staff finds this approach acceptable for use in the RAMONA5-FA ATWS-I calculations because these models account for all important fuel characteristics relevant to ATWS-I, including the [

].

Appendix A of Reference 5 includes an update to ANP-10346P that, among other changes, appends Appendix D, which presents modified fuel rod models that account for chromia doping of the UO_2 fuel pellets. The fuel thermal conductivity model was adapted from the approved RODEX4 model in Reference 18. The [ ] model was developed by benchmarking to the approved RODEX4 model in Reference 18. Because these models are based on previously reviewed and approved models for chromia doped fuel, the NRC staff finds these models acceptable for use in characterizing chromia doped fuel properties for ATWS-I analyses performed using the methodology as described in ANP-10346P.

4.2.2.6 Review of ANP-10346P Section 5.2.6 – Pellet Clad Gap Heat Transfer Coefficient

The gas gap between the fuel pellet and cladding may introduce a large thermal resistance which affects both the amplitude and phase shift of fluctuations in heat flux at the cladding outer surface during a given oscillation period. In turn, the decay ratio and oscillation frequency of predicted ATWS-I oscillations may be significantly affected, which may impact whether the fuel remains protected within the time required for the ATWS-I mitigation actions to take effect. Due to burnup and history effects, the fuel-clad gap in twice- and even once-burned fuel will typically be closed at normal operating conditions, resulting in only a small thermal resistance; however, after the recirculation pump trip during the postulated turbine trip with bypass (TTWB) and 2RPT ATWS-I events, the gap will typically re-open and result in significant thermal resistance that must be accurately accounted for in the ATWS-I methodology.

[ ]

[ ]

In RAI-2, the NRC staff requested additional information on how the fitting parameters for the gap conductance model were determined from measured data, particularly when direct experimental validation for each parameter was not possible or not available. In particular, RAI-2 requested additional information to better understand the method of [ ] Because these values cannot be directly measured but have a potentially significant effect on stability behavior, these values could have subsequently been adjusted [

] After examining the relevant models and experimental database, the NRC staff has concluded that the use of [ ] is acceptable because it provides a physically reasonable model of gap behavior as a function of fuel conditions (burnup, temperature, etc.) in relevant ATRIUM fuel types and, furthermore, the good agreement with [ ] provides strong evidence that these values remain applicable and acceptable under BWR stability conditions.

The NRC staff's review of the gap heat transfer coefficient model concluded that it includes the important physics required to calculate the intra-pin heat transfer behavior during a postulated ATWS-I event, including the initial transient, onset and growth of oscillations, dryout/rewet phase, and high-temperature failure-to-rewet phase of the event. The models for calculating [ ].

[ ] Because of this, the NRC staff issued RAI-4 to request justification that the gap heat transfer coefficient model provides a reasonable and accurate representation of gap behavior during postulated ATWS-I events.

In the RAI response, Framatome indicated that the [ ], as discussed above. [ ] While these data do not provide a direct "separate-effects" validation of the fuel and gap heat transfer models, the close overall agreement with these measured integral effects data, with little or no average bias in the errors, provided the NRC staff with confidence that the fuel and gap heat transfer was modeled in a reasonable and acceptable manner.

To provide the NRC staff with further understanding of the role of the gap model during ATWS-I events, the NRC staff issued RAI-11 which requests additional sensitivity results for one or more linear stability benchmark cases and a simulated ATWS-I event by adjusting the gap conductance values. In the RAI response, Framatome provided the sensitivity results for the linear and nonlinear cases by artificially adjusting the gap conductance [ ] Based on this [ ], the NRC staff concluded that the gap conductance model has [ ] on the stability predictions for regional mode cases. For the single linear benchmark case in which the global mode was dominant [

], respectively, compared to the result with non-adjusted conductance. Framatome indicated that [

] In this particular case, the more significant sensitivities were such that the base calculation represented a reasonably conservative result. However, the interrelationship between the gap conductance model and the stability phenomena is expected to be at least somewhat sensitive to the specific scenarios being analyzed.

The [ ] adjustment in gap conductance also had a relatively mild effect for the ATWS-I sample problem, for which the sensitivity results were provided in RAI-11 as well. [

] Any impact due to uncertainties in the gap conductance model that may result in challenges to the regulatory limit would be expected to occur in situations where operator action is necessary to prevent the PCT from exceeding regulatory limits, and where relatively small margins exist between the licensing basis operator action time and the time at which operator action would be too late to stop the PCT from increasing beyond the regulatory limit. In such cases, a significant increase in the decay ratio may lead to an earlier failure to rewet and allow the PCT to increase for a longer time prior to mitigation. Once the margin in operator action time is appropriately justified, including any consideration of the gap conductance uncertainty, the sensitivities are not expected to change significantly from cycle to cycle. However, they may change when a new fuel design is introduced that changes the characteristics of the geometry or materials used in modeling the fuel rod, gap, and cladding. Consequently, a limitation and condition is necessary to verify that the uncertainties associated with the gap conductance model continue to be small enough to be readily accommodated by the available margins in operator action time.

Based on the similarity [ ], inclusion of the important physics relevant to ATWS-I, close agreement of the RAMONA5-FA ATWS-I results to measured BWR stability data, and [ ] of the stability results under most scenarios to variations in gap conductance, the NRC staff concludes that the fuel rod heat transfer model, including the gap conductance model, are acceptable for use in the ATWS-I analyses. In order to address specific scenarios where the gap conductance model may become important, a limitation and condition was imposed to require evaluation of the uncertainty in gap conductance for certain changes in fuel design, as described in the previous paragraph.

4.2.2.7 Review of ANP-10346P Section 5.2.7 – Radial Power Deposition Distributions in Fuel Pellets

The ANP-10346P methodology determines the radial power distribution within fuel pellets using [ ]. The NRC staff has reviewed the methodology and determined that it provides the needed accuracy for calculating the radial power distribution in fuel pellets, including [ ] Therefore, the NRC staff finds the radial power distribution methodology to be acceptable.

4.2.3 Review of ANP-10346P Section 5.3 – Thermal-Hydraulic Model

The RAMONA5-FA ATWS-I methodology described in ANP-10346P utilizes a [ ] TH model comprised of [ ]. A description and the NRC staff evaluation of the thermal-hydraulic model is given in the following subsections.

4.2.3.1 Review of ANP-10346P Section 5.3.1 – General Description of the System Considered

The general system modeling in ANP-10346P consists of nine main components as shown in Figure 1 and is identical to the vessel methodology in the RAMONA5-FA LTSS methodology. [ ].

Accurate modeling of the pressure losses and flow inertia in the vessel flow path is important for correctly determining flow rates and other core parameters during ATWS-I events; this is especially true for in-phase (core-wide) oscillations in which the total core flow rate experiences large time-dependent changes which become coupled to time-dependent flow rate changes in the surrounding components. Vessel flow inertia is particularly dependent on the recirculation pump model which is evaluated in Section 4.2.5.1 of this SE; however, the pressure losses and flow inertia in the remaining vessel components are relevant to ATWS-I analyses as well.

Downcomer 1
Downcomer 2, with NDC2 parallel paths
Lower Plenum 1
Lower Plenum 2
Core, with NPC parallel paths
Core Upper Plenum
Standpipes
Steam Separators
Steam Dome

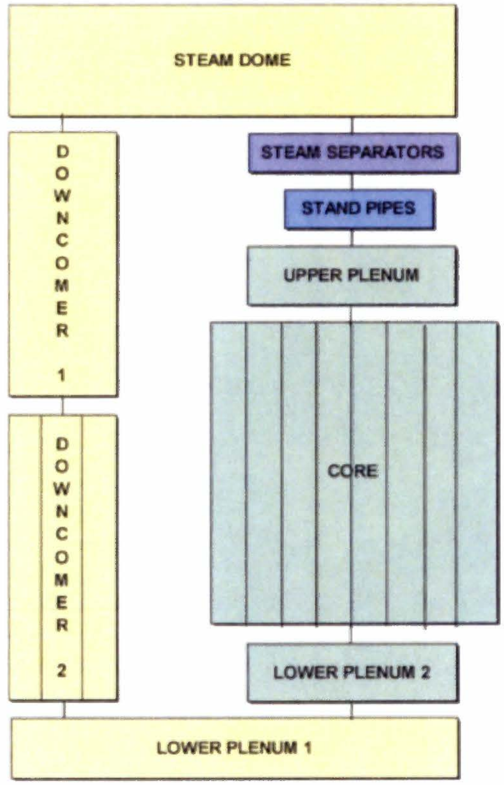

Figure 1 – Loop parts in the vessel hydraulics model (ANP-10346P Figure 5-3)

Because the ANP-10346P methodology contains the same vessel hydraulics treatment as the RAMONA5-FA LTSS methodology, which is approved for use in LTSS stability analyses including the analysis of in-phase oscillations where vessel pressure losses and flow inertia are important, these aspects of the vessel hydraulics model were not reviewed for ANP-10346P and the existing approval of these modeling aspects from the RAMONA5-FA LTSS methodology remains applicable. Furthermore, the RAMONA5-FA LTSS methodology is approved for 2RPT LTSS analyses, which is an identical event to ATWS-I 2RPT except that ATWS-I 2RPT assumes failure to scram. Although this failure to scram allows for larger-amplitude oscillations and therefore larger oscillations in flow rate and other thermal hydraulic parameters in the vessel components, these conditions do not impose additional physical modeling requirements on the vessel thermal hydraulic methodology, and this vessel methodology remains suitable for ATWS-I 2RPT.

The ATWS-I TTWB event is similar to ATWS-I 2RPT in that both involve a dual recirculation pump trip; however, the TTWB event requires modeling of turbine stop valve closure and turbine bypass valve opening, which impact the pressure response in the vessel including pressure wave propagation.

Additionally for the TTWB event, the decrease in feedwater temperature due to loss of feedwater heaters must be modeled, and the vessel model must be able to accurately model the mixing of the cold feedwater with the saturated liquid leaving the steam separator, and accurately transport this fluid through the downcomer and lower plenum to ensure proper timing

and magnitude of core inlet temperature decrease during the event. Accurate calculation of the time-dependent core inlet temperature is necessary to correctly predict the oscillation onset timing and magnitude. If the water level falls below the feedwater inlet to the vessel (feedwater spargers), significant heating of the subcooled liquid feedwater and condensation of the steam in the downcomer may occur, which may affect the core inlet temperature behavior as well.

Details on the steam line flow dynamics, recirculation pump model, jet pump model, steam separator model, and feedwater sparger condensation model are given Sections 5.4 through 5.5.4 of ANP-10346P and are evaluated later in this SE.

The core region consists of a number of parallel fuel assembly channels and [ ] or bypass channel. The bypass channel accounts for the inter-channel flow (between channel boxes) as well as the flow through the internal water rods in each assembly. There are numerous leakage paths from the lower plenum to the bypass region. [ ]

The NRC staff issued RAI-1 to better understand the process for passing thermal hydraulic information from MICROBURN-B2 to RAMONA5-FA for ATWS-I analyses and ensure consistent solutions between the two codes during initial steady-state conditions. In the RAI response, Framatome provided a detailed description of the [ ]. These inputs to the RAMONA5-FA ATWS-I methodology are the same as used in the approved RAMONA5-FA LTSS methodology. Both versions of the RAMONA5-FA code use these inputs to perform a thermal hydraulic calculation [ ]

[ ], the NRC staff concludes that the approach for determining the initial thermal hydraulic solution in the RAMONA5-FA ATWS-I methodology is acceptable.

Bypass flow, including water rod flow, constitutes a relatively small fraction of the total flow through the core, which limits its hydraulic impact during oscillations. However, direct gamma heating of the bypass flow may cause localized boiling in the bypass region, which may have a significant effect on the power level of neighboring fuel bundles due to neutronic feedback. Bypass voiding is most likely when stagnant or reversed bypass flow is experienced, which

occurs at very low core flow rates due to the relatively large gravitational pressure head of the bypass liquid column.

However, including [ ]. Therefore, the NRC staff concludes that modeling the bypass [ ] is expected to give conservatively high PCT results and is therefore acceptable for the ATWS-I analyses.

<u>Vessel Nodalization</u>

In RAI-9, the NRC staff requested additional information to ensure that the vessel nodalization provides sufficient fidelity for liquid and vapor transport in the vessel such that the system behavior including PCT is accurately predicted for ATWS-I events. In the response to RAI-9, Framatome specified the number of nodes used in each region of the vessel model under the base nodalization scheme, and these node numbers were [ ] for the nodalization study in RAI-9. Framatome clarified that the most limiting nodes in the model, [ ].

During in-phase oscillations, the coolant flow rate and void fraction in the primary circulation loop will oscillate along with the oscillations in the core. These time-varying thermal hydraulic quantities in the vessel will impact the recirculation loop momentum dynamics and may therefore impact the stability characteristics of the system. This impact is expected to be negligible for out-of-phase oscillations because the thermal hydraulic conditions outside the core remain essentially constant in this case.

In the vessel nodalization sensitivity cases provided in RAI-9, performed for the Brunswick sample problem, [ ], showed good agreement with the measured stability data using the base vessel nodalization. Furthermore, the NRC staff expects that other BWR plants will behave similarly with respect to vessel nodalization, with no significant differences that would be expected to materially change this finding. Therefore, the NRC staff concludes that the momentum-related effects associated with vessel nodalization are expected to be insignificant and the base vessel nodalization used in the TR is acceptable in this regard.

The vessel nodalization is expected to have an additional effect, with respect to the time-dependent core inlet subcooling. This is due to the effect of numerical diffusion on energy transport in the vessel liquid. Therefore, Framatome provided a plot of core inlet subcooling versus time during the Brunswick sample problem for each nodalization case. The time-dependent subcooling behavior was [

]

As a result of this small change in inlet subcooling, the [

] The relative insensitivity and [ ] in both the failure-to-rewet time and PCT, [ ] will consistently produce results more or less conservative than other nodalizations, leads the NRC staff to conclude that the base vessel nodalization as specified in the RAI-9 response is reasonable and acceptable for use in the RAMONA5-FA ATWS-I methodology.

4.2.3.2 Review of ANP-10346P Section 5.3.2 – [ ]

This section of ANP-10346P describes the spatial discretization scheme used by the RAMONA5-FA ATWS-I thermal hydraulic core channel solution. [

]. This method is suitable for determining the flow behavior during normal conditions (i.e., upward flow through the bundle) as well as transient behavior such as periodic flow reversal expected during large-amplitude ATWS-I oscillations. Therefore, the NRC staff finds this spatial discretization scheme, which is the same as used in the RAMONA5-FA LTSS methodology, to be acceptable for ATWS-I applications.

4.2.3.3 Review of ANP-10346P Section 5.3.3 – Vapor Generation Rate

The nodal vapor generation rate in ANP-10346P is calculated [

]. This model is the same as in the Monticello ATWS-I methodology, and similar to the RAMONA5-FA LTSS model except for the modifications to allow for the [

]; therefore, these modifications from the previously-approved methodology are appropriate. The NRC staff has reviewed the new model and determined that it acceptably models vapor generation during ATWS-I events.

4.2.3.4 Review of ANP-10346P Section 5.3.4 – Mass Conservation

The ANP-10346P methodology solves separate liquid and vapor mass conservation equations, using [

]

Large-amplitude oscillations may exhibit any of the following flow scenarios: co-current upward flow (liquid and vapor both flowing upward), co-current downward flow (liquid and vapor both flowing downward), and counter-current flow (liquid and vapor flowing in opposite directions). The NRC staff's review of the mass conservation model has determined that it properly accounts for all of these possible flow scenarios. This, in addition [ ] and its ability to give realistic behavior at or near fully-voided conditions, has led the NRC staff to conclude that the mass conservation model in ANP-10346P is acceptable for ATWS-I analyses.

4.2.3.5 Review of ANP-10346P Section 5.3.5 – Energy Conservation

The ANP-10346P methodology uses a [

]

Since scalar quantities such as enthalpy are defined at the center of control volumes and vector (or directional) quantities such as mass flow rate and velocity are defined at the edges of control volumes, the energy balance which includes energy entering or leaving through each edge of the control volume uses different scalar cell indices depending on the direction of flow through each control volume edge. In principle, these directions can be different at the bottom and top edge of each volume. The NRC staff determined that the energy equation formulation properly accounts for all possible combinations of flow directions for the bottom and top edges by making suitable adjustments to the enthalpy index for the donor cell scheme in calculating the rate of energy flow in or out of the control volume for each phase. Thus, the energy balance is properly conserved for any flow situation for both the liquid and vapor phases.

The ANP-10346P methodology differs in the implementation of the overall core energy balance by including an [

] In RAI-10, the NRC staff requested additional information on the behavior of the [ ] during postulated ATWS-I events. In the RAI response, Framatome provided a plot of [

] In any case, the NRC staff expects that this effect would have only a small impact, at most, on the ATWS-I results. Therefore, the NRC staff concludes that the implementation of the [ ] is acceptable.

4.2.3.6 Review of ANP-10346P Section 5.3.6 – [ ]

Section 5.3.6 of ANP-10346P discusses the approach for determining [

]

[

]

[

]

[

]

[

]

[

]

4.2.3.7 Review of ANP-10346P Section 5.3.7 – [ ]

The ANP-10346P methodology uses [ ], to account for flow inertia and acceleration terms and their effect on the time-dependent pressure drops. [ ] However, the NRC staff finds this implementation acceptable for the reasons given in Section 4.2.3.4 of this SE.

[ ]

[ ]

[ ] With respect to momentum conservation, the basic phenomena and modeling requirements remain the same for large-amplitude oscillations characteristic of ATWS-I, and the [ ] and acceptable for this use.

Special treatment is provided in ANP-10346P (the same as in the RAMONA5-FA LTSS methodology) to calculate the pressure response due to valve closures in the steam line, which is relevant for ATWS-I TTWB events. Details of this special treatment are given in Section 5.4 of ANP-10346P and evaluated in Section 4.2.4 of this SE. Since the pressure waves dissipate

rapidly once they reach the larger volumes of the vessel, this treatment is not necessary for the vessel and core regions. Therefore, the [ ] is acceptable for use in the vessel and core regions for the reasons stated above.

<u>Acceptability of the [</u>]

In a BWR assembly, the liquid and vapor phases will in principle travel at different velocities; these velocities depend on a mass and momentum balance for each phase separately, as well as on mass and momentum exchange between the phases. [

]

In methodologies such as the one described in ANP-10346P, [

]

[

]

[

]

[

]

[ ] Conclusion

Because the ANP-10346P methodology uses [ ], and because this approach remains suitable for large-amplitude ATWS-I oscillations – including the ability to model reversed and counter-current flow and the demonstrated conservatism of the single-momentum-equation approach – the NRC staff finds the momentum conservation model in ANP-10346P to be acceptable.

4.2.3.8 Review of ANP-10346P Section 5.3.8 – Pressure Calculation

Section 5.3.8 of ANP-10346P describes the methodology for calculating the time-dependent [

]

[

] Therefore, the NRC staff finds the pressure calculation methodology to be acceptable.

4.2.3.9 Review of ANP-10346P Section 5.3.9 – Steam Dome Equations

ANP-10346 describes an [

]

Therefore, the NRC staff finds the steam dome model acceptable.

4.2.3.10 Review of ANP-10346P Section 5.3.10 – Recirculation Flow

ANP-10346 describes an [

]

Both the TTWB and 2RPT ATWS-I event scenarios involve a dual recirculation pump trip, which is also true for the LTSS analysis scenarios for which the RAMONA5-FA LTSS methodology has been previously approved. [ ]; therefore, the NRC staff finds the recirculation flow model acceptable for ATWS-I applications.

4.2.3.11 Review of ANP-10346P Section 5.3.11 – Constitutive Equations

4.2.3.11.1 Review of ANP-10346P Section 5.3.11.1 – Friction and Two-Phase Friction Multiplier

ANP-10346 describes the same friction factor correlations as the RAMONA5-FA LTSS methodology and uses the [ ] which is also available in the RAMONA5-FA LTSS methodology. The implementation in the ANP-10346P methodology, [ ] via additional accounting for reverse and counter-current flow. This change is necessary for properly treating the reversed and/or counter-current flow experienced during large-amplitude ATWS-I oscillations. The NRC staff has determined that the implementation of the friction and two-phase multipliers is acceptable and reasonable for this application.

4.2.3.11.2 Review of ANP-10346P Section 5.3.11.2 – Local Pressure Loss Models

ANP-10346 describes essentially the same local pressure loss model as the RAMONA5-FA LTSS methodology, with the primary exception being that the ANP-10346P methodology accounts for the possibility of reversed flow. As in the previous section, the NRC staff has determined that this implementation, including treatment of reversed flow, is appropriate and acceptable for ATWS-I applications.

4.2.3.11.3 Review of ANP-10346P Section 5.3.11.3 – Abrupt Contraction/Expansion Pressure Change Model

ANP-10346 describes a similar abrupt contraction/expansion reversible pressure change model as the RAMONA5-FA LTSS methodology, except for additional accounting for reverse and counter-current flow compared to the RAMONA5-FA LTSS methodology. This change is necessary for properly treating the reversed and/or counter-current flow experienced during large-amplitude ATWS-I oscillations. The NRC staff finds this treatment of reversed and counter-current flow to be acceptable and reasonable for this application.

4.2.3.11.4 Review of ANP-10346P Section 5.3.11.4 – [ ]

[

] is acceptable for ATWS-I calculations.

4.2.3.11.5 Review of ANP-10346P Section 5.3.11.5 – Thermodynamic Steam-Water Properties

The ANP-10346P methodology uses the IF97 steam tables for fluid properties as a function of enthalpy and pressure, the same as used in the Monticello ATWS-I methodology. This is an improvement over the RAMONA5-FA LTSS methodology, [

] This provides the best available representation of thermodynamic fluid properties at the full range of possible conditions during ATWS-I, and therefore the NRC staff finds this implementation acceptable.

4.2.3.11.6 Review of ANP-10346P Section 5.3.11.6 – [ ]

In order to determine [

] As a result, the NRC staff finds the [ ] to be acceptable for the purpose of establishing the parameters of [ ] outside the core.

4.2.3.12 Review of ANP-10346P Section 5.3.12 – Numerical Integration Techniques

The RAMONA5-FA LTSS methodology utilizes [

]

[

]

[

]

[

]. Furthermore, the benchmark results presented in ANP-10346P demonstrate the numerically stable and robust performance of the methodology up to and including large amplitude oscillations typical of ATWS-I analyses. Therefore, the NRC staff finds the numerical integration technique acceptable.

Core Axial Nodalization

The NRC staff's experience has shown that the calculated decay ratio of thermal hydraulic oscillations in other codes may be significantly artificially dampened by numerical errors (numerical diffusion) in the underlying equations, and this effect may be strongly influenced by factors such as timestep size and spatial discretization scheme. [

]. In addition, the NRC staff has identified that the coarse nodalization associated with a 25 uniform axial nodalization scheme may lead to significant error in the oscillation decay ratio due to an insufficiently spatially resolved axial void profile, particularly near the bottom of the channel, and a resulting effect on neutronic feedback and oscillatory behavior.

The NRC staff issued RAI-8 to request justification that the axial nodalization scheme used in the ANP-10346P methodology provides sufficient numerical fidelity to accurately represent the stability behavior for ATWS-I. In the RAI response, Framatome provided both a discussion and numerical results to support this position. Framatome discussed three possible effects of numerical diffusion with regard to stability. The first effect is the kinematic diffusive "spreading" of solution variables over time as fluid moves along the channel. Framatome stated that this effect [

]. The NRC staff finds this reasoning to be logical and consistent with the theoretical formulation of the solution methodology presented in ANP-10346P.

[

] has only a small or minimal impact on the calculated stability behavior for the RAMONA5-FA ATWS-I methodology.

The third effect of diffusion discussed by Framatome is related to the momentum formulation itself, with respect to the effect on the momentum components of the density head and axial distribution of friction resulting from increased axial attenuation of density waves. Framatome stated that the improved axial resolution of void fraction gradients afforded by decreased node size, as proposed previously by the NRC staff and contractors, may be of more importance than the kinematic effect of numerical diffusion for which the Courant number plays a direct role. [

]

The discussion provided in the RAI-8 response supports a conclusion that the core axial nodalization scheme and associated numerical errors would be expected to be relatively small for ATWS-I applications. However, to confirm the effect that nodalization may have on the code results for ATWS-I applications, the NRC staff reviewed the results of the nodalization sensitivity study provided by Framatome in the RAI-8 response. In this study, Framatome increased the nodalization from [ ] axial nodes in the core, which is the proposed value for the methodology, to [ ] axial nodes in the core. The NRC staff reviewed Framatome's approach and determined that the nodalization increase was performed in a suitable manner, [ ]; this ensures a consistent approach for determining the neutronics and thermal hydraulics initial conditions to ensure that the conclusions of the RAMONA5-FA nodalization study are valid.

The finer nodalization resulted in increased decay ratios for all cases included in the nodalization study – namely, the KATHY stability tests, the linear reactor benchmarks, the Oskarshamn-2 nonlinear benchmark, and the Brunswick TTWB sample problem included in the nodalization study. For the KATHY linear stability tests and the linear reactor benchmarks, the decay ratio increased by amounts ranging from [ ] when comparing the finer nodalization results to the base nodalization results. [ ]. The effect of nodalization on frequency was [ ] Hertz (Hz) change compared to the base nodalization case). For the Oskarshamn-2 nonlinear benchmark and the Brunswick sample TTWB problem, the decay ratio [ ] The larger growth rate also led to earlier failure to rewet by approximately 20 seconds in the Brunswick TTWB sample problem.

After failure-to-rewet, the time-dependent PCT values appeared to be [ ] degrees Celsius (C)) on average for the finer nodalization case compared to the base case. As a result, the maximum PCT throughout the event was

[ ] C higher with the finer nodalization for this problem. However, the NRC staff concludes that this difference is likely due to the finer nodalization case reaching failure to rewet earlier than the base case, allowing failure to rewet to extend to lower elevations on the hot rod before the oscillations are suppressed, compared to the base case. These lower elevations would likely correspond to higher average LHGR values. Even if failure-to-rewet did not extend lower in the finer nodalization case, the smaller node sizes mean that the limiting node in the base nodalization case is split into two nodes in the finer nodalization case, and the lower of these two finer nodes would have a slightly higher LHGR than the larger node overlapping this location in the base case. The NRC staff suspects a cause for the higher PCT [

] Therefore, the NRC staff concludes that finer nodalization does not intrinsically cause higher PCT in the failure-to-rewet regime. Therefore, no penalty or added conservatism is necessary to account for this apparent increase in PCT for finer nodalizations due to the modest nature of the increase in PCT, inherent conservatisms in the methodology, and the lack of evidence that finer nodalization would capture new phenomena which could have a significant impact on the PCT.

Although the finer nodalization resulted in greater instability – faster oscillation growth and earlier failure to rewet – in all linear and nonlinear analysis cases, the NRC staff determined that the "base nodalization" of [ ] in the core is acceptable because it gives the most consistent and non-biased overall agreement with the measured decay ratio values across the various stability benchmarks. Further rationale for the acceptability of this nodalization scheme was provided by Framatome – namely, [

]

The NRC staff concludes that the "base nodalization" of [ ] nodes in the core is acceptable for use in the RAMONA5-FA ATWS-I methodology. This determination is based primarily on the good agreement of the RAMONA5-FA ATWS-I methodology with measured stability data across the broad range of experimental conditions when using [ ] nodes in the core, compared to the [ ] As discussed previously in this section of the SE, the NRC staff also considered potential sources of error due to numeric diffusion and determined that they would not be significant for the RAMONA5-FA ATWS-I methodology.

4.2.4 Review of ANP-10346P Section 5.4 – Steam Line Flow Dynamics

As discussed in Section 4.2.3.7 of this SE, the ANP-10346P methodology uses [

]. However, a special model for the steam line is included in the ANP-10346P methodology (identically to the RAMONA5-FA LTSS methodology) to calculate the propagation of pressure waves only within the steam line. For ATWS-I, this is relevant for calculating the pressure response after turbine valve closures following a turbine trip, as well as the pressure response following safety relief valve (SRV) closure and re-opening which may occur during the oscillatory phase of the TTWB event.

[

] The NRC staff reviewed this model and found it to be a logical and acceptable method for determining pressure response in the steam line and vessel. This is primarily because, as discussed in Section 4.2.3.7 of this SE, [

]. However, no detail is provided regarding how the steam line modeling and valve responses are verified to capture reasonable behavior during the ATWS-I event for specific plants. This is highly dependent on plant specific configurations, closure time, and setpoints, so a limitation and condition will ensure the resulting behavior (e.g., flow rates through the valves and pressure drop across the steam line(s)) from the steam line model is reasonably representative of expected plant-specific behavior during an ATWS-I event.

The NRC staff has determined that the steam line flow dynamics model provides an accurate and realistic approach for calculating pressure response during ATWS-I events including TTWB, and therefore the NRC staff finds this model acceptable with the condition that licensees must provide justification that their steam line modeling will appropriately capture expected variations in the pressure and flow boundary conditions for the ATWS-I event.

4.2.5 Review of ANP-10346P Section 5.5 – Special Models

4.2.5.1 Review of ANP-10346P Section 5.5.1 – Recirculation Pump Model

The recirculation pump model determines the relationship between pump rotational speed, pump torque, pump flow rate, and pump head. These define the steady state operating characteristics of the recirculation pumps as well as their transient behavior. The primary relevance to stability analyses is in determining the recirculation pump coastdown behavior after a recirculation pump trip, as well as the recirculation pump inertia which has an important impact on the growth rate and limit cycle amplitude of global flow oscillations. Note that the effect on regional flow oscillations is much smaller, as the total core flow rate remains relatively constant in that case.

The recirculation pump model in ANP-10346P is identical to the model in the RAMONA5-FA LTSS methodology, which was approved for LTSS analyses including the case of in-phase oscillations. The NRC staff has reviewed these models for ANP-10346P and has concluded

that the models include all necessary physics and remain acceptable for instability events up to and including large-amplitude limit cycle oscillations.

4.2.5.2 Review of ANP-10346P Section 5.5.2 – Jet Pump Model

Unlike the recirculation pump model, [ ]. However, as with any pressure term in the primary loop, the calculated pressure head may affect the transient behavior during rapid pressure changes (such as immediately following a turbine trip) as well as affect the stability behavior particularly during global oscillations.

The jet pump model in ANP-10346P is identical to the model in the RAMONA5-FA LTSS methodology, which was approved for LTSS analyses, including the case of in-phase oscillations. The NRC staff has reviewed these models for ANP-10346P and has concluded that the models include all necessary physics and remain acceptable for instability events up to and including large-amplitude limit cycle oscillations.

4.2.5.3 Review of ANP-10346P Section 5.5.3 – Steam Separator Model

The modeling of the steam separator – in particular, its flow inertia, as well as the flow rate of vapor leaving the circulation loops and entering the steam dome above the coolant level (known as carry-under) – may have a significant effect on the stability characteristics of the reactor system. Flow inertia has a particularly strong impact for core-wide (in-phase) oscillations.

The steam separator model in ANP-10346P is identical to the model in the RAMONA5-FA LTSS methodology. This model determines the steam separator flow inertia based on [ ].

The flow conditions and behavior of the steam separator follow the same physical principles and exhibit the same general characteristics under ATWS-I conditions as under smaller amplitude LTSS oscillation conditions, and therefore the NRC staff concludes that the steam separator model, which was previously approved for the RAMONA5-FA LTSS methodology, is applicable and acceptable for ATWS-I applications in ANP-10346P.

4.2.5.4 Review of ANP-10346P Section 5.5.4 – Feedwater Sparger Condensation Model

When the water level in the vessel downcomer is below the level of the feedwater inlet (feedwater spargers), significant heating of the subcooled feedwater liquid as well as condensation of the saturated steam may occur as the liquid flows downward through a steam environment. This can affect the core inlet temperature as well as the system pressure.

ANP-10346 describes the same model as the RAMONA5-FA LTSS methodology to model the condensation rate as a function of [ ].

Once the water level falls below the feedwater spargers, the nature of this condensation phenomenon is the same during ATWS-I as it is during other events currently approved for

analysis using the RAMONA5-FA LTSS methodology. Additionally, the model provides physically reasonable and realistic relationships with physical parameters. Therefore, the NRC staff finds the feedwater sparger condensation model acceptable for use in ANP-10346P for ATWS-I analyses.

4.2.5.5 Review of ANP-10346P Section 5.5.5 – Dryout and Rewetting Model

The prediction of dryout and possible subsequent rewet of the hot rod is of primary importance to ATWS-I analyses due to the dramatic increase in PCT associated with sustained dryout. Under sufficiently high cladding-to-coolant heat flux for a sufficient duration, all liquid in contact with the cladding surface evaporates, leaving only vapor in contact with the cladding surface ("dryout" conditions). Because vapor is much worse than liquid at conducting/convecting heat from the cladding surface, the temperature of the cladding (and also the fuel pellet) quickly increases after the onset of dryout. Due to the large, rapid changes in flow rate and thermodynamic quality of the coolant adjacent to the cladding surface during thermal hydraulic oscillations, there is a possibility that liquid will once again come into direct contact with the cladding surface ("rewet"), lowering the cladding temperature due to improved heat transfer. However, rewetting of the cladding surface becomes more difficult as the cladding temperature (and therefore the evaporation capability) increases; this may lead to a runaway condition in which the liquid flow is no longer able to come into contact with the cladding surface for long enough to fully reverse the increase in cladding temperature. Under such a condition, the cladding temperature may "ratchet" up through multiple cycles of heatup and limited cooldown due to rewetting, or experience a continuous increase in temperature due to loss of rewetting ability. If the cladding temperature increase is not mitigated, very high cladding temperatures which might challenge the ATWS-I acceptance criteria may result. In this case, the cladding temperatures can only be brought down again by reducing the heat generation rate (power level) in the fuel; during ATWS-I, this is done either by increasing the average void fraction in the core (accomplished via water level reduction) or injection of soluble boron into the core (via the standby liquid control system).

Because of its strong impact on the PCT during ATWS-I events, the dryout and rewetting model in ANP-10346P was one of the primary focuses of the NRC staff's review. The ANP-10346P methodology provides the same fundamental approach as the Monticello ATWS-I methodology, but a fundamentally different approach than the RAMONA5-FA LTSS methodology, to determine dryout and rewet of the hot rod surface. The RAMONA5-FA LTSS methodology **[**

] The wetting or dryout status of the cladding surface is used to determine the heat transfer regime (nucleate boiling, transition boiling, or film boiling heat transfer regimes), and heat transfer coefficients are applied correspondingly.

However, based primarily on analysis of the KATHY dryout/rewet test data presented in Section 6.5 of ANP-10346P, Framatome concluded that a different modeling approach for determination of dryout and rewet behavior provided a better fit to the data under oscillatory conditions representative of ATWS-I. This model, similar to the one provided in the Monticello ATWS-I methodology, **[**

]

A thorough review of a similar dryout/rewet model was performed by the NRC staff in its review of the Monticello ATWS-I methodology. In that review, the NRC staff concluded that the model was acceptable for the plant-specific application for which the methodology was submitted. For the ANP-10346P review, the NRC staff reviewed the dryout/rewet model with particular focus on determining the acceptability of differences in the model relative to the Monticello ATWS-I methodology, as well as the applicability and acceptability of the model for generic application.

The NRC staff's review determined that the ANP-10346P dryout/rewet model is largely similar to the dryout/rewet model used in the Monticello ATWS-I methodology, including [

]. Although the NRC staff's review and approval of the Monticello ATWS-I methodology was performed on a plant-specific basis, in its evaluation of the Monticello ATWS-I methodology dryout/rewet model the NRC staff did not note any particular limitations or shortcomings of the model which may specifically limit its use for other plants or operating conditions. For ANP-10346P, the NRC staff further examined the model and determined that the experimental benchmarking, as discussed in Section 4.3.5 of this SE, covered a sufficiently broad range of representative ATWS-I conditions such that the model may be acceptably applied to the current fleet of BWRs on a generic basis.

However, some differences were noted between the Monticello ATWS-I methodology and ANP-10346P dryout/rewet models, and these are evaluated in additional detail in the remainder of this section. These differences include the [

].

[

] The NRC staff finds the [

] to be conceptually reasonable on a physical basis, and the strong – and perhaps slightly improved – agreement of the dryout/rewet model with the cyclic dryout/rewet behavior observed in the KATHY experiments leads the NRC staff to find this revised model to be acceptable.

The NRC staff examined the addition of a [ ] to the ANP-10346P model, relative to the Monticello ATWS-I methodology, based on consideration of the flow conditions under which the dryout/rewet model was derived. [

] In RAI-5, the NRC staff asked for further information to justify the applicability [

].

In the RAI response, Framatome indicated that [

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Specifically, Framatome applied the [ ] to the base critical power reduced order model (CPROM). As depicted in Figures 8 through 14 of the RAI-5 response, the [

], the NRC staff concludes that the dryout/rewet model with [ ] provides an acceptable representation of dryout behavior for the full range of quality and void fraction conditions expected during ATWS-I.

ANP-10346 describes a process for fitting the CPROM correlation to steady-state CPR data which differs from the [ ] process used for the Monticello ATWS-I methodology. This new process was determined by the NRC staff to be acceptable based on the evaluation provided in Section 4.2.9 of this SE.

In RAI-2a, the NRC staff requested additional information on how the fitting parameters for the dryout/rewet model were determined from measured data, particularly when direct experimental validation for each parameter was not possible or not available. In the RAI response, Framatome described the fitting process in detail, including a combination of [ ]. The NRC staff reviewed the RAI response in detail and determined that all parameters were fitted to the data in a consistent, logical, and well-defined fashion to provide the most accurate and acceptable prediction of both steady state and transient behavior applicable to ATWS-I.

The NRC staff has reviewed the dryout/rewetting model in detail, including evaluating the generic applicability of the model as well as evaluating the differences relative to the similar model in the Monticello ATWS-I methodology. The NRC staff has concluded that the model is acceptable because it is based on realistic physical principles, exhibits close agreement to measured CPR data under steady state conditions, and agrees closely with measured data under transient conditions for a wide range of operating conditions which reasonably encompass the expected range of conditions expected to occur during postulated ATWS-I events in the current fleet of BWRs.

4.2.6 Review of ANP-10346P Section 5.6 – Plant Control and Protection Systems

The NRC staff reviewed the plant control and protection systems methodology provided in Section 5.6 of ANP-10346P, including the implementation of:

- Pressure control system consisting of turbine control, bypass valve and safety and relief valve (SRV),
- Plant protection systems (PPS) including recirculation pump trips,
- Feedwater control system, for water level control

The pressure control system model is required to accurately model the system pressure response following a turbine trip and possible cycling of the SRVs during a TTWB event. Modeling of the recirculation pump trip function of the PPS is relevant for both the TTWB and 2RPT ATWS-I events, to determine realistic timing of the recirculation pump trip and associated core flow rate reduction. Modeling of the feedwater control system is relevant for both the TTWB and 2RPT ATWS-I events to allow the water level to automatically adjust to the setpoint value by adjusting the feedwater flow rate, both during the initial event progression as well as after the operator action to reduce the water level setpoint.

These models are essentially the same as in the RAMONA5-FA LTSS methodology, with the primary exception that ANP-10346P added a manual operator actions model to reduce the water level during an ATWS-I event by allowing the user to specify the start time of operator actions as well as a setpoint to which the water level will be reduced. The model assumes the feedwater pumps trip at the specified start time and calculates the coast down behavior of the feedwater pumps, after which the feedwater controller maintains the water level at the new lower level based on the user-defined setpoint.

The NRC staff has reviewed these models and has concluded that these models are acceptable because they realistically and adequately represent the plant control and protection systems behavior during postulated ATWS-I events.

4.2.7 Review of ANP-10346P Section 5.7 – Numerical Time Integration

The numerical scheme used for time integration (time marching) in the neutron kinetics, thermal hydraulics, and fuel rod thermodynamics equations has a significant effect on the numerical robustness and accuracy of the solution. In particular, the large temporal and spatial gradients associated with rapidly-changing conditions within the core and vessel during ATWS-I oscillations increase the potential for large numerical errors in the solution, particularly during sharp changes in the solution such as the expected changes due to the dryout and rewet phenomena. Such errors may lead to results which depart significantly from reality and prevent accurate determination of the system response with respect to the ATWS acceptance criteria.

In the ANP-10346P methodology, the neutron kinetics, fuel thermodynamics, and vessel hydraulics equations are solved separately, with [ ] integrate each of these three calculation domains. Enforcing [ ] in each domain ensures numerical consistency, improves numerical robustness, and avoids loss of accuracy in the overall coupled solution.

As described in Sections 4.2.1 and 4.2.2 of this SE, the neutron kinetics and fuel thermodynamics equations are integrated [ ] in time. The most significant impact of this is [ ].

[ ]. Discussion and sensitivity studies regarding vessel and core nodalization were requested in RAI-9 and RAI-8, respectively, and these are discussed in Sections 4.2.3.1 and 4.2.3.12 of this SE.

The NRC staff issued RAI-12 to request additional details to determine the acceptability of the timestep control scheme and the values used for the benchmarks and sample problem in ANP-10346P. In the RAI response, Framatome described the timestep control parameters specified in the input file. The NRC staff reviewed these parameters and determined that they provide adequate capability to control the numerical timestep size to ensure robustness and accuracy of the solution. Importantly, the NRC staff determined from the RAI response that the same timestep control parameters were used for all benchmark cases and the sample problem in ANP-10346P. This ensures consistency and validity of the methodology across all benchmarks. [ ]. The NRC staff accepts this disposition and has placed a limitation and condition on the methodology to enforce this approach for selecting timestep control parameter values.

Results of a timestep sensitivity study were provided in the RAI-12 response, by varying the values of the timestep control parameters for the Brunswick sample problem. These sensitivity cases, which modeled timestep size differences that varied by up to [ ] throughout the event, exhibited only minor differences in oscillation growth rate and failure-to-rewet times (no more than 10 seconds across the sensitivity cases). The failure-to-rewet time varied in an unpredictable fashion with no clear trend as a function of the timestep control parameter values. Because of this lack of a clear trend, and because the “base” timestep control values

demonstrated good agreement with measured data across all benchmarks documented in ANP-10346P, the NRC staff finds the base timestep control values to be acceptable for use in RAMONA5-FA ATWS-I applications.

4.2.8 Review of ANP-10346P Section 8.0 – Calculation Procedure

A portion of the ATWS-I analysis methodology described in ANP-10346P is not captured within the RAMONA5-FA code or various assessments of the performance of its constituent models and correlations. In order to perform an ATWS-I analysis, an analyst must follow prescribed steps in order to ensure that the plant-specific analyses are performed in a manner consistent with the assumptions within the code and the intent of the methodology in demonstrating regulatory compliance. A discussion of the key guidance provided in ANP-10346P for performance of ATWS-I analyses is provided in the following subsections.

Statepoint Definition

Section 8.0 defines the procedure to be used for plant-specific ATWS-I analyses using the ANP-10346P methodology. The procedure defines the characteristics of the statepoint to be analyzed. This includes [

] The NRC staff has reviewed the proposed statepoint definition and determined that it is specified in an acceptable manner for ATWS-I and is consistent with the approach for MELLLA+™ applications that has previously been reviewed and approved by the NRC staff (e.g., Ref. 14).

Overview of the Analysis Procedure Defined in ANP-10346P

Details of the analysis procedure and their acceptability are described in this section. Note that ATWS-I plant-specific calculations are performed for the first cycle planned for utilization of the EFW operating domain, and subsequently, new calculations are performed only when a new fuel is introduced or another change that could impact the ATWS-I analysis is implemented that requires a new license amendment. Therefore, the calculation procedure must ensure that the

ATWS-I results bound cycle specific variations, such that the analyses remain applicable for all EFW cycles that would be allowed by implementation of the proposed license amendment.

The calculation procedure relies on the evaluation of [

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[

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[

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[

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[

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Evaluation of Step 3

After review, the NRC staff identified concerns with the calculation procedure, beginning with the first paragraph described under Step 3. [

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[

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[

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As a result of these concerns, the NRC staff has determined that the procedure defined in the first paragraph of Step 3 of ANP-10346P does not give sufficient assurance that a PCT of 2200°F will not be exceeded for all operating cycles and exposure points.

For reasons discussed below, the NRC staff has concluded that the procedure described in Steps 3.a through 3.c, as well as in the response to RAI-15 (with the additional requirement to address core designs that deviate significantly from the reference equilibrium core design used in the ATWS-I analyses, such as transition cores), provide an acceptable means of obtaining reasonable assurance that a PCT of 2200°F will not be exceeded for any cycle and exposure point during EFW operation. Alternatively, plant-specific applications may choose to augment the analyses and/or discussions provided in the first paragraph of Step 3 to provide this reasonable assurance, in lieu of performing Steps 3.a through 3.c. The NRC staff will review these justifications on a plant-specific basis.

<u>Evaluation of Step 3.a through Step 3.c, and the RAI-15 and RAI-16 Responses</u>

In RAI-15, the NRC staff requested details of a specific process that may be used to determine that the margin described in Step 3.a is sufficient to ensure applicability to all EFW cycles. In RAI-16, the NRC staff requested additional information to ensure that the modeling assumptions remain appropriate when considering their effect on the time of oscillation onset.

The process provided by Framatome in the RAI-15 response involves [

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The response to RAI-15 clarifies that "sufficiency," in the context of "sufficient margin to failure to rewet," refers to satisfying either of the two acceptance criteria listed above; in other words, the margin is sufficient if failure to rewet is avoided or if failure to rewet occurs and the PCT remains below 2200°F. In the proposed methodology, if the margin is sufficient based on these criteria, no further TTWB analyses are required, as discussed in Step 3.b. If the margin is insufficient, Step 3.c must be followed.

The NRC staff concludes that the definition, in the RAI-15 response, [

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Based on the NRC staff's concerns, as discussed above, that cycle-specific variations may potentially lead to large changes in PCT even if failure to rewet has occurred, the NRC staff has concluded that all transition cycles (i.e., up to two transition cycles) occurring during EFW operation must be addressed, either by explicit analysis as in Step 3.a or by appropriate justification, to provide adequate assurance that the most limiting cycle is analyzed. As such, a limitation and condition is being imposed to ensure that this is the case.

The RAI-15 response applies a [

], based on the NRC staff's experience this is expected to provide sufficient added conservatism to compensate for the possibility of the most limiting point not occurring precisely at one of the [ ] analyzed.

In the RAI-16 response, Framatome discussed that the process described in RAI-15 employs conservative assumptions with respect to the time of oscillation onset. [

] Framatome also discussed that the assumed FW temperature reduction rate in all cases must conservatively bound the expected plant-specific behavior. Furthermore, the definition of exposure points and transition cycles to be analyzed, as well as [ ] discussed above, provides additional assurance that the results remain bounding when considering cycle-specific and exposure-specific variations in oscillation onset time. The NRC staff finds this approach acceptable consistent with the discussions above, provided that the applicable limitations and conditions are met.

The NRC staff finds Step 3.c acceptable because the NRC staff will review the approach used for Step 3.c on a plant-specific basis.

Selection of the Limiting Event

Framatome proposed to perform Steps 3 through 3.c for the TTWB event, and – as discussed in Step 4 – 2RPT analyses are only required if TTWB did not experience failure to rewet. The rationale for this, as understood by the NRC staff, is as follows. The feedwater heaters remain active throughout the 2RPT event but not in the TTWB event, resulting in higher feedwater temperatures and therefore lower average power, less severe oscillations, and lower PCT values than in the TTWB event.

However, because the time-critical operator action is defined with respect to the time of identification of an ATWS, and because the 2RPT event involves manual initiation of scram by the operator if the RPV level does not increase to the high level turbine trip setpoint (compared

to automatic scram initiation in the TTWB event), plant-specific analyses must justify the time required for operators to initiation manual scram following the 2RPT and must add this time to the time-critical operator action time following identification of ATWS. This effective delay in operator actions relative to the initiating event allows additional time for the feedwater temperature to decrease during the 2RPT event, such that the feedwater temperature at the time of operator actions may potentially be less than that for the TTWB event. Furthermore, for plants with steam-driven feedwater pumps, the TTWB event may include a trip of the FW pumps prior to operator action, which may mitigate the consequences of the ATWS-I event (due to reduction in water level before action is taken). Consequently, the TTWB event may not be limiting for all plants.

Additionally, the TTWB event, due to turbine valve closure, leads to a higher system pressure than the 2RPT event. Based on the NRC staff's experience, competing effects may exist which lead to an unclear relationship between system pressure and stability behavior, which is difficult to ascertain *a priori*.

Because of these two concerns, the NRC staff concludes that plant-specific applications must justify the operator action time and feedwater temperature assumptions for both the TTWB and 2RPT events, and must perform analyses for both events using these assumptions to determine which event is limiting with respect to PCT. However, the NRC staff acknowledges that the limiting event is expected to be primarily a function of the operator action and feedwater temperature assumptions used in the analyses, which remain the same regardless of operating cycle or exposure conditions. Therefore, there is reasonable confidence that the limiting event – 2RPT or TTWB – will remain the same across all cycles and exposure points for a given plant-specific application.

As a result, the NRC staff requires that both TTWB and 2RPT analyses be performed initially for Step 3; and once the most limiting event is determined, only that event must be considered for the additional justifications for Step 3 or the additional analyses in Steps 3a through 3c. Plant-specific applications must justify the operator action and feedwater temperature assumptions for both events. The NRC staff also notes that due to the additional work necessary to justify cycle-independent application of the ATWS-I analyses, the value of providing the results of the non-limiting event to the NRC for review will be limited.

<u>Boron Injection</u>

In addition to taking action to reduce water level, the reactor operators must initiate SLCS boron injection within the time-critical action interval following identification of an ATWS. The effect of SLCS injection is to deliver borated water to the core which provides sufficient negative reactivity to shut the core down. This shutdown is capable of terminating any oscillations as well as limiting the impact of high core power on the containment heat load. However, the operator actions to reduce water level are expected to mitigate the oscillations well before the borated water reaches the core. **[**

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Calculation Procedure Conclusions

Based on the evaluations given above, the NRC staff finds the calculation procedure to be acceptable for its intended purpose, with four limitations and conditions as discussed in Section 5.0 of this SE.

4.2.9 Review of ANP-10346P Appendix A – Steady State Dryout Correlation CPROM

Dryout of a fuel rod, unless it is quickly followed by rewet, leads to very large increases in cladding temperature. Therefore, accurate calculation of the timing and location of dryout in the fuel bundles is of high importance in determining the PCT and has a strong effect on whether the ATWS acceptance criteria are met during postulated ATWS-I scenarios. The approach used to develop the models for dryout (and rewet) in ANP-10346P is twofold. [

] dryout/rewet model described in Section 5.5.5 of ANP-10346P and evaluated in Section 4.2.5.5 of this SE.

The CPROM correlation was previously presented in the Monticello ATWS-I methodology and was reviewed and accepted by the NRC staff for plant-specific application at Monticello. However, the correlation was developed from [

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- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

[

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With the exception of stagnant or reversed flow – which may occur during ATWS-I, especially near the inlet of the bundle – the range of operating conditions shown above encompasses the expected ATWS-I conditions for the current BWR fleet and is therefore suitable for generic use in BWR ATWS-I analyses. [

] For these reasons, the NRC staff finds the use of the CPROM correlation within the dryout/rewet model in ANP-10346P to be generically acceptable for ATWS-I analyses.

[

], to be acceptable.

Because of the wide range of operating conditions over which the ANP-10346P CPROM [ ] is validated, and because the modifications relative to the previously reviewed the Monticello ATWS-I CPROM [ ] are reasonable and result in comparable or improved accuracy relative to the measured data, the NRC staff finds the CPROM steady state correlation in ANP-10346P to be acceptable for use as the underpinning critical power correlation for use in the ANP-10346P [ ] model on a generic basis for limiting ATWS-I analyses.

4.3 Code Assessment

Following the review guidance provided in Chapter 15.0.2 of the SRP, the next area of review for transient and accident analysis methods focuses on assessment of the code. The associated acceptance criteria indicate that all models need to be assessed over the entire range of conditions encountered in the transient or accident scenarios. The review procedures provided in Section III of Chapter 15.0.2 of the SRP also indicate that the assessment of these models is commensurate with their importance and required fidelity. This assessment is generally performed via comparison of predicted results against both separate effects tests and integral effects tests.

Separate effects tests are generally used to demonstrate the adequacy of individual models and the closure relationships contained therein. Complementary to these types of tests are integral tests, which are generally used to demonstrate physical and code model interactions that are determined to be important for the full-size plant. The NRC staff evaluation of the individual elements of the code assessment suite provided in ANP-10346P are presented in the following subsections.

4.3.1 Review of TR Section 6.1 – Test Suite and Acceptance Criteria

The following acceptance criteria were proposed by Framatome for validation against measured data:

- Calculated void fraction [ ] of measured
- Calculated pressure drop [ ] percent of measured
- Calculated decay ratio [ ] of measured, with the exception that higher decay ratios above this range are considered acceptable
- Calculated oscillation frequency [ ] of measured
- [ ]
- Acceptance criteria for nonlinear benchmarks given on a case-specific basis

These acceptance criteria ranges for void fraction, pressure drop, decay ratio, and oscillation frequency correspond to the uncertainties determined in the prediction of these parameters in STAIF (Ref. 15) and the RAMONA5-FA LTSS methodology. As discussed in Section 4.4, ATWS analyses are not required to explicitly account for modeling uncertainties, as is required for design-basis event analyses such as those for which STAIF and the RAMONA5-FA LTSS methodology are used. However, the NRC staff finds the approach of defining validation acceptance criteria for ANP-10346P based on relevant uncertainty bounds for the previously-approved stability analysis methods STAIF and RAMONA5-FA to be acceptable because this demonstrates that the ANP-10346P methodology has similar or not significantly greater modeling uncertainty than the previously approved stability methodologies.

The NRC staff finds the added stipulation that calculated decay ratios are allowed to be more than 0.2 higher than measured to be acceptable for this application because higher decay ratios are conservative and [ ] in calculated-versus-measured decay ratio was seen in the benchmarking to KATHY stability tests (Section 4.3.4 of this SE), indicating good predictive capability of the ANP-10346P methodology across a wide range of conditions.

The suitability of the acceptance criteria for the pin-dependent CPROM term and for the nonlinear benchmarks is discussed as part of their own separate subsections later in this SE.

4.3.2 Review of TR Section 6.2 – Benchmarking to Void Fraction Tests

As described in Sections 4.2.3.4 and 4.2.3.11.6 of this SE, the ANP-10346P methodology uses the [ ] to determine the relationship between quality and void fraction in the fuel bundles. The [ ] was approved for use in the RAMONA5-FA LTSS methodology.

Section 6.2 of ANP-10346P states that the following steady-state void fraction data sets were used to validate this correlation:

- FRIGG (314 test points)
- ATRIUM-10 KATHY ([ ])
- ATRIUM 10XM KATHY ([ ])

These data include a wide range of pressure, inlet subcooling, and mass flow rate conditions. The ATRIUM-10 and ATRIUM 10XM data include a maximum void fractions of [ ] and [ ], respectively. These data are the same as were used to validate a different void fraction correlation in the Monticello ATWS-I methodology. However, all data collected for pressures outside the range of [ ] were discarded for the current application. The NRC staff finds this acceptable because this pressure range encompasses the expected pressure during postulated limiting ATWS-I events. Additionally, [ ].

However, the remaining data used for validation – especially the ATRIUM-10XM data, which covers the broadest range of conditions – provides very good coverage of the range of expected operating conditions during ATWS-I. The calculated void fraction demonstrates good agreement with the measured data, and includes benchmarking directly relevant to the specific geometric configuration of current fuel types (ATRIUM-10 and ATRIUM 10XM).

The [ ] As discussed in Section 4.2.3.7 of this SE, the dynamic transient behavior – especially under rapidly-changing conditions such as large amplitude oscillations – may cause significant departure of the relative phase velocity behavior from the behavior under steady-state conditions, and this behavior is not directly validated by the steady-state void fraction tests.

However, the linear and nonlinear stability tests discussed in later sections provide an integral validation of the overall code behavior, including the void fraction correlation. The calculated stability behavior (e.g., oscillation growth rate) is highly sensitive to the void fraction correlation due to its impact on the pressure drop response and density reactivity feedback; therefore, the close agreement of the ANP-10346P methodology with measured linear and nonlinear stability data under a wide range of conditions provides additional assurance that the void fraction correlation does not impose any significant nonconservative error trend or bias in the calculated results under expected oscillatory conditions during ATWS-I.

4.3.3 Review of TR Section 6.3 – Benchmarking to KATHY Pressure Drop Tests

Results from the RAMONA5-FA ATWS-I code for pressure drop were validated against the following steady state pressure drop measurements:

- KATHY ATRIUM-10
- KATHY ATRIUM 10XM

Benchmarking against pressure drop data allows for validation of the total pressure drop calculated in the ANP-10346P methodology under steady-state conditions. For single-phase flow, the total pressure drop depends directly on the single-phase friction factor (as well as the liquid density thermophysical correlation, which has low uncertainty). For two-phase flow, the total pressure drop depends primarily on the single-phase friction factor, two-phase friction

multiplier, and void-quality correlation (which determines the density and velocity of the fluid). Because the void-quality correlation was directly validated by the steady-state void fraction benchmarks and shown to give good agreement, the pressure drop tests are particularly useful in validating the single-phase friction factor and two-phase multiplier.

The measurements include a broad range of pressure, inlet temperature, and mass flow rate. A total of [ ] were included in the validation, covering single-phase and two-phase conditions. For ATRIUM-10, the mean relative error for the single-phase (two-phase) data points was [ ]. For ATRIUM 10XM, the single-phase (two-phase) mean relative error was [ ]. Both the single-phase and two-phase tests (for both fuel types) demonstrate close agreement between calculated and measured total pressure drop with no observable trends, and the NRC staff concludes that the ANP-10346P methodology is well-validated for calculating pressure drop over a wide range of operating conditions. As with the [ ] discussed above, the calculation of pressure drop is an important parameter to correctly determine thermal hydraulic stability, and the close agreement with measured linear and nonlinear stability data discussed below provides added assurance that the single-phase friction factor and two-phase multiplier and their implementation for oscillatory transient conditions is accurate and acceptable for ATWS-I applications.

4.3.4 Review of TR Section 6.4 – Benchmarking to KATHY Stability Tests

Results from the RAMONA5-FA ATWS-I code for single-assembly thermal hydraulic stability were compared to measured stability data in KATHY for the following fuel designs:

- ATRIUM-10 ([ ] test points)
- ATRIUM 10XM ([ ] test points)

These tests included stable conditions (decay ratio less than one) as well as unstable conditions (decay ratio greater than one). For the stable test points, the decay ratio and resonance frequency were determined from analysis of noise in the output signals using well-established numerical techniques. For the unstable points, the decay ratio and resonance frequency were determined from analysis of the coherent oscillation signals above noise level in the output data.

The benchmarking results show acceptable agreement between measured and calculated decay ratios, with the majority of the calculated decay ratios [ ]. Most of the points that have [ ] are near or above the stability boundary, and the RAMONA5-FA ATWS-I code [ ]. The mean error in calculated versus measured frequency is [ ], with very few points exhibiting error larger than [ ]. The comparison of calculated to measured decay ratio and frequency satisfies the acceptance criteria discussed in Section 4.3.1 of this SE and demonstrates the ability of the RAMONA5-FA ATWS-I methodology to accurately predict the channel thermal hydraulic stability behavior of ATRIUM fuel types. These tests provide an integral validation of the thermal hydraulic phenomena important for stability, including fluid mass, momentum, and energy transport as well as constitutive relations such as the void-quality, friction factor, and wall heat transfer coefficients in the subcooled and two-phase nucleate boiling regimes.

4.3.5 Review of TR Section 6.5 – Benchmarking to KATHY Dryout/rewet Tests

The RAMONA5-FA ATWS-I code was benchmarked against KATHY dryout/rewet stability tests for the following fuel types:

- ATRIUM-9 ([ ] test points)
- ATRIUM 10XM ([ ] test points)

These experiments were reviewed and evaluated in the SE for the Monticello ATWS-I methodology and were found to provide a realistic representation of the thermal hydraulic behavior, including the impact of neutronic feedback, during large amplitude oscillations characteristic of ATWS-I conditions up to and including failure to rewet. In particular, the inclusion of realistically simulated neutronic feedback allows significant inlet flow reversal to occur and promotes the occurrence of dryout at low elevations as expected for ATWS-I events.

The NRC staff examined Figures 6-6 through 6-20 in ANP-10346P and has concluded that the RAMONA5-FA ATWS-I methodology provides a reasonable and realistic agreement with the qualitative behavior of the KATHY dryout/rewet tests, including the onset and growth of oscillations, cyclic dryout and rewet, and eventual failure to rewet. Furthermore, for each test case, [

] As

discussed in Section 4.2.5.5 of this SE, [

]

[

]

The KATHY dryout/rewet experiments serve as an extension of the model validation described in the previous section and additionally provide validation of [

].

4.3.6 Review of TR Section 6.6 – Benchmarking to Linear Reactor Stability Benchmarks

Section 6.6 of ANP-10346P describes the benchmarking performed with the RAMONA5-FA ATWS-I code for linear reactor stability data for the following BWR plant events:

- [ ]
- [ ]
- [ ]
- [ ]

These linear reactor stability benchmarks were also included in benchmarking suites for the approved RAMONA5-FA LTSS and STAIF methodologies. These events involved measured oscillations with a decay ratio of approximately [

]. Therefore, similarly to the KATHY stability tests, these benchmarks provide an integral validation of the fluid mass, momentum, and energy transport as well as constitutive relations in terms of their impact on system stability. Specifically, these benchmarks are used to validate the prediction of the timing of stability onset and the oscillation growth rate, but not the prediction of dryout or rewet that could potentially occur during the later stages of postulated ATWS-I events.

However, compared to the KATHY single-assembly stability tests, the linear reactor stability benchmarks also provide validation of the effect of neutronic feedback – including the effect on wall-to-fluid heat transfer as a function of space and time – on the overall system oscillation characteristics.

Another key difference relative to the KATHY stability tests is that these benchmarks involve mixed cores with multiple fuel types other than ATRIUM 10 and ATRIUM 10XM. The NRC staff issued RAI-6 to obtain additional information regarding the linear reactor stability benchmarks, including information on operating conditions and fuel types for each benchmark case. In the response to RAI-6, Framatome provided the requested list of fuel types and conditions present in each of the benchmarked cores. Framatome indicated that the majority of fuel-specific inputs were available from the past benchmarking for the STAIF and RAMONA5-FA codes. In some cases, some input for the [

]. The NRC staff reviewed the information that was inferred and determined that this inference was performed in an acceptable manner and that any differences between the inferred and actual values for these fuel types would be expected to have a minor impact on the results. For these benchmarks, all neutronic and thermal

hydraulic data were taken directly from the benchmarking suites for the approved STAIF and RAMONA5-FA codes, with no additional neutronic or thermal hydraulic data required.

Also in the response to RAI-6, Framatome provided a table of operating conditions and power distribution information for each of the linear reactor benchmarks. The benchmarks consist of stability tests performed at off-rated conditions during reactor startup ([ ]) or a stability event from off-rated conditions ([ ]). Although these operating conditions may not strictly correspond to the conditions that would occur during a 2RPT or TTWB ATWS event at each plant, the tests encompass a reasonably wide range of power, flow rate, inlet subcooling, outlet quality, and axial power shapes (including highly bottom-peaked axial power profiles) similar to those expected during the initial oscillation growth phase of ATWS-I events. The oscillation decay ratios and frequencies calculated by the RAMONA5-FA ATWS-I code were within the acceptance criteria listed in Section 4.3.1 of this SE for all [ ] linear benchmark cases, which included both regional and core wide oscillation modes, and the results showed no discernible trend with respect to operating conditions, fuel types, or other plant-specific differences that exist among the four benchmark cases. Therefore, the NRC staff concludes that the linear stability benchmarks provide good additional assurance, beyond the single-assembly KATHY stability benchmarking, that the code is able to predict the onset and initial growth rate of oscillations that would occur during postulated ATWS-I events on a generic basis.

4.3.7 Review of TR Section 6.7 – Benchmarking to Non-Linear Reactor Benchmarks

Section 6.7 of ANP-10346P describes the benchmarking performed with the RAMONA5-FA ATWS-I code for the following two nonlinear stability events:

- Oskarshamn turbine trip with non-linear oscillation
- BWR A feedwater temperature transient with non-linear oscillation

Unlike the linear reactor benchmarks, the nonlinear reactor benchmarks provide direct validation of the system response during events leading up to reactor instability, as well as validation of the onset timing and growth of oscillations up to relatively large amplitude (approximately [ ] in both benchmarks) before oscillation suppression via scram.

Limited details were given in ANP-10346P for the boundary conditions and assumptions used in the RAMONA5-FA ATWS-I methodology for these benchmarks. To assist the NRC staff in determining whether the events were analyzed in an acceptable manner, the NRC staff requested further details in RAI-7 on the fuel types, fuel-specific data, boundary conditions, and other assumptions used for these two cases. In the RAI response, Framatome indicated that [ ]. However, as was the case for the linear reactor benchmarks, some input data for [ ] were not directly available for all fuel in the core, and in these cases the additional input data were inferred by comparisons to similar [ ]. The NRC staff finds this acceptable for the same reasons as stated in the previous section.

Additionally, in the response to RAI-7, Framatome provided further information on the boundary conditions used in both models. For the Oskarshamn-2 benchmark, most initial conditions and

boundary conditions were taken from the OECD/NRC Oskarshamn-2 BWR Stability Benchmark specifications (Ref. 16). Notably, this included the feedwater temperature time-dependent behavior provided in the benchmark specifications, in which the feedwater temperature was assumed to decrease earlier than the measured value to account for the heat-conduction-related time delay between the actual and measured feedwater temperature during the event. Feedwater flow rate was decreased from the measured data to ensure reasonable water level calculation in the RAMONA5-FA ATWS-I code, and two sets of runs were made for pump speed: one run using the measured pump speed versus time and a second run using a modified pump speed to more closely match the measured core flow rate. The NRC staff reviewed these assumptions and found them to constitute a reasonable and acceptable representation of the Oskarshamn-2 instability event.

[

]

For the BWR-A benchmark, [

]

Results for the two benchmarks are shown in Figures 6-24 through 6-30 of ANP-10346P. In the Oskarshamn-2 case, the calculated core-average power matches the measured core-average power closely up to the point of instability, and the calculated oscillation onset time and oscillation frequency appear to match the measured values closely while the calculated oscillation growth rate appears to be noticeably larger than measured, which is conservative. The two different assumptions used for pump speed affected the oscillation growth rate, but this growth rate was higher than measured in both cases.

[

]

Therefore, the NRC staff concludes that both nonlinear reactor benchmark cases demonstrate the accurate or possibly conservative prediction of oscillation onset time and growth rate during measured BWR instability events with relatively large oscillation amplitude.

4.4 Uncertainty Analysis

As opposed to analyses for design basis events, which should explicitly account for modeling uncertainties to ensure that the safety criteria are met, ATWS analyses may use best-estimate or reasonably bounding modeling approaches to demonstrate acceptable consequences to the public under limiting ATWS events. No explicit requirement or guidance is given for analyzing uncertainties in the calculated results for these events. The rationale for this is that ATWS events, which are beyond design basis events, have very low probabilities of occurrence compared to design basis events.

Although ATWS analyses may use a best-estimate approach, some understanding of the impact of variations in specific parameters or modeling assumptions in relation to satisfying the ATWS acceptance criteria is important, in order to properly evaluate the models and assist in determining acceptable input and modeling requirements for the given application. Framatome provided a PIRT in Section 4.15 of the TR to assist in this process. As discussed in Section 4.1 of this SE, Framatome ranked various phenomena by their importance for three figures of merit – oscillation inception, limit cycle amplitude, and post-dryout – which affect the ATWS-I event progression in different ways and contribute to the overall PCT in a given calculation. The NRC staff performed its review of the RAMONA5-FA ATWS-I methodology partly based on this PIRT, as a means of focusing the review preferentially on phenomena and corresponding models with higher importance. For example, the NRC staff issued RAI-4 to obtain more information on the validation of the gap model and RAI-11 to request sensitivity studies by adjusting the gap conductance values, in part because the [ ] were dispositioned as parameters of high or medium importance for the oscillation inception and post-dryout figures of merit. Additionally, although the gap model was applicable to the linear and nonlinear core benchmarks, it was not applicable to [ ]. Therefore, numerical sensitivity analyses were particularly useful for this model to understand the model's impact on stability calculations. The NRC staff's evaluation of these RAIs is presented in Section 4.2.2.6 of this SE.

Several other highly-ranked phenomena such as total core power, total core flow, feedwater temperature, core size, and core design are determined or justified uniquely for each plant-specific application and therefore were not subjected to sensitivity studies.

Additional highly-ranked phenomena such as [ ] are related to processes involving fluid transport, heat transfer, and neutronic coupling, which impact the stability behavior of the system. The models used to determine [ ]. Their behavior under transient conditions – specifically, their impact on stability – was validated in an indirect manner through their impact on the KATHY and full-core stability benchmarking, which provide an integral validation of the stability-related dynamic processes as discussed in Sections 4.3.4-4.3.7 of this SE. Due to the extensive benchmarking of the stability predictions of the code under a wide range of operating conditions, which provides confidence that the relevant phenomena are calculated accurately, additional sensitivity calculations were not requested by the NRC staff for these models.

However, in RAI-8 and RAI-9, the NRC staff did request justification that the core and vessel nodalization were sufficient to provide reasonable and accurate prediction of PCT during

ATWS-I events. The discretization used in the numerical solution of the models impacts the transport of mass, momentum, and energy in the system – in particular, by impacting numerical diffusion – and therefore it may have high importance on the calculated stability behavior. Discussion and evaluation of the response to RAI-8 and RAI-9 is given in Sections 4.2.3.12 and 4.2.3.1 of this SE, respectively.

Additional sensitivity studies were provided by Framatome in ANP-10346P, including sensitivities on [ ]

[ ] The NRC staff's experience shows that the rod node associated with initial failure-to-rewet is not necessarily the hottest (highest peaking factor) node in the core, and additional, higher-power nodes which fail to rewet later in the event may cause large, rapid increases in PCT, potentially on the order of hundreds of degrees.

In addition to possible PCT increases associated with changes to the limiting PCT node location, the PCT at a given limiting node location is also expected to increase with increasing core inlet subcooling because this increases the core average power level. This behavior was observed in the time-dependent results for the sample problem provided in ANP-10346P as well as the additional sensitivity results provided in the responses to RAI-8 through RAI-12. In these results, the PCT following failure to rewet appears to increase and decrease in tandem with the core inlet subcooling.

[ ]

The determination of core inlet subcooling is not straightforward and depends on the code's ability to accurately model the mixing of injected feedwater from the vessel feedwater spargers into the vessel downcomer liquid – or vapor, if the water level is low enough. Additionally, the code must accurately model the mass and energy transport of this fluid through the vessel downcomer and lower plenum in order to properly determine the core inlet subcooling as a function of time. This calculation further depends on the code's ability to accurately model the feedwater flow rate, which responds to changes in the steam flow rate exiting the vessel and is characterized by a time delay based on the balance-of-plant dynamics. As a result of these dynamic effects, the core inlet – and, relatedly, the PCT – may continue to increase for a significant length of time ([ ])

after operator actions are performed, such that the timing and magnitude of peak PCT is determined by competing dynamic effects associated with feedwater flow rate, water level, and the mass and energy transport of fluid through the vessel.

Based on the sensitivity results provided in ANP-10346P as well as in the responses to RAI-8 through RAI-12, the discretization assumptions – including vessel nodalization, core nodalization, and timestep size – as well as core modeling assumptions such as gap conductance appear to have a **[** **]** effect on the core inlet temperature response for the Brunswick sample problem. These results highlight the importance of accurately modeling the vessel and recirculation loop as well as the balance of plant dynamics on a plant-specific basis, as these are expected to be the primary determinants of the core inlet temperature response during ATWS-I events. Modeling assumptions which impact the core response, such as core nodalization and gap conductance, may have some effect on core inlet temperature, but the more important effect of these parameters appears to be in impacting the stability behavior of the core itself and the timing of oscillation growth with respect to operator actions. The NRC staff expects that this conclusion likely also holds for cycle-specific changes because such changes would primarily impact the core behavior – via, for example, changes in the radial and axial power distribution – while the recirculation loop and balance of plant dynamics would typically not change between cycles. As discussed in Section 4.2.8 of this SE, any changes to the plant-specific configuration other than cycle-specific changes to the fuel in the core will require an evaluation to ensure that the ATWS-I analyses remain bounding, or reanalysis of the ATWS-I event with the updated plant configuration. In addition, ATWS-I analyses must be justified to reasonably bound the behavior for future cycles when considering possible changes in oscillation onset timing and mode behavior which are expected to be caused primarily by differences in core fuel loading and operational changes even when the ex-core plant configuration remains unchanged.

Even though the NRC guidance for beyond design basis accidents such as the ATWS-I event does not require uncertainties to be accounted for within the analysis conclusions, Framatome provided some sensitivity analyses to demonstrate the relative sensitivity of the ATWS-I results to specific parameters that were not explicitly evaluated through code validation. As discussed above, most of the sensitivities were relatively modest, except for **[** **]**. The NRC staff found that the guidance provided in ANP-10346P, as supplemented in the RAI responses and augmented by the limitations and conditions listed in Section 5.0 of this SE, was adequate to ensure that the important sensitivities are adequately addressed for each application of this methodology.

5.0 LIMITATIONS AND CONDITIONS

As discussed previously in this report, the following conditions and limitations have been applied to NRC approval for use of the RAMONA5-FA methodology to analyze the ATWS-I event as described in ANP-10346P. Table 8-1 of ANP-10346P contains additional limitations and conditions imposed on the applicability of the RAMONA5-FA ATWS-I methodology by Framatome; these limitations and conditions are considered to be part of the proposed methodology and are not included below.

1. The gap conductance sensitivity shall be repeated or otherwise justified for transitions to new fuel designs.

2. If the acceptance criteria for the first paragraph in Step 3 of Section 8.0 of the TR are met, additional justification must still be provided to demonstrate adequate margin in operator action timing for variations in neutron kinetics response from specific core designs. This justification may be provided by following Steps 3.a through 3.c, as amended by the response to RAI 15, or providing an alternative justification on a plant-specific basis.

3. Plant-specific evaluations that are intended to be bounding of all core designs must be confirmed to provide reasonable assurance that neutron kinetics characteristics such as possible differences in dominant oscillation modes or the potential for multiple oscillation modes to be active simultaneously are bounded by the analysis of record.

4. Due to the unique neutron kinetics characteristics associated with transition cycles, all transition cycles must be dispositioned in a manner consistent with Limitations and Conditions #2 and #3.

5. The ATWS-I analysis must be performed for both the TTWB and 2RPT events during the initial implementation of this methodology, to confirm which event is limiting. Subsequent evaluations may only consider the event determined to be limiting, except when changes are made to the plant design or operation that may affect stability behavior during ATWS, such as: turbine bypass capability, fraction of steam-driven feedwater pumps, and changes expected to significantly increase core inlet subcooling during ATWS events.

6. The steam line and valve modeling options shall be confirmed to accurately capture the expected plant-specific system performance during ATWS-I events.

7. Plant-specific applications must justify that the selected settings and modeling options are appropriate, including core and vessel nodalization, time step control parameters, and noise parameters. In particular, the modeling should be reasonably consistent with both the characteristics of the plant in question and the validation basis for the RAMONA5-FA ATWS-I methodology as discussed in this SE.

6.0 CONCLUSIONS

In ANP-10346P, Framatome presented a proposed methodology to analyze the ATWS-I event using the RAMONA5-FA code. The following conclusions are provided here in summary as they apply to BWR/3-6 submittals.

ANP-10346P presents a description of the ATWS-I event, the relevant phenomena, the applicable Figures of Merit (FoMs), and a ranking of the phenomena for any applicable FoMs. This information was reviewed and compared to similar information available to the NRC staff (e.g., Ref. 9) and confirmed to be consistent with previous approvals of ATWS-I or other stability related methodologies.

The application of the RAMONA5-FA code for the purpose of analyzing ATWS-I events involved the incorporation of several new models in the RAMONA5-FA code relative to what the NRC staff has previously reviewed and approved for LTSS analyses. Many of these models had been reviewed and approved by the NRC staff as part of a plant-specific ATWS-I methodology adopted at Monticello. The NRC reviewed the previously approved RAMONA5-FA models, the

previously approved models from the Monticello ATWS-I application, and new models developed specifically for the purpose of the ANP-10346P methodology. The NRC staff confirmed that the previously approved models and new models are applicable to analysis of the ATWS-I event.

ANP-10346P also presents a procedure for analysis of the ATWS-I event, which [ ]. Since the intent of the proposed ATWS-I analysis methodology is to perform a single evaluation upon initial implementation at a specific plant without subsequent confirmatory analyses on a cycle-specific basis, the NRC staff carefully considered how different characteristics of future cycles might affect the results of a cycle-independent evaluation. In addition to changes in fuel assembly designs (including transition core designs), the NRC staff considered whether cycle or plant configuration changes might affect the limiting PCT or the margin to operator action timing. As a result of sensitivities of the coupled neutronic/thermal-hydraulic feedback to cycle-specific variations in the core neutronic or plant system response, conditions (1) through (5) were found to be necessary to ensure that a plant-specific analysis will bound future core designs and changes to relevant plant system parameters. Additionally, conditions (6) and (7) were identified to ensure that the plant-specific models used for analysis of the ATWS-I event are consistent with the underlying validation and assessment of the methodology as described in ANP-10346P and the RAI responses.

In order to demonstrate the capability of the RAMONA5-FA code to analyze the ATWS-I event, assessments were made against separate effects tests and integral benchmarks. Separate effects tests helped validate the RAMONA5-FA code for prediction of parameters important to the ATWS-I event, such as void fraction, pressure drop, single channel stability characteristics, and dryout/rewetting response during large amplitude oscillations. The integral benchmarks provided confidence in the RAMONA5-FA code's ability to model full scale stability events. In some cases, sensitivity studies were used to demonstrate that the RAMONA5-FA ATWS-I methodology was either conservative or insensitive to variations in specific parameters. This provided assurance that relevant uncertainties in the ATWS-I analysis methodology and model parameters would not change the conclusions of an ATWS-I evaluation done in accordance with ANP-10346P. Based on a general review of the tests, benchmarks, and sensitivity studies, the NRC staff determined that the methodology was appropriately confirmed to yield acceptable predictions for all parameters and phenomena important to the ATWS-I event, provided that the limitations and conditions are met.

In summary, the NRC staff finds that the assessment of the RAMONA5-FA code, as described in ANP-10346P and responses to NRC staff RAIs, adequately demonstrates that RAMONA5-FA is suitable to analyze the ATWS-I event by demonstrating acceptable performance in each of the highly ranked phenomena. In addition, the NRC staff finds that the procedure described in ANP-10346P for performance of the ATWS-I analyses provides appropriate guidance to perform ATWS-I analyses that will bound cycle-specific variations at a given plant, subject to limitations and conditions. As such, NRC staff approval of the ANP-10346P methodology for analysis of the ATWS-I event is contingent on adherence to the conditions and limitations set forth in Section 5.0.

7.0 <u>REFERENCES</u>

1. Letter from Gary Peters, Director, Licensing & Regulatory Affairs, AREVA, Inc., to USNRC Document Control Desk, "Request for Review and Approval of ANP-10346P, Revision 0,

'ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA,'" dated December 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17355A231).

2. AREVA NP Inc. Topical Report ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," December 2017 (ADAMS Accession No. ML17355A233 (Non-Proprietary)/ML17355A235 (Proprietary)).

3. AREVA NP Inc. Licensing Topical Report EMF 3028P-A, Volume 2, Revision 4, "RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain: Theory Manual" (ADAMS Accession No. ML131550602 (Proprietary)).

4. AREVA NP Inc. Report ANP-3274P, Revision 2, "Analytical Methods for Monticello ATWS-I," July 2016 (ADAMS Accession No. ML16221A275 (Non-Proprietary)/ML16221A278 (Proprietary); Approved via License Amendment in ADAMS Accession No. ML17054C394).

5. Letter from Gary Peters, Director, Licensing & Regulatory Affairs, Framatome, Inc., to USNRC Document Control Desk, "Response to Request for Additional Information Regarding ANP-10346P, Revision 0, 'ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA,'" dated March 8, 2019 (ADAMS Accession No. ML19071A274).

6. Letter from Jonathan Rowley, Project Manager, Licensing Processes Branch, Division of Policy and Rulemaking, USNRC, to Gary Peters, Director, Licensing & Regulatory Affairs, Framatome, Inc., "Audit Report for the June 11-13, 2018, Audits in Support of the Review of ANP-10346P, Revision 0, 'ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA' (EPID: L-2017-TOP-0067)," April 25, 2019 (ADAMS Accession No. ML18249A225).

7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition," Chapter 15.8, "Anticipated Transients Without Scram," Revision 2 (ADAMS Accession No. ML070570008).

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition," Chapter 15.0.2, "Review of Transient and Accident Analysis Methods," Revision 0 (ADAMS Accession No. ML070820123).

9. NUREG/CR-6743, "Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel," USNRC, September 2001 (ADAMS Accession No. ML012850300).

10. NUREG/CR-7179, "BWR Anticipated Transients Without Scram in the MELLLA+ Expanded Operating Domain, Part 1: Model Development and Events Leading to Instability," USNRC, June 2015 (ADAMS Accession No. ML15169B064).

11. NUREG/CR-7180, "BWR Anticipated Transients Without Scram in the MELLLA+ Expanded Operating Domain, Part 2: Sensitivity Studies for Events Leading to Instability," USNRC, June 2015 (ADAMS Accession No. ML15169A168).

12. Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053500170).

13. AREVA NP Inc. Licensing Topical Report BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," February 2008 (ADAMS Accession No. ML081340208 (Non-Proprietary)/ML081340383 and ML081340385 (Proprietary)).

14. GE-Hitachi Nuclear Energy Licensing Topical Report NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," June 2009 (ADAMS Accession No. ML091800530).

15. Siemens Power Corporation Licensing Topical Report EMF-CC-074(P)(A), Volume 4, Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," August 2000 (ADAMS Accession No. ML090750216 (Proprietary)).

16. Kozlowski, T., Gajev, I., Lefvert, T., Christer, N., Roshan, S., et al., "BWR Stability Event Benchmark based on Oskarshamn-2 1999 Feedwater Transient," Royal Institute of Technology Stockholm, Sweden, OECD Nuclear Energy Agency, 2013.

17. Yarsky, P., "Applicability of TRACE/PARCS to MELLLA+ BWR ATWS Analyses – Revision 1," USNRC Office of Nuclear Regulatory Research, November 18, 2011 (ADAMS Accession No. ML113350073).

18. Framatome Inc. Licensing Topical Report ANP-10340P-A, Revision 0, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," May 2018 (ADAMS Accession No. ML18171A119 (Non-Proprietary)/ML18171A120 (Proprietary)).

Attachment: Resolution of Comments

Principal Contributors: Scott Krepel, NRR/DSS/SNPB
Aaron Wysocki, Oak Ridge National Laboratory

Date: October 30, 2019.

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT ANP-10346P, REVISION 0, “ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA”

FRAMATOME, INC.

PROJECT NO. 728/DOCKET NO. 99902041

This attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by Framatome Inc. (Framatome) on the draft safety evaluation (SE) for Topical Report (TR) ANP-10346P, Revision 0, “ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA.” Framatome provided the comments by letter dated June 20, 2019 (Agencywide Documents Access and Management System Accession No. ML19175A122).

Change Number	Page Number	Line Number/s	Comment	Resolution
1.	24	16	Change “more” to “less”, less conservative means lower PCTs	NRC staff agrees, and change has been incorporated as-is.
2.	29	26-27	Change “[ ].” to “[ ].”	NRC staff agrees, and change has been incorporated as-is.
3.	46	41	Add the unit of Hz to the frequency acceptance criteria.	NRC staff agrees, and change has been incorporated as-is.
4.	15	35	A statement should be added at the end of Section 4.2.2.5 to clarify and provide acknowledgement of the acceptance and approval with Cr-doped fuel, which was described in Appendix A of the RAI responses (ANP-10346Q1P).	A paragraph was added at the end of Section 4.2.2.5 to explicitly acknowledge the submission and approval of Appendix D to ANP-10346P.
5.	Throughout	-	Information that should be marked as Proprietary is highlighted in Yellow.	NRC staff agrees, and change has been incorporated as-is.

0414-12-F04 (Rev. 001, 03/10/2016)

AREVA

ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA

ANP-10346NP
Revision 0

Topical Report

December 2017

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Tables

Figures

Nomenclature

Acronym	Definition
2RPT	Two Recirculation Pump Trip
ANS	American Nuclear Society
ATWS	Anticipated Transient Without Scram
ATWS-I	Anticipated Transient Without Scram With Instability
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CPR	Critical Power Ratio
CPROM	Critical Power Reduced Order Model
DC	Downcomer
DNB	Departure from Nucleate Boiling
DIVOM	Delta over Initial Versus Oscillation Magnitude
EFW	Extended Flow Window
EM	Evaluation Model
EOC	End of Cycle
EOI	Emergency Operating Instructions
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
FoM	Figure of Merit
GDC	General Design Criteria
HPCI	High Pressure Coolant Injection
HTC	Heat Transfer Coefficient
KATHY	Karlstein thermal-hydraulic loop test facility
LCA	Limit Cycle Amplitude
LHGR	Linear Heat Generation Rate
LP	Lower Plenum
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio

MELLLA	Maximum Extended Load Line Limit Analylsis
MSIV	Main Steam Isolation Valve
OI	Oscillation Inception
PBDA	Period Based Detection Algorithm
PCT	Peak Clad Temperature
PDO	Post Dryout
PHE	Peak Hot Excess
PIRT	Phenomena Identification and Ranking Table
PPS	Plant Protection System
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
SLC	Standby Liquid Control
SP	Standpipe
SRP	Standard Review Plan
SRV	Safety/Relief Valve
SS	Steam Separators
TR	Topical Report
TTWB	Turbine Trip With Bypass
UP	Upper Plenum
USNRC	United States Nuclear Regulatory Commission

Abstract

This report presents a generic methodology for licensing the Extended Flow Window (EFW) operation of BWR3 to BWR6 plants with regard to Anticipated Transient without Scram with Instability (ATWS-I). The method aims at addressing the fuel specific impacts of the ATWS-I event. The methodology presented in this report utilizes the RAMONA5-FA computer code. The presented version of RAMONA5-FA has been modified to be capable of simulating the severe power and flow oscillations associated with core instabilities unsuppressed with scram. The code applies advanced models for post-dryout heat transfer for the calculation of the cladding temperature excursion. These post-dryout models are based on, and benchmarked against, data obtained from the Karlstein thermal-hydraulic loop (KATHY) test facility where a full scale electrically heated ATRIUM 10XM bundle was tested under realistic ATWS-I conditions of severe unstable density waves with simulated reactivity and power feedback.

The RAMONA5-FA code has been benchmarked against both single bundle KATHY measurements as well as regional and global oscillations in actual BWR plants. The KATHY benchmarks include pressure drop, void fraction, and stability measurements for several bundle designs. The plant stability benchmarks include benchmarks in both the linear and nonlinear instability domains.

In addition to benchmarking, sample ATWS-I transients, along with selected sensitivities are presented to better understand the event. Finally, a calculation procedure is described to define the process and conservatisms to be used in licensing calculations.

1.0 INTRODUCTION AND SCOPE

This document presents a description of BWR instability transients that are not terminated by scram, and thus power and flow oscillations grow to large amplitudes (see References 1 and 2). This class of transients is referred to as Anticipated Transients Without Scram with Instability (ATWS-I). This report documents a RAMONA5-FA based method for evaluating the fuel-related aspects of these events.

The methodology complies with the USNRC requirements of Section 15.8 of NUREG-800, Reference 3. The method is applicable to the limiting ATWS-I events covering the event from the initiation through inception of oscillations and final suppression through operator action.

Section 2 presents a summary of the method and its intended application. Section 3 identifies the regulatory requirements, as well as the potential limiting ATWS-I transients. Section 4 is devoted to a detailed qualitative discussion of the physical phenomena pertinent to the ATWS-I evaluation. With this background, the development campaign to prepare the transient system code, RAMONA5-FA, for the task of calculating ATWS-I transients is put in perspective. The modifications to the approved RAMONA5-FA code (Reference 22) are presented in Section 5. The validation of the code capabilities with regard to reactor oscillations as well as transient hydraulic test loop measurements is presented in Section 6. Having demonstrated the capability of the code to calculate an ATWS-I transient, Section 7 presents the sensitivity to important phenomena to demonstrate that the methodology is robust. The remaining element of the methodology is the ATWS-I licensing basis procedure for demonstrating that the ATWS/Stability Mitigation Actions are effective in maintaining core coolability described in Section 8.

Appendices present special topics; Appendix A is devoted to presenting a step by step derivation of the new critical power reduced order model (CPROM) formulation, Appendix B presents the special topic of [

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2.0 SUMMARY

A new generic methodology for the evaluation of Anticipated Transients Without Scram with Instability (ATWS-I) is presented. This report documents a RAMONA5-FA based method for evaluating the fuel specific portion of the event. This method is intended to cover the initial ATWS-I event through the time that operator actions suppress core oscillations for BWR3 through BWR6 plants. This method is not intended for use in evaluating containment effects or over pressurization during the event.

A key improvement of the RAMONA5-FA code is the addition of [

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RAMONA5-FA has been benchmarked against both linear and non-linear stability transients. These benchmarks include hydraulic-only oscillations at the KATHY test facility as well as full core reactor transients. These benchmarks demonstrate that RAMONA5-FA can adequately predict oscillatory behavior.

A sample ATWS-I application problem based on the Brunswick BWR4 reactor has been presented. Various sensitivities have been performed to examine the sensitivity to important parameters and to help validate the conclusions of the PIRT.

3.0 REGULATORY REQUIREMENTS

The regulatory requirements for the ATWS-I event are defined in 10 CFR 50.62. Specifically, 10 CFR 50.62 lays out the required design features and the requirements for the assessment of their adequacy. Additional regulatory requirements that could be affected by fuel design for these events are compliance with General Design Criteria (GDC) 12 and 35 in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Specifically:

> "GDC 12: 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 35: A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."

While NUREG-0800 Standard Review Plan (SRP) Chapter 15, Section 15.8 (Reference 3) also identifies GDC 14, 16, 38, and 50 as applicable to an ATWS-I event. These general design criteria define requirements for the vessel and containment and [ ] and are therefore not addressed in this report.

The ATWS-I event was originally addressed generically for operation within the MELLLA domain (Reference 1), SRP 15.8 states:

> "BWRs that implement both extended power uprate (EPU) and expanded power-flow domains (e.g., MELLLA+), the licensee will demonstrate that the ATWS/Stability Mitigation Actions are effective in maintaining core coolability."

The modification of the RAMONA5-FA evaluation model (EM), that was originally developed and approved for application to DIVOM linear instabilities in Reference 7, has been extended to be the transient simulator that will be used to assess the fuel specific impact of the mitigation actions on the ATWS-I event. This methodology is not intended to cover system or containment impacts of the ATWS-I event.

3.1 *Compliance with Standard Review Plan, Chapter 15, Section 15.0.2*

NUREG-0800 Chapter 15, Section 15.0.2 provides guidance to the Nuclear Regulatory Commission (USNRC) reviewer with respect to reviewing the technical contents of a submittal, typically a Topical Report (TR) pertinent to transient and accident analysis methodology. To accomplish this task, the USNRC provided the reviewer a list of criteria in Chapter 15.0.2 of the SRP as a means of arriving at a sound technical conclusion. The areas of considerations listed in Chapter 15.0.2 are provided below:

Item 1. Documentation. This document satisfies this requirement.

Item 2. Evaluation Model. Refer to Section 5.0.

Item 3. Accident Scenario Identification Process. Refer to Section 3.2.

Item 4. Code Assessment Database. Refer to Section 3.2.

Item 5. Uncertainty Analysis. Refer to Section 8.0.

Item 6. Quality Assurance (QA) Plan. Refer to Reference 4. Reference 4 is the basis for quality assurance in both the development and application of RAMONA5-FA for ATWS-I.

3.2 *ATWS-I Transient Scenarios*

Two events are considered as potentially limiting. Both events result in a trip of the recirculation pumps. This trip reduces core flow to natural circulation while core power remains high which can quickly put the reactor deep within the instability region. These two events are (1) Turbine Trip With Bypass (TTWB), and (2) Two Recirculation Pump Trip (2RPT).

3.2.1 Turbine Trip

The first event, the TTWB, is initiated by a turbine trip. The recirculation pumps are tripped either by the turbine stop valve closure, or an ATWS high pressure pump trip. Core flow drops rapidly to natural circulation which drives core power lower. As the core settles at natural circulation, the feedwater temperature begins to decrease due to the loss of extraction steam. The feedwater temperature decrease causes core power to rise. If no action is taken, then core power will rise to the point that it crosses the instability boundary. Without the ability to scram, and if mitigating operator action is sufficiently delayed, oscillations will begin to grow and will reach large magnitudes, accompanied by reverse flow at the inlet of the hot channels. Once the oscillations reach sufficient magnitude, cyclical dryout and rewetting of the cladding surface will begin. If the oscillations continue to grow, they can reach sufficient magnitude that rewetting is no longer possible and large excursions of clad temperature that could potentially challenge acceptance criteria will occur.

In order to prevent large amplitude oscillations plant Emergency Operating Instructions (EOI) or Emergency Operating Procedures (EOP) instruct the operators to reduce water level upon recognition of an ATWS scenario. This reduction in water level has two primary effects. The first is to reduce core flow by reducing the density head on the downcomer side of the loop. The resulting reduction in core flow also serves to reduce core power. In addition, the water level is reduced to a level that is below the elevation of the feedwater spargers. At this point in the transient scenario, the cold feedwater is spraying through a steam environment and falling a significant distance until it reaches the water level. Passing through the steam environment heats the feedwater to the point that it approaches saturation. This significantly reduces the core inlet subcooling, which not only directly adds a stabilizing force to the system, but also reduces core power further stabilizing the system. The suppression of oscillations and reduction in power terminate any temperature excursion in the fuel and ends the fuel-specific portion of the ATWS-I event.

The EOI/EOP also specify that operators initiate boron injection upon recognition of an ATWS condition. Boron injection normally happens after water level reduction due to system delays in the Standby Liquid Control (SLC) system. The boron injection serves to shutdown core power and terminates the ATWS-I event. As this action occurs after the limiting fuel portion of the event, this operator action will not be simulated.

3.2.2 Two Recirculation Pump Trip

The two recirculation pump trip event (2RPT) evolves similarly to the TTWB event with two exceptions, the event initiation and the feedwater temperature excursion. Since the 2RPT event does not involve a turbine isolation, the turbine remains online and extraction steam to the feedwater heaters is maintained so the feedwater temperature remains significantly higher than the TTWB event. Because of this, the power excursion in the 2RPT event is mild which results in a less severe event for the same operator intervention times. One difference between the two transient scenarios is that the TTWB event should generate an automatic scram very early in the event leading to earlier identification of the ATWS scenario by the operators. There may not be a scram signal at the initiation of the 2RPT which means the event may be allowed to progress further due to a delayed identification of ATWS by the operator. As such, the 2RPT may become limiting if operator action in the TTWB event occurs fast enough to suppress oscillations prior to dryout.

4.0 PHYSICAL PHENOMENA

In this section, a review of the basic physical phenomena that occurs during an ATWS-I event is presented. These can be divided into the phenomena leading to power and flow oscillations, and the dryout phenomena under these oscillatory conditions. [

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4.1 *Unstable Oscillations in General*

A feedback system is unconditionally unstable in the case of positive feedback, i.e., a perturbation in a system parameter results in enforcing the perturbation. This kind of divergence is not oscillatory. In systems with negative feedback, i.e., a perturbation in a system parameter results in reducing the perturbation may or may not be stable depending on other system characteristics. Immediate negative feedback makes the system unconditionally stable. On the other hand, delayed negative feedback may render the system unstable if the magnitude of the feedback is sufficiently large. In the case of strong delayed negative feedback, the corrective effect of the feedback overshoots the original perturbation and the system undergoes oscillations of exponentially increasing magnitude. This type of oscillation is possible in BWRs.

The simple description of the feedback effects outlined above applies to linear systems, or nonlinear systems that behave as a linear system when the oscillation magnitude is sufficiently small. As the oscillation magnitude grows, the magnitude of the feedback is no longer proportional to the original perturbation due to the nonlinear effects. The nonlinear effects can be stabilizing, and in this case an initially exponentially growing oscillation will grow at a slower rate as the oscillation magnitude increases, and finally reach a stable limit cycle. Nonlinear effects may also act in the opposite direction, and an initially exponentially growing oscillation will accelerate its growth rate further as the oscillation magnitude increases. Normally in a complex nonlinear system, like a BWR, there are regions of different nonlinear effects. [

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4.2 *Density Waves in a Boiling Channel*

The mechanism capable of inducing a strong and delayed negative feedback in a boiling channel is the propagation of density waves (See Reference 5 for a comprehensive review). The kinematic description of density waves can be best described in the idealized boiling channel where the rate and axial distribution of the heat source remain invariant, and the pressure drop between the inlet and exit of the channel is kept constant. A perturbation of the inlet mass flow rate travels up the channel and its magnitude changes and phase lag increases. The mass flow rate wave generates a corresponding change in the steam quality and void fraction and equivalently the mixture density. The single-phase and two-phase friction components will also respond to the perturbation in the mass flow rate and the resulting steam quality response. In a slow (quasi-steady state) perturbation, the net resulting feedback is negative, that is for a positive inlet mass flow perturbation, the average void fraction decreases lowering the density head that drives the flow and the frictional pressure drop will increase forcing the restoration of the original inlet mass flow. The inlet flow perturbation can take any functional form, which can be linearly decomposed into sinusoidal waves of different magnitudes and frequencies. The variation in density results in gravitational head change, while the mass flux variation results in friction variations. The net pressure drop variation across the channel due to the gravitational and frictional components must be compensated for by flow acceleration in order to satisfy the constant channel pressure drop boundary condition. The feedback strength is maximal for an inlet flow perturbation with a frequency comparable to the inverse of the delay time, and if the magnitude of the feedback is sufficiently strong, the channel hydraulic parameters will oscillate at that preferred frequency with an increasing

magnitude. The hydraulic stability of the density waves depends on the strength of the feedback processes.

The quantitative parameter for measuring the degree of stability is the decay ratio defined as the oscillation magnitude at a given cycle relative to the previous cycle's magnitude. Under typical BWR conditions, the decay ratio is increased (less stable) with the following system variables:

- High power to flow ratio: This increases the density contrast along the channel (and hence the gravity head) which drives the instability.
- Low flow: In addition to being the denominator in the power-to-flow ratio, low flow is destabilizing because it decreases the preferred frequency (because lower flow speed increases bubble transit time) and thus reduces the axial attenuation of the mass flux and void fraction.
- Bottom-skewed power peaking: This power shape results in creating the bubbles at lower elevations which remain for a longer time, thus increasing the average density contrast and thus is destabilizing.
- Low system pressure: Low system pressure is destabilizing as the difference in saturated liquid and vapor densities increases which drives the gravitational component, hence a destabilizing effect.
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- High inlet subcooling: The inlet subcooling does not have a monotonic effect on stability, as very high and very low inlet subcooling are both stabilizing. Very high inlet subcooling prevents boiling and suppresses density response by preventing

phase change. Reducing inlet subcooling to allow boiling, while remaining sufficiently high such that the boiling boundary is high, the two-phase-to-single-phase pressure drop ratio is low and the system remains stable. [

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The discussion of the hydraulic density waves, idealized under constant pressure drop and constant rate of heating, remained in the linear small amplitude regime. The large oscillation amplitude effects will be discussed separately.

4.2.1 Density Waves in Parallel Channels between Two Plena

In describing the idealized density wave instability a constant channel pressure drop was imposed as a boundary condition, which can be assured using a recirculation loop much larger than the boiling channel. In the case the recirculation loop is not so large the oscillating flow will result in pressure drop boundary changes which are stabilizing. Two identical boiling channels connected in parallel to the same recirculation plena will be coupled only if the recirculation loop is finite and the common pressure drop responds to the net flow change in the two boiling channels. The coupling results in the two channels oscillating out-of-phase (180 degrees phase shift) such that the pressure drop boundary fluctuation is minimized. For three identical boiling channels connected in parallel to the same plena, the coupling forced by the recirculation loop will result in the channels oscillating 120 degrees apart. However, for four channels, there are two possibilities, either the four channels will oscillate 90 degrees apart, or two channels will oscillate in-phase with each other and out-of-phase with the other two channels. The situation can become very complicated when hundreds of channels are connected to the same plena. In real situations, the channels are not identical and therefore have natural frequencies that are not identical, and their respective degrees of stability are also different. Coherent oscillations where many channels share the frequency and phase depend on the coupling mechanism of the neutron flux in addition to the

recirculation loop. Yet, hydraulically unstable channels, if sufficiently destabilized, may break away and oscillate independently from other channels and experience a superposition of multiple oscillation modes.

4.2.2 Density Wave with Power Oscillations due to Density-Reactivity Coupling

The propagation of the density wave along the boiling channel results in an oscillation of the bundle average coolant density (equivalently void fraction). The change in void fraction changes the neutron absorption and fission cross sections and produces a neutron reactivity response. The reactivity oscillation in turn produces a fission power response. There are two components of the power response, the first is the fission power deposited in the UO_2 pellets, and the second is the power deposited directly in the coolant as gamma radiation and neutron moderation.

The direct energy response is practically immediate, i.e. in-phase, with the original density change and results in an opposing effect on coolant density, i.e. negative feedback. The in-phase negative feedback of the direct energy deposition in response to density change has a stabilizing effect on the density wave.

The fission energy deposition is eventually transported to the coolant via heat conduction through the fuel rod. The dynamics of the transport of heat through the fuel rod to the coolant are governed by the heat capacity and the various thermal resistance components between the pellet interior and the coolant. These thermal resistances include the UO_2 pellet, the Zircaloy clad wall, the gap between pellets and clad, and the coolant in contact with the outer clad wall. The result of the thermal resistance and heat capacity inertia is a delay of the heat transport to the coolant, i.e. phase lag, of slightly less than 90 degrees. It also accounts for considerable attenuation of the heat source to the coolant. The attenuation of the heat flux amplitude relative to the fission power oscillation amplitude is of the order of 10^{-1} and increases with increasing the conduction time constant, which in turn increases with increasing the fuel rod diameter and increasing pellet-gap resistance. Unlike the direct energy deposition in the coolant, the

time lag of the coolant heating response through clad wall heat flux relative to the perturbation of the fission power deposition in the pellet results in destabilizing the density wave.

The void reactivity-to-power feedback not only provides the coupling needed for the different channels to oscillate coherently, but also has a destabilizing effect that makes it possible for the system to be unstable even when every channel in it is stable hydraulically. The coherence is broken if a single channel becomes hydraulically unstable at which point the flow in that channel will reflect a superposition of its intrinsic instability and the driven component via the oscillating power. Unstable single channel oscillations have been observed in unusual situations, for instance when a BWR bundle is not properly seated and deprived of flow (as occurred in Forsmark-I and Brunsbüttel Reference 6). Single channel instability also occurred in Garigliano during a special test (Reference 5). Aside from these unusual situations, single channel instabilities have been predicted and special effort has been made to exclude the possibility of their occurrence in the approved AREVA methodology for DIVOM calculations (Reference 7).

4.3 *Oscillation Modes – The Global Mode*

As mentioned earlier, several boiling channels connected in parallel to two plena may not oscillate coherently absent a mechanism for coupling the density waves among the individual channels. In the case of a BWR core, the density-reactivity feedback provides the required coupling. The neutron flux in the core responds to reactivity changes anywhere in the core due to neutron diffusion. Thus the reactivity change in one channel results in a corresponding power change not only in that channel but to all other channels - with varying strength.

The oscillation mode where the power in every channel oscillates coherently, and in-phase with the power in all other channels, is called the global mode. The inlet mass flows in all the channels oscillate similarly, with the same frequency and in-phase. As the inlet mass flow rate in all channels is in-phase, the total core inlet mass flow rate

must also oscillate, and similarly for the core exit mass flow rate. The core pressure drop (between the upper and lower plena) must also oscillate. Accordingly, the recirculation loop flow must interact with the core flow, and its dynamics must be considered in the analysis of the global mode oscillations. Generally, the friction and inertia of the recirculation loop exert a stabilizing influence on the global mode, and the extent of this stabilizing effect depends mostly on the dynamics of the steam separator assembly.

In a BWR core oscillating in-phase the fundamental mode of the neutron flux distribution function is excited. The excitation of all the other planar harmonics is not needed for the global mode. The axial flux harmonics must be driven as a result of the density waves causing the observed phase lag between neutron detector responses of the upper core elevation relative to lower elevation. [

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4.4 *Oscillation Modes – The Regional Mode*

The regional mode is characterized with half the core bundles oscillating out-of-phase with the other half. The two core halves are separated by a vertical plane, which is also called the neutral line when a planar projection is considered. The net core flow remains unchanged during the regional oscillation provided its magnitude is not so large as to introduce nonlinear effects that do not cancel out.

The main reason the hydraulic channels prefer to oscillate out-of-phase is the cancellation of the recirculation loop damping. The regional mode oscillation in a BWR

is forced to be coherent with half the core bundles oscillating in-phase and the other half oscillating with a 180 degree shift due to neutronics coupling. The half-core oscillation is preferred because it excites the first azimuthal neutron flux mode and thus receives the highest possible amplification. The other flux harmonics that can be excited by other channel groupings are characterized by larger subcritical reactivities, and therefore are significantly damped.

It is important to notice that the decay ratios of the regional and global modes are comparable. The regional mode is preferred for

- Large cores, which result in small eigenvalue separation for the first azimuthal flux mode.
- Low center power peaking (ring of fire), which also decreases the eigenvalue separation.
- Loose inlet orifice, which destabilize the hydraulic channels. This effect favors the regional mode in the absence of recirculation loop damping. It must be emphasized that the regional oscillations are isolated, and thus independent from the recirculation loop.

4.5 *Oscillation Modes – The Rotational Mode*

In the regional oscillations described above, the neutral symmetry line is stationary. The rotational mode is similar to the regional mode where the neutral line is oscillating or rotating (See Reference 8). The rotational mode essentially results from the simultaneous excitation of two orthogonal azimuthal modes. Assuming the core loading and control rod patterns are symmetric, the first two azimuthal modes are degenerate (approximately equal eigenvalues), and are thus indistinguishable. In the case the stability threshold is crossed with a decay ratio slightly greater than unity, and the core symmetry is not exact, it is expected that only one first azimuthal flux harmonic is excited leading to a regional mode oscillation with fixed neutral line. In the case the core is destabilized further, the orthogonal azimuthal mode is excited next, and interference patterns emerge depending on the relative amplitude and frequency differences between the two modes. The neutral line may oscillate or rotate slowly in

response to unequal magnitude and frequency of the two excited azimuthal modes. The most interesting case is when the two azimuthal modes are degenerate and oscillate with the same frequency and amplitude, which leads to the neutral line rotating at the same frequency.

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4.6 *Oscillation Modes – Axial Power Shape*

As the density wave propagates upward, not only the total reactivity oscillates, but also the axial reactivity distribution is altered where oscillating reactivity difference between the upper and the lower parts of the core is created. As a result, the axial neutron flux harmonic is driven by density waves. The effect of the axial power shape oscillation on the decay ratio is minimal but it is noticeable as the cause of the phase lag of the upper local power range monitor (LPRM) power signal relative to the power signal from the lower LPRM on the same string.

The axial mode excitation is significant when the global or regional/rotational oscillation amplitude is large and large axial power shape changes are expected during the oscillations. [

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4.7 *Large Amplitude and Limit Cycles of Global Mode Oscillations with Linearized Hydraulics*

As the oscillation magnitude increases, nonlinear effects are introduced (Reference 9). The neutron kinetics nonlinear effects become significant before the hydraulic nonlinear effects. This is the case because the reactivity oscillations required to induce large neutron flux response can be produced by relatively small coolant mass flow oscillation magnitude. In the idealized case of assumed linear thermal-hydraulics, with only the nonlinear effects of reactivity on the power response being allowed, a stabilizing effect has been observed which eventually leads to saturating the growth of the oscillation until a stable limit cycle is reached. The nonlinear stabilizing effect originates in the negative reactivity shift that is produced in response to the average power increases, the latter is due to the oscillating reactivity where the increase in reactivity during half a cycle increases power more than compensated for by an equal reactivity decrease in the subsequent half cycle. This asymmetric power response to reactivity oscillation is also responsible for generating high and sharp power peaks compared with the flat power minima.

The power drift under oscillatory reactivity results in an average power increase that is balanced by the negative reactivity due to the increased average void fraction. [

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4.8 *Large Amplitude Regional Mode Oscillations with Linearized Hydraulics*

The power oscillation magnitude considered here is sufficiently large for the nonlinear neutron kinetics effects to manifest, but not high enough for the nonlinear effects of the hydraulics to become important. The regional oscillation of large amplitude differs in

basic ways from a global oscillation (Reference 10). Most importantly, there is no reactivity bias associated with the first azimuthal harmonic excitation and growth, unlike the fundamental flux excitation and growth in the global mode oscillation. The only negative reactivity that reduces the first azimuthal mode growth is the subcriticality associated with its steady state eigenvalue being less than unity, and this subcriticality is not affected by the oscillation magnitude and therefore not a nonlinear effect. The main nonlinear effect of the growth of the first azimuthal mode is the emergence of a driven fundamental mode oscillation component with relative magnitude proportional to the square of the first harmonic magnitude at low oscillation magnitudes and at double its frequency (see References 11 and 12). The double frequency fundamental mode will grow until it becomes equal to the first harmonic in magnitude. A negative reactivity shift is generated, [

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For growing regional oscillations, unlike the global mode, the nonlinear effects accelerate the rate of growth, and the oscillation magnitude is not self-limited (Reference 10). The eventual arrest of the regional mode oscillation growth is due to [ ] This is one reason why large amplitude regional oscillations are of special interest and can be considered limiting compared with the nonlinearly self-limiting global oscillations.

4.9 *Large Amplitude Pure Thermal-Hydraulic Density Waves*

Flow in a boiling channel includes highly nonlinear processes. For example, the frictional pressure drop is approximately proportional to the square of the flow rate, and the void fraction dependence on the steam quality is also nonlinear. When unstable density waves in a boiling channel, without neutron kinetic feedback, are allowed to grow the oscillation magnitude may reach a limit cycle [

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Detailed numerical models are needed to simulate the behavior of a boiling channel as an integral system whereas purely analytical models are of limited use for understanding the effects of various phenomena particularly for large amplitude oscillations. However, it is still possible to discern the role of these phenomena by observing the behavior of oscillating channels in test loops, and in simulations, and guided by knowledge of the fundamentals of flow dynamics. Using these tools, a qualitative description of the nonlinear effects and their influence on density wave oscillations growth is offered here.

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The main nonlinearly destabilizing effect originates from the [

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] The maximum oscillation amplitude at the channel inlet is negative, where the reverse flow magnitude [

] The inlet flow oscillation has broad peaks and sharp minima signifying the nonlinear processes involved in the generation of these high amplitude oscillations.

4.10 *Very Large Nonlinear Oscillations of Global and Regional Types*

For small amplitude oscillations, the system behavior is linear and the principle of superposition is applicable. Accordingly, all the possible unstable modes will be manifested without coupling to each other, for example in the case the decay ratios of the global and regional modes are comparable and greater than unity, both types of instabilities will be excited and the resultant oscillations will reflect a superposition mix. This is not the case for large oscillations where nonlinear effects are significant. [

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] An exception for this behavior is the case of two regional modes representing the excitation of a first azimuthal flux harmonic and a nearly degenerate mode where the corresponding neutral lines are orthogonal. The nonlinear effects of the growth of one of these modes will impact both modes, and simultaneous growth of the two modes would lead to oscillating or rotating neutral line, i.e. the mixed mode oscillation of the rotational type is not discouraged by nonlinear effects.

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4.11 *Prompt-Criticality*

Very large power oscillations result from very large reactivity oscillations due to the severe flow oscillations. It is important to consider that reactivities in excess of the delayed neutron contribution may occur, i.e. prompt-super-criticality. The possibility of prompt-super-criticality requires the neutron kinetics models to be able to handle it properly with finite neutron velocity and Doppler reactivity feedback. However, from theoretical analysis (see References 9 and 10) and experience with numerical calculations (Reference 2), it has been found that prompt-criticality may be expected only under unrealistically rapid rate of oscillation growth, before the system has time to respond by increasing the average power and shift the average reactivity to a large negative value. Even in this case, the prompt criticality is exceeded by only a few cents, not dollars like in reactivity insertion accidents. No qualitatively distinct power pulses result from small super-prompt-critical reactivity.

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4.12 *Effect of Bypass Flow with Possible Boiling*

Boiling is possible in the upper part of the core bypass at natural circulation under relatively high power (Reference 13). This effect is modeled in steady state simulators which provide the initial conditions and neutron cross sections to the transient codes

used in this application. The main effect of the bypass boiling is a shift of the axial power shape to more bottom-peaking. The important question here is the transient response of the bypass, with or without boiling, in the presence of large regional mode oscillations. Under regional oscillations, the core pressure drop remains nearly invariant as the effects of flow in the two halves of the core oscillating out-of-phase tend to cancel out. [

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4.13 *Cyclical Dryout and Rewetting*

Large oscillations of flow and power in a BWR bundle can result in conditions of degraded heat transfer and clad temperature excursions beyond the safe limits designated to maintain fuel coolability.

In steady state operation, the conditions of heat transfer degradation are associated with the inception of dryout. Dryout correlations based on critical heat flux or critical quality concepts are used to obtain the critical power ratio, which provide quantitative measure of allowable bundle power and define the safety margins to protect that limit. Quasi-steady state application of dryout correlations has been the basis for protecting the fuel against dryout. Steady state dryout correlations were extended for applications to DIVOM oscillatory transients (Reference 7), which are rather mild, compared with power and flow oscillations accompanying anticipated transient without scram. [

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With the detailed accounting of the phenomena governing the cyclical dryout and rewetting, the limiting consideration for fuel safety is shifted from dryout inception to failure to rewet. Accordingly, cyclical dryout and rewetting is not considered a threat to fuel integrity as long as high clad temperature excursion does not occur. [

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A detailed [ ] model for cyclical dryout and rewetting with possible failure to rewet [

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4.13.1 Impact of Cyclical Dryout and Rewetting on Very Large Oscillations

The phenomena of large density wave oscillations and cyclical dryout are interlinked. The previous section addresses the cyclical dryout and rewetting, with possible failure to rewet, [

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4.14 *Vessel Considerations*

Aside from the vessel impact on global oscillations as described in Section 4.3, there are several systems that can have a direct impact on the evolution of the ATWS-I event. One such category is the systems involved in pressure control. For a 2RPT, normal pressure control is maintained by the pressure control system through the positioning of the turbine control valves. Pressure change is gradual resulting in minimal effect on the system. For a turbine trip, the turbine is isolated and pressure is controlled by the bypass valves and Safety/Relief Valves (SRV). The fast opening and closing of the SRVs cause pressure to oscillate in the vessel. [

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The mitigating actions of the operators will also impact the progression of the ATWS-I event. There are two primary mitigating actions required to shut down the ATWS-I event: water level reduction and boron injection. The water level reduction begins by operator action at an assumed time following entry into the ATWS condition. Water level is reduced to a level that is typically near the top of the core. This reduction in water level to a point significantly below the feedwater spargers means that once water level control is re-established at the lower level, the cold feedwater is now sprayed through a steam environment. The large distance between the spargers and the water level means that the feedwater will be heated to near saturated conditions. This heated water will both reduce core power and stabilize the core terminating oscillations.

The boron injection process begins as soon as the ATWS condition is recognized (same time as the beginning of the water level reduction). However, system delays due to pump run up, the time required to pump boron from the holding tank to the vessel, and the time required for boron to accumulate in the core in sufficient concentration to suppress power mean that this action can be delayed relative to the water level reduction. As such, the water level reduction will mitigate the oscillation prior to boron injection. In this case, the boron injection serves to shut the core down and limit impact of high core power on the containment heat load. The RAMONA5-FA ATWS-I methodology will not include boron injection as the base analysis definition. If, in the unlikely scenario, the boron injection is needed to be modeled, it can be modeled
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4.15 *Phenomena Ranking*

A comprehensive phenomenon identification and ranking process has been performed to identify the dominant phenomena for the target scenario (ATWS leading to core instability). During the development process, this identification and ranking is the basis for determining which closure relations require development or enhancement for the intended application. The process aids in methodology development because it prioritizes the model development activities and defines the types of validation and sensitivity studies that are needed to support the methodology. The culmination of the process is a phenomenon identification and ranking table (PIRT) that summarizes all the phenomena and provides their importance ranking relative to selected figure of merit (FoM). This PIRT utilizes AREVA's extensive history and experience with analyzing and testing density wave phenomena including the core minimum critical power ratio (MCPR) response for both global and regional mode instabilities as a function of the power oscillation magnitude (Reference 7). In the approved EO-III TR (Reference 30) AREVA included [

]. In addition, AREVA has previously participated in industry efforts to develop a previous ATWS-I PIRT (Reference 31).

When developing the PIRT, it is important to note that the development of very large power and flow oscillations follows a progressive path from the initial inception of instability, exponential oscillation growth from noise level to mild amplitudes, and further growth to large amplitudes that can be prevented from further growth by nonlinear effects. Intuitively, the phenomena and parameters participating in all the stages of the evolution of the transient from inception to maximum oscillation amplitude are ranked such that any important phenomenon at any stage remains important for the ultimate maximum oscillation event under consideration. [

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[

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Acceptance criteria for the target scenario are derived from General Design Criteria 35:

> "A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."

For the ATWS-I event, maintaining the PCT below the 10CFR 50.46 limit of 2200 °F (1204 °C) ensures that the acceptance criteria of maintaining core coolability is met.

[

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The first FoM is the oscillation inception (OI). Since the purpose of the evaluation is to demonstrate that operator actions occur in sufficient time to prevent severe overheating of the clad, the method must be able to accurately determine when oscillations would begin relative to the operator intervention.

The second FoM is the limit cycle amplitude (LCA). While not all events will reach the full limit cycle, this point determines the ultimate magnitude of the oscillations and defines the worst case oscillation.

The final FoM is post-dryout (PDO). This FoM encompasses the time between the initial prediction of dryout through to rewet or failure to rewet. The ability to accurately determine dryout and rewetting can directly determine the severity of the event.

The ranking of the phenomenon and processes are presented in Table 4-1.

Table 4-1: Ranking of Phenomena and Processes

H - High
M - Medium
L - Low
N - N/A

Table 4-1: Ranking of Phenomena and Processes (cont.)

[ ]

H - High
M - Medium
L - Low
N - N/A

Table 4-1: Ranking of Phenomena and Processes (cont.)

H - High
M - Medium
L - Low
N - N/A

Table 4-1: Ranking of Phenomena and Processes (cont.)

H - High
M - Medium
L - Low
N - N/A

4.16 *Major Assumptions*

Assumptions are necessary measures and approximations needed to create any practical analytical or numerical tool such as done in this work. The listing of the key or major assumptions is desirable, at least in part, to put the accuracy and expectations of the model performance in the right perspective, enlighten the user as to the application limitations, and point to areas of future improvements.

The identification and justification of assumptions is often an exercise of engineering common sense more than a quantitative analysis with objective metrics. Fortunately in this particular case, the RAMONA5-FA ATWS-I model and code are based on a solid foundation of practice and experience with codes of similar nature, namely the previously approved RAMONA5-FA (Reference 22). The assumptions which represent simplifications or improvements or any deviation of significance from the experience base of the original RAMONA5-FA will be listed and discussed. Complementing assumptions in models [ ] will also be discussed depending on their particular significance to the correspondence between the particular model/assumption and the important ATWS-I phenomena which is the key application of the code.

The key assumptions are listed below along with the associated consequences and justifications. The neutron kinetics representation using [

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[

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New water property functions based on the IF97 formulation are used [ ]. The use of the most up-to-date formulation is assumed to imply no significant changes to the nature of the oscillation transient, and any change no matter how small is an improvement. The density of liquid water at temperatures (or equivalently enthalpies) higher [

]

5.0 CODE DESCRIPTION

A detailed description of the theory is given in this section. Description of the various components of the model, the neutron kinetics, the fluid flow, pin heat conduction and heat transfer to the fluid, is given in separate subsections. Notice that the nomenclature is defined in each subsection as the model description is presented and the meaning of symbols may differ within each model. For example, ρ refers to reactivity in the description of the neutron kinetics, while the same symbol is used for density in the fluid flow model.

Model description is brought to the level of discretized formulation, with the aid of governing differential equations [

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5.1 *Neutronics*

An adaptive 3-D neutron kinetics module has been developed for application to BWR transient codes. [

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[

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5.1.1 Adaptive Two-Group 3-D Neutron Kinetics

The adaptive kinetics theory is presented in this section along with a derivation of the 3-D governing equations in two-energy-groups. The two-group neutron diffusion equations in the steady state simulator MICROBURN-B2 are:

$$-\nabla \cdot \left(D_1(\mathbf{r})\nabla\Psi_1(\mathbf{r})\right)+\Sigma_1(\mathbf{r})\Psi_1(\mathbf{r})=\frac{1}{k_0}\left(\nu\Sigma_{f1}(\mathbf{r})\Psi_1(\mathbf{r})+\nu\Sigma_{f2}(\mathbf{r})\Psi_2(\mathbf{r})\right) \tag{5.1}$$

$$-\nabla \cdot \left(D_2(\mathbf{r})\nabla\Psi_2(\mathbf{r})\right)+\Sigma_{a2}(\mathbf{r})\Psi_2(\mathbf{r})=\Sigma_{12}(\mathbf{r})\Psi_1(\mathbf{r}) \tag{5.2}$$

where

D_1	Fast group diffusion coefficient
D_2	Thermal group diffusion coefficient
k_0	Effective multiplication factor (eigenvalue)
$\mathbf{r}$	Space vector
Σ_1	Fast neutron removal cross section (by absorption and slowing down)
Σ_{a2}	Thermal neutron absorption cross section
Σ_{12}	Slowing down cross section
$\nu\Sigma_{f1}$	Fast fission neutron production cross section
$\nu\Sigma_{f2}$	Thermal fission neutron production cross section
Ψ_1	Fast flux steady state distribution
Ψ_2	Thermal flux steady state distribution

The transient form is given by

$$\frac{1}{v_1}\frac{\partial \Phi_1(\mathbf{r},t)}{\partial t} = \nabla \cdot \left(D_1(\mathbf{r},t)\nabla\Phi_1(\mathbf{r},t)\right) - \Sigma_1(\mathbf{r},t)\Phi_1(\mathbf{r},t) + \left(1-\beta(\mathbf{r},t)\right) \times \frac{1}{k_0}\left(\nu\Sigma_{f1}(\mathbf{r},t)\Phi_1(\mathbf{r},t) + \nu\Sigma_{f2}(\mathbf{r},t)\Phi_2(\mathbf{r},t)\right) + \sum_{n=1}^{N}\lambda_n(\mathbf{r},t)\,C_n(\mathbf{r},t) \tag{5.3}$$

$$\frac{1}{v_2}\frac{\partial \Phi_2(\mathbf{r},t)}{\partial t} = \nabla \cdot \left(D_2(\mathbf{r},t)\nabla\Phi_2(\mathbf{r},t)\right) - \Sigma_{a2}(\mathbf{r},t)\Phi_2(\mathbf{r},t) + \Sigma_{12}(\mathbf{r},t)\Phi_1(\mathbf{r},t) \tag{5.4}$$

$$\frac{\partial C_n(\mathbf{r},t)}{\partial t} = \frac{\beta_n(\mathbf{r},t)}{k_0}\left(\nu\Sigma_{f1}(\mathbf{r},t)\Phi_1(\mathbf{r},t) + \nu\Sigma_{f2}(\mathbf{r},t)\Phi_2(\mathbf{r},t)\right) - \lambda_n(\mathbf{r},t)\,C_n(\mathbf{r},t), \quad n=1,\cdots,N \tag{5.5}$$

where

t	Time
v_1	Fast neutron velocity
v_2	Thermal neutron velocity
N	Total number of delayed neutron energy groups
β_n	Delayed neutron fraction in the group n
λ_n	Decay constant for the delayed neutron precursor of the group n
C_n	Concentration of the delayed neutron precursor of the group n
Φ_1	Time- and space-dependent fast flux distribution
Φ_2	Time- and space-dependent thermal flux distribution

The total delayed neutron fraction is the sum of the group-wise fractions, thus

$$\beta(\mathbf{r},t) = \sum_{n=1}^{N}\beta_n(\mathbf{r},t) \tag{5.6}$$

The fast removal cross section is the sum of absorption and slowing down components, thus

$$\Sigma_1(\mathbf{r},t) = \Sigma_{a1}(\mathbf{r},t) + \Sigma_{12}(\mathbf{r},t) \tag{5.7}$$

where

Σ_{a1} Fast absorption cross section.

Notice that the eigenvalue, k_0, is retained to maintain consistency with the steady state cross sections and force the initial condition to exact criticality. [

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[

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5.1.2 Numerical Solution

The cross sections and fluxes are defined on the same 3-D mesh as MICROBURN-B2. The x-coordinate is associated with the index i, the y-coordinate with j, and the vertical z-coordinate with k. For example

$$\Psi(\mathbf{r}) = \Psi(x, y, z) = \Psi_{i,j,k}$$

[

]

[ ]

5.1.2.1 Interfacial Diffusion Coefficient Approximation

[ ]

5.1.2.2 Steady State and Initialization

5.1.2.3 Time Integration Procedure

5.1.3 Cross Section Representation

5.1.4 Power Generation

The thermal-hydraulic calculations in RAMONA5-FA require the heat generation or power density in each computational cell. The power density at any point is the sum of two components, namely:

- Prompt fission heat: the amount of energy released promptly in the fission process, which is proportional to the fission rate.
- Decay heat: the amount of energy released by the decay fission products. It is delayed relative to the prompt fission heat and, hence, depends on the fission rate history.

The power density can be written as

$$q(\mathbf{r},t)=q_P(\mathbf{r},t)+q_D(\mathbf{r},t) \tag{5.66}$$

where

q total heat generation rate,
q_P prompt component,
q_D delayed component,
t time,
$\mathbf{r}$ spatial coordinate.

5.1.4.1 Prompt Fission Heat

5.1.4.2 Decay Heat

The amount of delayed energy released by the decay of the fission products following the shutdown of a reactor is fitted with a series of decay heat groups. Thus, the model is similar to the handling of delayed neutrons. The concentration of decay heat precursors in a decay group can be expressed in each neutronic node by the following differential equation

$$\frac{dD_i}{dt} = \alpha_i \left(\kappa\Sigma_{f1}\, \Phi_1 + \kappa\Sigma_{f1}\, \Phi_2 \right) - \lambda_i D_i = \alpha_i\, \kappa F(\mathbf{r},t) - \lambda_i D_i \tag{5.69}$$

where

D_i concentration of decay heat precursors

α_i fraction of the total energy appearing as decay heat in decay group

λ_i decay constant [1/sec]

i decay group number

The decay heat model is based on the American National Standard for Decay Heat Power in Light Water Reactors [Reference 23]. This standard uses 23 groups to fit the decay heat power t seconds after a fission pulse from a fissionable nuclide with a series of exponential terms. Each group is characterized by the fraction of the total fission energy appearing as decay heat in the group, α_i , and by a decay constant, λ_i.

The decay heat component can be written for RAMONA5-FA as:

$$q_D(\mathbf{r},t) = \sum_{i=1}^{23} \lambda_i D_{H,i} \tag{5.70}$$

The concentration of the decay heat precursors, D_H , for the group i is given by the following differential equation, which is very similar to (5.69)

$$\frac{\partial D_{H,i}}{\partial t} = \gamma_i\, \kappa F(\mathbf{r},t) - \varepsilon_i\, D_{H,i} \tag{5.71}$$

If we assume that $F(\boldsymbol{r},t)$ is a pulse at time zero, then the solution of (5.71) is given by

$$D_{H,i} = \gamma_i \, \kappa \exp(-\varepsilon_i t) \tag{5.72}$$

However, the ANS Standard gives the expression and the values of the coefficients for the function $f(t)$ which represents the decay heat power t seconds after a fission pulse,

$$f(t) = \sum_{i=1}^{23} \alpha_i \exp(-\lambda_i t) \tag{5.73}$$

Combining Equation (5.72) with (5.70), we obtain for a fission pulse

$$q_D(\mathbf{r},t) = \sum_{i=1}^{23} \gamma_i \, \varepsilon_i \, \kappa \exp(-\varepsilon_i t) \tag{5.74}$$

Comparing Equations (5.73) and (5.74), expressions for the coefficient set (γ,ε) are obtained in terms of the coefficients of the ANS standard (α,λ). Thus,

$$\begin{aligned} \varepsilon_i &= \lambda_i \\ \gamma_i &= \frac{\alpha_i}{\kappa \lambda_i} \end{aligned} \tag{5.75}$$

The coefficients γ_i must be corrected to take into account two effects. First, if the fuel has been irradiated for a finite period of time, T, the coefficients are redefined as

$$\gamma_i = \frac{\alpha_i}{\kappa \lambda_i} \left[1 - \exp(-\lambda_i T) \right] \tag{5.76}$$

The second correction takes into account for the fact **[**

].

The total fraction of fission energy appearing as decay heat is obtained from

$$\gamma_T = \sum_{i=1}^{23} \gamma_i \tag{5.77}$$

which is used in the prompt power density calculation in Equation (5.67).

5.1.5 Power Deposition

RAMONA5-FA takes into account the fact that the fission energy is deposited as thermal energy both inside the fuel pellet, where the fission takes place, and outside the pellet due to neutron slowing down and gamma attenuation. [

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[

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5.2 *Fuel Thermodynamics*

In this section, the modeling of heat conduction in the heating elements is presented. The transient heat conduction serves as a link between the energy deposition from nuclear fission and the dynamics of the fluid flow in the fuel assemblies. The heat conduction in fuel rods consisting of pellet and cladding and pellet-clad gap is needed for the dynamic modeling of the reactor core, while modeling of heat conduction in electrically heated pins is needed for performing benchmarking calculations for comparison with test loop data. For either fuel or heater rods, the models for heat transfer between the rod surface and the coolant covering different heat transfer regimes are presented. For the analysis of fuel rods, [

]

[

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5.2.1 ATWS-I Fuel Pin Heat Conduction

Heat conduction in the fuel pin is assumed to be azimuthally symmetric with no axial component. The transient heat conduction equation is thus

$$\rho C \frac{\partial T}{\partial t} = \frac{1}{r} \frac{\partial}{\partial r}\left(k\, r \frac{\partial T}{\partial r} \right) + q''' \qquad (5.82)$$

where

r	radial coordinate from the fuel pin center
t	time
$T(r,t)$	temperature
$q'''(r,t)$	volumetric heat generation rate
ρ	density
C	specific heat at constant pressure
k	thermal conductivity

The fuel rod is made of 3 components, fuel pellet, clad and pellet-clad gap. [

]

The boundary condition at the pellet center is

$$\frac{\partial T}{\partial r} = 0 \quad at \quad r = 0 \qquad (5.83)$$

The boundary condition at the outer clad surface is obtained as heat flux continuity, thus

$$-k_c \frac{\partial T}{\partial r}\bigg|_{r=R} = h\left(T_{wall} - T_{ref} \right) \qquad (5.84)$$

where

T_{ref} heat sink (coolant) temperature

R outer clad radius

T_{wall} outer clad surface temperature

h heat transfer coefficient

k_c clad thermal conductivity

The pellet-clad gap is modeled as a thin layer with no thermal inertia but finite thermal resistance. Thus the inner clad radius is approximated as equal to the outer pellet radius, R_f. The heat flux across the gap is obtained from

$$-k_c \frac{\partial T}{\partial r}\bigg|_{r=R_f+} = -k_f \frac{\partial T}{\partial r}\bigg|_{r=R_f-} = h_{gap}\left(T_f - T_{ci}\right) \tag{5.85}$$

where

h_{gap} gap heat conductance

T_f temperature at fuel pellet outer radius

T_{ci} temperature at clad inner radius

k_c clad thermal conductivity

k_f pellet thermal conductivity

R_f pellet outer radius (R_{f-} and R_{f+} refer to the pellet and clad sides of the gap respectively)

[ ]

[ ]

Figure 5-1: Fuel rod discretization

[ ]

5.2.2 ATWS-I Heater Rod Heat Conduction

In order to properly model the KATHY test facility, [

]. The transient heat conduction equation in the radial direction is written as

$$\rho C \frac{\partial T}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r\,k \frac{\partial T}{\partial r} \right) + q''' \qquad (5.121)$$

The different variables used in Equation (5.121) are defined below

ρ	Heater rod material density (kg/m^3)
C	Heater rod material heat capacity (J/kg.K)
k	Heater rod material thermal conductivity (W/m.K)
q'''	Time-dependent Ohmic heat source deposited in the heater per unit volume (W/m^3)
T	Time- and space-dependent temperature in the rod wall (C)
t	Time (s)
r	Radial distance (m)

[

]

Figure 5-2: Heater rod discretization

The steady state solution is obtained by setting the time derivative on the left hand side of Equation (5.121) to zero to get

$$\frac{1}{r}\frac{d}{dr}\left(r\,k\frac{dT}{dr}\right)=-q''' \tag{5.123}$$

[ ]

5.2.3 Heat Transfer Coefficient

The heat transfer coefficient calculations include wetted conditions under single- or two-phase flow, partially or fully dry conditions, and the transitions between these different regimes. With possible heat transfer from pin surface to liquid and/or vapor [

]

Wet heat transfer coefficient

[ ]

[

]

Dry heat transfer coefficient

[

]

[ ]

5.2.4 Hot Fuel Pin Model

The following assumptions are made to model the hot pin in addition to the ones described in the previous section:

[ ]

5.2.5 Material Properties

5.2.6 Pellet-Clad Gap Heat Transfer Coefficient

[ ]

5.2.7 Radial Power Deposition Distributions in Fuel Pellets

[ ]

[ ]

5.3 *Thermal-hydraulic Model*

The RAMONA5-FA ATWS-I thermal-hydraulics model is a [ ] equation, non-homogeneous, non-equilibrium, one-dimensional two-phase flow model with constitutive equations for thermodynamic state variables, as well as heat transfer and vapor generation/condensation. Thermal non-equilibrium between the phases is accounted for by allowing the liquid in a two-phase mixture to depart from saturated conditions. Similarly, [ ]. A description of the hydraulics is given in this section.

The channel is divided into N nodes, which are control volumes [ ]. The flow area and hydraulic diameter for each node is allowed to vary in order to account for specific bundle design features such as part-length fuel rods. The 1-D formulation

[

]

[

] These equations are written directly for control volumes which directly correspond to the as programmed model. The control volume formulation is straightforward, and there is no need to follow the customary style of first writing the partial differential equation set and applying finite differencing over the control volume as approximations.

It is important to note that a volume conservation equation is not an independent equation with physical significance like the mass, momentum, and energy equations. However, the fact that the volume of the fluid species in a node sum up to the volume of that node, is useful in arranging the information content of the physical balance equations.

5.3.1 General Description of the System Considered

The hydraulic loop is divided into nine main parts as shown in Figure 5-3, namely:

Downcomer 1
Downcomer 2, with NDC2 parallel paths
Lower Plenum 1
Lower Plenum 2
Core, with NPC parallel paths
Core Upper Plenum
Standpipes
Steam Separators
Steam Dome

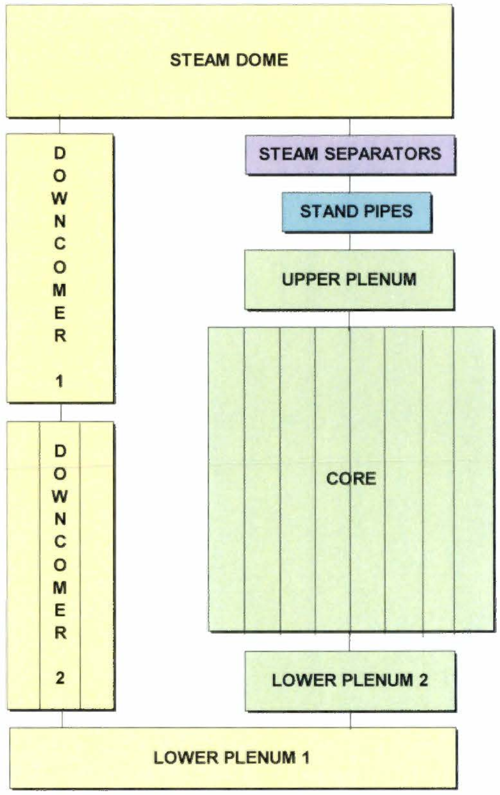

Figure 5-3: Loop Parts in the Vessel Hydraulics Model

All the loop parts are vertical except for Lower Plenum 1, which is horizontal. As indicated in Figure 5-3, Downcomer 2 is divided into NDC2 parallel paths, and the core is divided into NPC parallel channels consisting of NCC coolant channels and one moderator or bypass channel. The bypass channel accounts for the inter-channel flow as well as the internal water structures in the assemblies.

The parallel paths of Downcomer 2 (DC2) are hydraulically coupled at the Downcomer 1 (DC1) and at the Lower Plenum 1 (LP1). The core channels are also hydraulically coupled at the Lower Plenum 2 (LP2) and core Upper Plenum (UP). The Upper Plenum

(UP), Standpipes (SP), and Steam Separators (SS) can be modeled separately or lumped into one volume (riser component).

[

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The position of the feedwater inlet is limited to Downcomer 1 (DC1) and the pump may be anywhere in Downcomer 1 or Downcomer 2 (DC2). Systems with multiple pumps can be explicitly modeled.

[

]

Power is supplied to the loop in two different ways:

- Heat transfer from the surfaces of the fuel rods to the coolant (see Section 5.2.1)
- A fraction of the nuclear power is directly generated in the coolant and moderator (see Section 5.1.5 above)

5.3.2 [ ]

[ ]

[

]

5.3.3 Vapor generation rate

The vapor generation rate is modeled [

]

[

]

5.3.4 **Mass conservation**

The mass conservation equation is solved for the liquid and vapor phases. [

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[

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[

]

5.3.5 Energy Conservation

Energy conservation, in the ATWS-I formulation, requires solving [

]

[ ]

5.3.6 [ ]

[ ]

[ ]

The wall surface temperature is obtained by solving the heat conduction equation for the fuel rod, which is presented in Section 5.2.4.

5.3.7 [ ] Momentum Conservation

[ ]

[

]

5.3.7.1 [ ] Formulation in the Vessel Components

[

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[

]

5.3.7.2 [ ] Formulation in the Core

[

]

[

]

5.3.8 Pressure Calculation

[

] The equations describing the transient pressure calculation are presented in this section.

The mass balance equations for the vapor and liquid phases respectively are

[

]

[ ]

5.3.9 Steam Dome Equations

[ ]

[ ]

5.3.10 Recirculation Flow

The recirculation flow calculation is based on [

]

Figure 5-4: Recirculation Loop

[ ]

5.3.11 Constitutive Equations

The most important constitutive equations used in the standard thermal-hydraulics are

- Friction and two phase friction multiplier
- Local pressure loss models
- Abrupt contraction/expansion pressure change models
- [ ]
- Thermodynamic steam-water properties
- [ ]
- Heat transfer correlations
- Evaporation rate correlation

5.3.11.1 Friction and Two Phase Friction Multiplier

Two options are available in RAMONA5-FA to specify the single-phase friction factor. The first option is of the form:

[ ]

5.3.11.2 Local Pressure Loss Models

Local pressure loss models are available in RAMONA5-FA to predict the local pressure loss effect of spacer grids, tie plates, etc. contained in the BWR core and vessel. Local pressure losses are captured in the Δp_{loc} term of Equation (5.244).

[

]

5.3.11.3 Abrupt Contraction/Expansion Pressure Change Model

In addition to pressure losses related to loss coefficients, the local pressure drop is affected by models for treating the pressure change resulting from abrupt contractions and expansions that occur at the inlet and outlet of core component hydraulic channels.

[

]

[ ]

5.3.11.4 [ ]

[ ]

[

]

5.3.11.5 Thermodynamic Steam-Water Properties

Thermodynamic variables are determined corresponding to the system pressure and liquid or vapor enthalpy. The water properties in RAMONA5-FA utilize the new IF97 formulation [Reference 29].

5.3.11.6 [ ]

[

]

5.3.12 Numerical Integration Techniques

Table 5-1: Summary of State Equations

[

]

5.4 *Steam Line Flow Dynamics*

RAMONA5-FA has a steam line model to make it capable of simulating acoustic effects in the steam line due to sudden valve closures or openings, leading to pressure waves traveling back and forth in the steam line. Strong pressure pulses in the steam line will in turn induce pressure pulses in the steam dome and core, thus causing a void collapse in the core and a sudden increase in reactivity.

As the time-scale of these phenomena is much shorter than the time scale of the dominating phenomena inside the reactor vessel, the need for a separate model to be able to resolve these effects is obvious.

[

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Figure 5-5: Steam Line Model

[

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[

]

5.5 *SPECIAL MODELS*

Special models are included in RAMONA5-FA to simulate specific flow conditions. They include at present:

- Recirculation flow pumps
- Jet pump model
- Feedwater sparger
- Steam separators
- Dryout/rewet model

5.5.1 Recirculation Pump Model

The recirculation pumps, one in each recirculation loop, drive the recirculation flow through the recirculation loops into the jet pumps or drive the core flow in the plants with internal pumps. The recirculation pumps are used to control the core flow, consequently the inlet subcooling temperature, the non-boiling heat and, thereby, the vapor void fraction, moderator density and fission power. The purpose of the recirculation pump model is to predict the pressure rise $\left(\Delta p_{pump}\right)$ used in the drive force (Equation (5.242)). This pressure rise depends not only on the flow rate in the pump but also in the angular speed (N_{pump}) of the recirculation pump. The angular speed is computed from the equation of motion for the pump, the motion being governed by the balance between the electrical torque, the friction torque and the hydraulic torque.

There are three different recirculation pump types implemented in RAMONA5-FA.

[ ]

5.5.1.1 Pump Model 1

5.5.1.2 Pump Model 2

5.5.1.3 Pump Model 3

[ ]

Table 5-2: Homologous Pump Curves Definition

[ ]

5.5.2 Jet Pump Model

The jet pump model calculates the pump head of the internal jet pumps. [

]

5.5.3 Steam Separator Model

When modeling the steam separators, there are two main effects which will be accounted for in this section:

- the flow inertia in the spiral path of the separators
- the volumetric flow of saturated vapor leaving the circulation loops and entering the steam dome above the coolant level (entrainment of vapor or carry under)

The effect of the form loss is taken into consideration in the same manner as other flow components.

5.5.3.1 Flow Inertia

5.5.3.2 Carry under Flow

[ ]

[ ]

Figure 5-6: The Loop Parts of the Vessel Hydraulics

[ ]

5.5.4 Feedwater Sparger Condensation Model

When the water level falls below the feedwater inlet, a special model takes into account the condensation due to the sub-cooled water injected into the saturated steam. This term is added to the vapor generation rate from Section 5.3.3 to determine the total evaporation/condensation in the downcomer. This condensation rate is calculated according to the following expression

[ ]

[ ]

5.5.5 Dryout and Rewetting Model

The Critical Power Reduced Order Model (CPROM) for dryout and rewetting is based on [ ]

[ ]

5.6 *PLANT CONTROL AND PROTECTION SYSTEMS*

Below follows a short summary of the plant control and protection system models implemented in RAMONA5-FA:

- Pressure control system consisting of turbine control, bypass valve and safety and relief valve (SRV).
- Plant protection systems (PPS) including high-pressure coolant injection system (HPCI), control rod, main steam isolation valve (MSIV) and recirculation pump trips.
- Reactor core isolation cooling system (RCIC)
- Feedwater control system, for water level control
- Feedwater trip.

5.6.1 Pressure Control System

In a BWR, an increase in the vessel steam pressure results in an increase in steam flow through the steam line and the collapse of steam void in the reactor. The latter means that the water density increases and, hence, there is an increase in neutron moderation. This in turn leads to an increase in power, which has the effect of increasing pressure. In order to deal with this positive feedback a pressure regulator is provided. To maintain an approximately constant system pressure, a turbine control valve is provided. The bypass valve also directs the excess steam flow to the primary heat sink during load rejection events. Since the turbine and bypass flow controllers will influence the system response during many transients, it is important to have them properly represented in RAMONA5-FA.

5.6.1.1 Turbine Valve Regulator

The pressure regulator is composed of a pressure sensor transmitter, comparator, PID element, lag element, and a valve actuator. This model allows RAMONA5-FA to reasonably simulate the actual plant pressure control system.

5.6.1.2 Bypass Valve Regulator

The turbine bypass valve is not operative during normal plant operation. It opens to dump the excess steam to the condenser during abnormal transients such as turbine trip, load rejection. The bypass valve regulator is identical to the turbine control valve regulator. However, the operational characteristics may be quite different than the turbine control valve.

5.6.2 Feedwater Controller

The purpose of the feedwater controller is to determine the proper feedwater flow rate by comparing water level demand with level set point. If the error in the water level combined with the imbalance between steam and feedwater flow rates exceeds operational tolerance, proper action will be taken by the feedwater controller to correct this deficiency. The feedwater controller model employed in RAMONA5-FA is similar to the ones suggested in the literature with some adjustments and simplifications. In addition, RAMONA5-FA allows for an operator initiated reduction in water level in order to properly model the mitigation actions during an ATWS-I event.

5.6.3 Safety and Relief Valves

In a BWR, there are valves in the steam lines, which serve as safety and relief valves to maintain the system pressure below prescribed limits. RAMONA5-FA represents up to five different banks; each bank representing one or more safety or relief valves. Each bank can be activated on either steam line or dome pressure or at a user defined time.

[ ]

[

]

5.6.4 Main Steam Isolation Valve (MSIV)

The purpose of the MSIV model is to predict the mass flow rate of steam through the valve during its closure. [

]

5.6.5 Plant Protection System (PPS)

The BWR plant protection system (PPS) is to assure that the consequences of all postulated conditions do not exceed the specified safety limit in the reactor system. It should provide the required protection by sensing the necessity and implementation of reactor scrams, pump trips, and turbine trip.

In RAMONA5-FA, the PPS functions include both manual and automatic modes. In the manual mode, the operator's action is simulated through a user specified "trip time", at which the desired shutdown system is activated. In the automatic mode, important system variables are processed through appropriate PPS subsystem trip functions for possible protective action in response to selected instrumentation signals. Following the PPS signal, a user specified trip time delay must be exceeded before PPS action is initiated.

5.6.6 Manual Operator Actions

Manual operator actions to reduce water level during an ATWS-I event are simulated in RAMONA5-FA through the feedwater control system. The time at which operator actions begin is specified by input, and is simulated by a reduction in the water level setpoint to a user specified value. [

] The feedwater pumps coast down until the feedwater controller takes control to maintain the water level at the new lower level.

5.7 *NUMERICAL TIME INTEGRATION*

This section summarizes the different time-integration techniques used to integrate the RAMONA5-FA differential equations and the intertwining of the numerical integration of the neutronics with that of the main hydraulics and the steam line integration.

5.7.1 Neutron Kinetics

[

]

5.7.2 Fuel Thermodynamics

[

]

5.7.3 Vessel Hydraulics

[

]

[

]

5.7.4 Coupling of the Neutronics and Hydraulics Integration

[

]

6.0 CODE VALIDATION

The RAMONA5-FA ATWS-I code theory is described in Section 5.0. The code calculates the transient thermal-hydraulic response of a BWR with a detailed core representation of one channel per fuel assembly. It applies a [

] There are no limitations with respect to flow direction, and the severe flow oscillations accompanied with inlet flow reversal can be simulated.

RAMONA5-FA ATWS-I applies [

] The steady state simulator provides automated input coupling for the hydraulic and neutron cross section parameters. [

] The dynamic functionality of the algorithms which is important in stability and oscillation calculations is verified by benchmarking to stability specific data.

The benchmarking of RAMONA5-FA ATWS-I is divided into two parts. The first part is focused on the pure thermal-hydraulic modeling and is accomplished by comparing the code results with the stability tests of various fuel designs performed at the KATHY loop as well as other publically available data. Both steady-state and transient benchmarks will be performed. The steady-state benchmarking includes measurement for void fraction in ATRIUM-10 and ATRIUM 10XM as well as FRIGG3. KATHY steady state pressure drop data will also be benchmarked. The measured decay ratio and frequency

for a large number of test points for ATRIUM-10 and ATRIUM 10XM are compared with the RAMONA5-FA ATWS-I calculated values. The agreement provides the needed evidence of the validity of the RAMONA5-FA ATWS-I thermal-hydraulic models.

The second part of the benchmarking is an integral exercise where both the thermal-hydraulic and the neutron kinetics are coupled in the simulation of oscillations that occurred in actual BWR plants. These benchmarks will be separated into linear stability and non-linear stability benchmarks. The linear stability benchmarks are the same used to benchmark the NRC-approved frequency domain stability code STAIF (Reference 21), and the original time-domain code RAMONA5-FA (Reference 7).

6.1 *Test Suite and Acceptance Criteria*

The RAMONA5-FA ATWS-I code validation includes the following cases:

- Comparison to KATHY void faction tests for:
  - ATRIUM-10
  - ATRIUM 10XM
  - FRIGG3
- Comparison to steady state pressure drop tests for:
  - KATHY ATRIUM-10
  - KATHY ATRIUM 10XM
- Comparison to KATHY loop stability tests. Comparisons include the decay ratio and frequency for all the runs in the test suite for the following fuel designs
  - ATRIUM-10
  - ATRIUM 10XM
- Comparison to KATHY dryout/rewet tests for ATRIUM 10XM and [ ].

- Benchmarking to all the linear instability tests in actual BWR plants included in the RAMONA5-FA test suite (Reference 7). These are:

[ ]

- Benchmarking to the non-linear stability events in actual BWR plants. These include:
  - o Oskarshamn turbine trip with non-linear oscillation
  - o BWR A feedwater temperature transient with non-linear oscillation

The acceptance criteria are satisfied if the RAMONA5-FA ATWS-I code results agree with the experimental results. The criteria are chosen to be consistent with how this data is used in other approved methodologies. Quantitatively, the criteria are

- Calculated void fractions are within [ ] of the measured value.
- Calculated pressure drops are within [ ] of the measured value.
- Calculated decay ratios are within [ ] of the measured value. Conservative trends [ ] are acceptable.
- Calculated frequencies are within [ ] Hz of the measured values.
- For dryout/rewet cases, [ ]
- For non-linear oscillations, [ ]

6.2 *Benchmarking to Void Fraction Tests*

The FRIGG experiments, Reference 32 have been included in the validating database because of the broad industry use of these experiments in benchmarking activities, including TRAC, RETRAN, and S-RELAP5. The experiments include a wide range of pressure, subcooling, and quality from which to validate the general applicability of a void correlation. However, the experiments do not contain features found in modern rod bundles such as part length fuel rods and mixing vane grids. The lack of such features makes the data less useful in validating correlations for modern fuel designs.

Because of its prototypical geometry, the ATRIUM-10 and ATRIUM 10XM void data collected at KATHY was useful in validating void correlation performance in modern rod bundles that include part length fuel rods, mixing vane grids, and prototypic axial/radial power distributions. The characteristics of the void fraction validation database are listed in Table 6-1. For ATWS-I benchmarking the range of data [

]

Figure 6-1 provides comparisons of predicted versus measured void fractions for the AREVA multi-rod void fraction validation database using the [ ] correlation. This figure shows reasonable predictions versus the measured data. The mean predictions fall within [ ] (predicted – measured) error with a mean error of [ ] and a standard deviation of [ ].

Table 6-1: Summary of Void Fraction Test Conditions

	FRIGG (Reference 32)	ATRIUM-10 KATHY	ATRIUM 10XM KATHY
Axial Power Shape	**uniform**	[ ]	[ ]
Radial Power Peaking	mild peaking	[ ]	[ ]
Bundle Design	circular array with 36 rods + central thimble	[ ]	[ ]
Pressure (psi)	435, 725, 1000, and 1260	[ ]	[ ]
Inlet Subcooling (^{0}F)	5.7 to 56.99	[ ]	[ ]
Mass Flow Rate (lbm/s) *(Based on mass flux assuming ATRIUM-10 inlet area)*	10.14 to 42.56	[ ]	[ ]
Max Void at Measurement Plane (fraction)	0.848	[ ]	[ ]
Number of Data	69 tests, 314 points	[ ]	[ ]

Figure 6-1: Validation of [ ] using FRIGG, ATRIUM-10 and ATRIUM 10XM Void Data

6.3 *Benchmarking to KATHY Pressure Drop Tests*

Steady state pressure drop measurements have been performed for the ATRIUM-10 and ATRIUM 10XM at KATHY. The geometry was prototypic and includes all necessary features such as [ ]. Pressure measurements were made at several elevations in the bundles. In addition measurements include both single phase and two phase statepoints. The characteristics of the pressure drop validation database are listed in Table 6-2.

Figure 6-2 provides comparisons of predicted versus measured pressure drops for the AREVA ATRIUM-10 multi-rod pressure drop validation. This figure shows that the majority of the predictions fall within [ ] (predicted – measured) error bands. The mean and standard deviation of the error were calculated using the method used in Reference 35. The single phase ATRIUM-10 data showed a mean error of [ ] and a standard deviation of [ ]. The two phase ATRIUM-10 data showed a mean error of [ ] and a standard deviation of [ ]. These statistics meet the NRC criteria for both average error and standard deviation.

Figure 6-3 provides comparisons of predicted versus measured pressure drops for the AREVA ATRIUM 10XM multi-rod pressure drop validation. This figure shows that the majority of the predictions fall within [ ] (predicted – measured) error bands. The mean and standard deviation of the error were calculated using the method used in Reference 35. The single phase ATRIUM 10XM data showed a mean error of [ ] and a standard deviation of [ ]. The two phase ATRIUM 10XM data showed a mean error of [ ] and a standard deviation of [ ].

Table 6-2: Summary of Pressure Drop Test Conditions

	ATRIUM-10 KATHY	ATRIUM 10XM KATHY
Axial Power Shape	[ ]	[ ]
Bundle Design	[ ]	[ ]
Pressure (bar)	[ ]	[ ]
Inlet Temperature (^{0}C)	[ ]	[ ]
Mass Flow Rate (kg/s)	[ ]	[ ]
Number of Two Phase Data Points	[ ]	[ ]
Number of Single Phase Data Points	[ ]	[ ]

[ ]

Figure 6-2: Validation of Pressure Drop Prediction versus KATHY ATRIUM-10 Measured Data

[ ]

Figure 6-3: Validation of Pressure Drop Prediction versus KATHY ATRIUM 10XM Measured Data

6.4 ***Benchmarking to KATHY Stability Tests***

Several measurement campaigns were performed in the KATHY loop under natural circulation configuration to quantify the stability characteristics of ATRIUM-9, ATRIUM-10, and ATRIUM 10XM BWR fuel designs. The results of these measurements were used for developing and validating AREVA's stability methodologies. Of particular interest here is the ATRIUM-10 stability test series (Bundle ID STS-49.1) and the ATRIUM 10XM test series (Bundle ID STS115.1). [

]

The ATRIUM-10 tests were performed during the period May 31-June 17, 1999. The test bundle STS-49.1 was characterized by a down-skew power profile with [

] From this data, decay ratios and frequencies were calculated.

The ATRIUM 10XM tests were performed during the period October 15-October 23, 2009. The test bundle STS-115.1 was characterized by a down-skew power profile with [

] From this data, decay ratios and frequencies were calculated.

The data collected from the stability testing of the ATRIUM-10 and ATRIUM 10XM bundles in the KATHY loop are used for benchmarking RAMONA5-FA ATWS-I. The operating conditions of power and inlet subcooling were varied and data were collected for each operating point. [

]

The number of ATRIUM-10 data points used in this benchmarking is [ ] while the number of ATRIUM 10XM data points used in this benchmarking is [ ]. The measured operating parameters for each test point are used as input to RAMONA5-FA ATWS-I. The decay ratio and frequency for each test point are compared with the corresponding measured values.

The following figures depict good agreement between the measured and calculated decay ratios and frequencies.

Figure 6-4: Validation of Decay Ratio Prediction versus KATHY Measured Data

Figure 6-5: Validation of Frequency Prediction versus KATHY Measured Data

6.5 *Benchmarking to KATHY Dryout/Rewet Tests*

For stability and severe oscillation testing of ATRIUM 10XM, a full scale electrically heated test section is used. The axial power shape is bottom-peaked [ ]. The heated rods include full-length and part-length rods. [ ]

The procedure in the pure thermal-hydraulic testing is similar to the previous stability testing campaigns with ATRIUM-9 and ATRIUM-10. [ ]

Information about dryout behavior is extracted from the pure thermal-hydraulic stability testing by simply allowing the flow oscillations to increase, and apply additional power increase steps as needed. Cyclical dryout and rewetting were observed in these tests by recording the responses of the many thermocouples attached to the heater rods at different elevations. [ ]

It is well-known from numerical and theoretical studies of density waves in BWRs that the reactivity-to-power feedback has a destabilizing effect. By implementing such feedback in the KATHY test loop, the loop is operated at conditions closely resembling the actual conditions in an unstable BWR. With the power feedback turned on, [

] oscillations of the flow and now power to reach high amplitudes with significant inlet flow reversal [

]

[

]

The test results were studied and important data were extracted. These data allowed the transient extraction of [

] The measured and extracted information from the tests were essential in developing the post-dryout models, and serve to benchmark it.

The entire test run database was reviewed with regard to dryout occurrence under oscillation. All the test runs that were identified as experiencing dryout at any spacer, with or without failure to rewet, are processed and used for the benchmarking of post-dryout models. [

]

The post-dryout models were originally developed and validated using the SINANO code, Reference 34. These models have been further improved as described in Section 5.9.5, and were validated with the SINANO code. The models from the SINANO code have been integrated within the RAMONA code system, and the same benchmarks were performed with RAMONA to ensure the models were properly integrated.

Figure 6-6 through Figure 6-20 show the results of the benchmarking using CPROM coefficients consistent with the ATRIUM 10XM design discussed in Appendix A. All plots compare predicted temperatures at the spacer to the measured temperatures. In some instances nodes above and below are shown to better highlight the model behavior.

Figure 6-6 through Figure 6-20 provide a qualitative evaluation of the accuracy of the post-dryout models. In order to provide a quantitative evaluation of model accuracy, the benchmarks were re-analyzed with [

] and demonstrates that models can accurately reproduce the data.

In order to validate the post-dryout models for fuel designs besides ATRIUM 10XM, [

] Results of these cases are plotted in Figure 6-22 and Figure 6-23.

Table 6-3: Dryout and rewetting ATRIUM 10XM KATHY test runs

Figure 6-6: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-7: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-8: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-9: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-10: Limiting Temperature for KATHY ATRIUM 10XM test [ ]

Figure 6-11: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-12: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-13: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-14: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-15: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-16: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-17: Lmiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-18: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-19: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-20: Limiting Temperature for KATHY ATRIUM 10XM [ ]

[ ]

Figure 6-21: Limiting Temperature for KATHY ATRIUM 10XM [ ]

Figure 6-22: Limiting Temperature for KATHY [ ]

Figure 6-23: Limiting Temperature for KATHY [ ]

6.6 *Benchmarking to Linear Reactor Stability Benchmarks*

In order to validate the functions and additions to the RAMONA5-FA ATWS-I code, several sets of linear stability reactor benchmarks were performed. [

]

utilize neutronic input taken from the benchmark material used to qualify the frequency domain code STAIF (Reference 21).

6.6.1 [ ]

[

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* [

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[

]

6.6.2 [ ]

[

]

6.6.3 [ ]

[

]

* [

]

6.6.4 [ ]

[

]

6.7 *Benchmarking to Non-Linear Reactor Benchmarks*

In order to validate the functions and additions to the RAMONA5-FA ATWS-I code, several sets of non-linear stability reactor benchmarks were performed. These benchmarks include:

- Oskarshamn Non-Linear Instability
- BWR A Feedwater Transient with Non-Linear Instability

A description of the benchmark analyses is given in the following sections along with the RAMONA5-FA ATWS-I calculated decay ratios and frequencies.

6.7.1 Oskarshamn Nonlinear Instability

On February 25, 1999 as maintenance work was being performed on a switchyard outside of Oskarshamn Unit 2, the power supply to a bus bar was interrupted for 150 ms. A load rejection signal was sent to the main breaker that connects the unit to the grid. This signal caused the turbine to trip, however due to a failure in the relay circuit, this signal was not transmitted to the reactor protection system. The system continued at power until operators initiated a partial scram. Following the partial scram, power and flow were decreased. The feedwater temperature continued to decrease causing reactor power to further increase until the system crossed the stability boundary. The reactor continued to oscillate until a high power scram was reached.

The Oskarshamn-2 non-linear stability was analyzed with RAMONA5-FA ATWS-I. Two separate scenarios were evaluated. In the first scenario, the measured pump speed was imposed and the recirculation flow was calculated based on the specified pump performance curves. The plots of relevant parameters to actual measured data for this scenario are given in Figure 6-24, Figure 6-25, and Figure 6-26. The results show good agreement with measured values with a small overestimation of the event growth rate. However, Figure 6-26 shows that the code overestimates the final recirculation flow during the final portion of the event. A second scenario was then run in which the pump speed versus time was modified to produce a recirculation flow approximately equal to the measured value. The results of this evaluation are given in Figure 6-27, Figure 6-28, and Figure 6-29. The results of this scenario show a slightly greater overestimation with the measured power results.

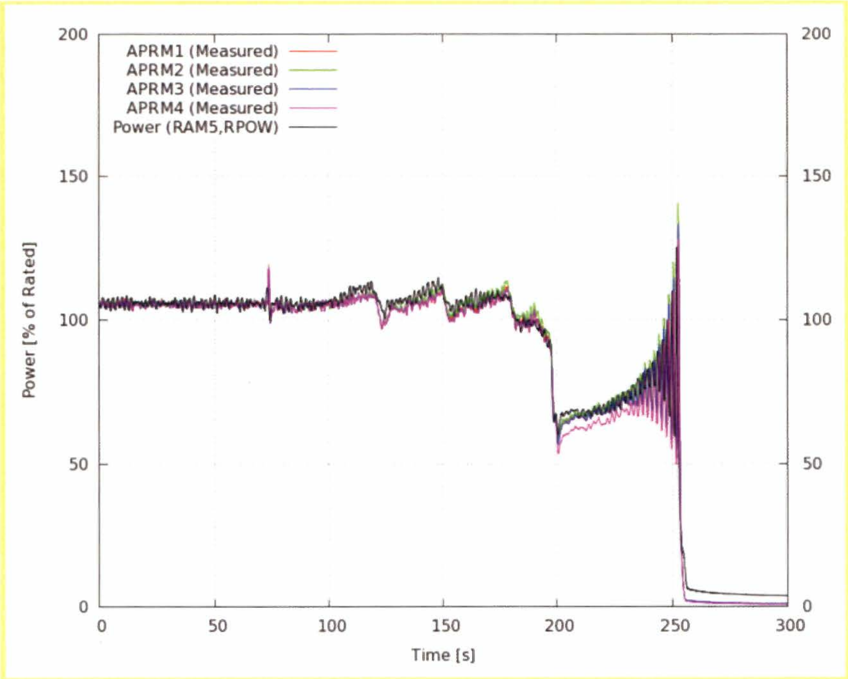

Figure 6-24: Oskarshamn-2 Non-Linear Stability Simulation Core Power, Imposed Pump Speed

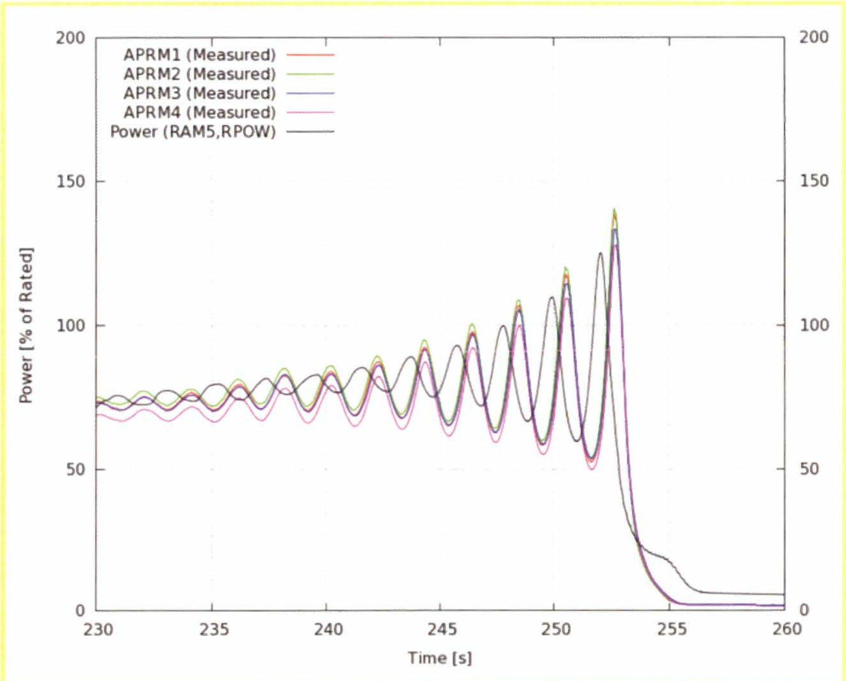

Figure 6-25: Oskarshamn-2 Non-Linear Stability Simulation Core Power - Zoom, Imposed Pump Speed

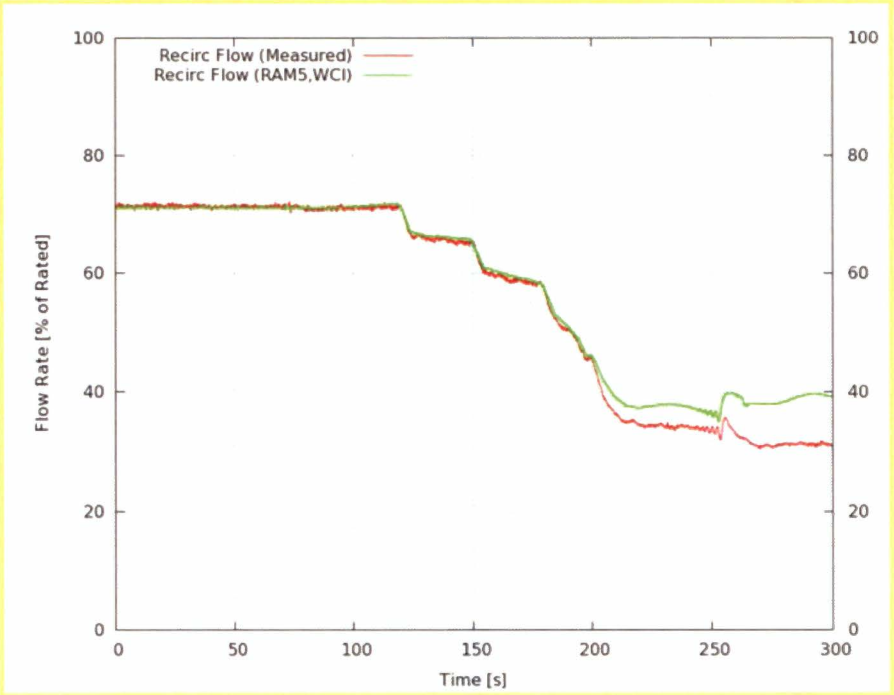

Figure 6-26: Oskarshamn-2 Non-Linear Stability Simulation Recirculation Flow, Imposed Pump Speed

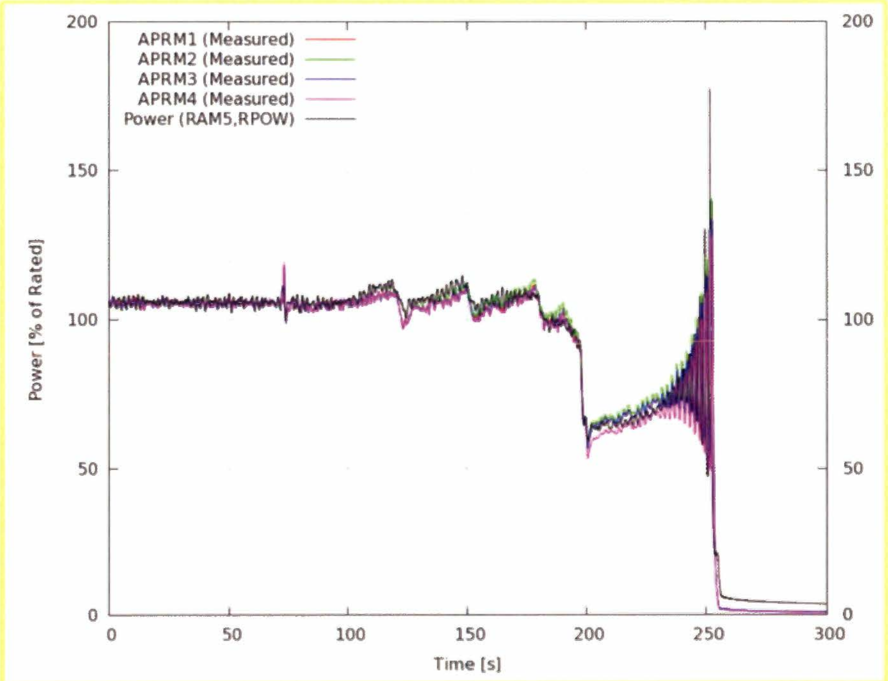

Figure 6-27: Oskarshamn-2 Non-Linear Stability Simulation Core Power, Match Recirculation Flow

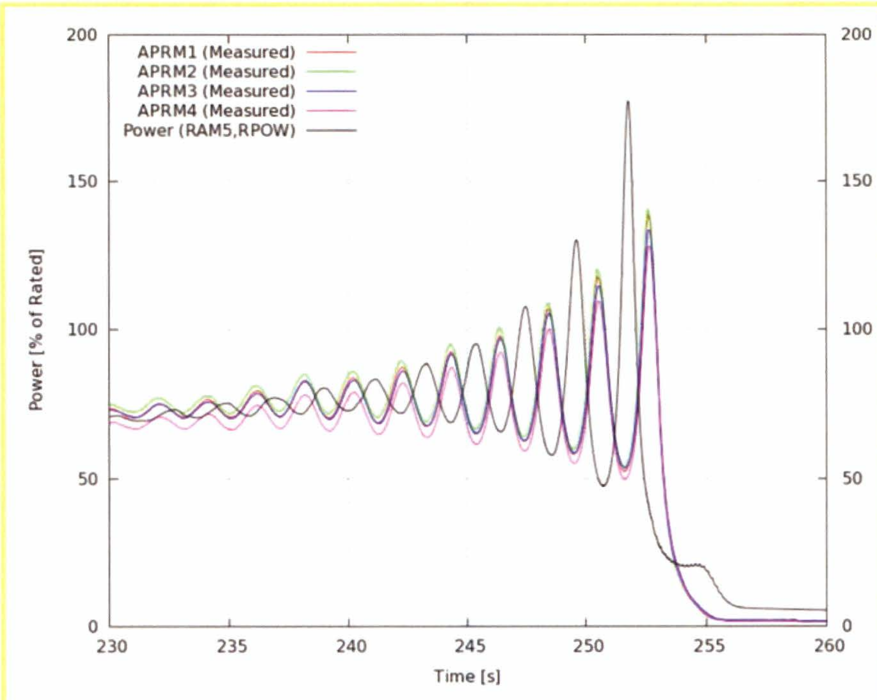

Figure 6-28: Oskarshamn-2 Non-Linear Stability Simulation Core Power - Zoom, Match Recirculation Flow

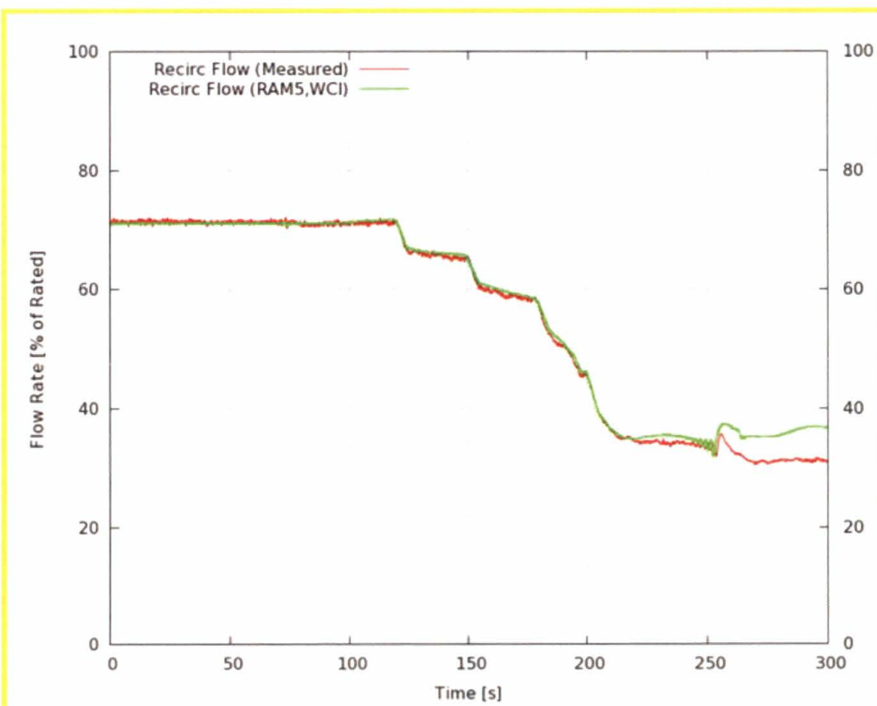

Figure 6-29: Oskarshamn-2 Non-Linear Stability Simulation Recirculation Flow, Match Recirculation Flow

6.7.2 BWR A Feedwater Transient with Non-Linear Instability

[

] was analyzed with RAMONA5-FA ATWS-I. The plots of relevant parameters to actual measured data for this scenario are given in

Figure 6-30. The results show very good agreement between the measured and calculated [ ]

[ ]

Figure 6-30: BWR [ ] Transient Non-Linear Stability Simulation

7.0 SAMPLE PROBLEMS

In addition to the benchmarking described in the previous section, the general core performance against theoretically predicted trends was also investigated. This is accomplished by running a sample ATWS-I transient and performing checks to demonstrate the ability of the code to generate very large oscillations in power and flow and demonstrate large amplitude inlet flow reversal in some bundles. Sensitivity calculations for key parameters are examined.

The sample problem is based on the Brunswick BWR4 reactor. A full core of ATRIUM 10XM was modeled, and both TTWB and 2RPT transients were simulated. [

]

7.1 *Turbine Trip*

A turbine trip with bypass was simulated [

]

Numerous sensitivity studies have also been performed to examine the sensitivity of the models to various operational variables, as well as modeling variables. The sensitivity to operational variables includes:

[ ]

To examine the effect of model variations, the following sensitivity studies were performed.

The results for the [ ] sensitivity studies are presented in Table 7-1.

Table 7-1: TTWB Sensitivity Study Results

[ ]

Figure 7-1: TTWB Base Case Core Power

[ ]

Figure 7-2: TTWB Base Case Core Inlet Flow

Figure 7-3: TTWB Base Case Core Inlet Subcooling

Figure 7-4: TTWB Base Case Limiting Channel Inlet Flow

[ ]

Figure 7-5: TTWB Base Case Vessel Water Level

[ ]

Figure 7-6: TTWB Base Case Limiting Channel PCT

7.2 *Two Recirculation Pump Trip*

A 2RPT was simulated [

]

Table 7-2: 2RPT Sensitivity Study Results

[ ]

[ ]

Figure 7-7: 2RPT Base Case Core Power

Figure 7-8: 2RPT Base Case Core Inlet Flow

Figure 7-9: 2RPT Base Case Limiting Channel Inlet Flow

Figure 7-10: 2RPT Base Case Core Inlet Subcooling

Figure 7-11: 2RPT Base Case Vessel Water Level

Figure 7-12: 2RPT Base Case Limiting Channel PCT

8.0 CALCULATION PROCEDURE

The RAMONA5-FA ATWS-I evaluation entails the analysis of simulated ATWS events at several state points throughout that cycle. This section is devoted to the general description of the methodology, the procedure for performing this calculation, and the limitation and conditions of the methodology which is presented in Table 8-1. In summary, the application procedure includes the following elements:

[ ]

Table 8-1: ATWS-I Limitation and Conditions

[ ]

9.0 REFERENCES

1. "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047-A, Class I June 1995.

2. W. Wulff et al., "BWR Stability Analysis with the BNL Engineering Plant Analyzer," NUREG/CR 5816, BNL/NUREG-52312, October 1992.

3. USNRC Standard Review Plan, *15.8 Anticipated Transient Without Scram*, NUREG-0800, Revision 2, March 2007.

4. FQM, Revision 3, *AREVA NP Fuel Sector Quality Management Manual*, AREVA NP, June 2008.

5. J. March-Leuba, "Density-Wave Instabilities in Boiling Water Reactors," NUREG/CR-6003, ORNL/TM-12130, September 1992.

6. Carsten Lange et al., "Comments on local power oscillation phenomenon at BWRs," Progress in Nuclear Energy 60 (2012) 73-88.

7. BAW-10255(P)(A) Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008.

8. A. Wysocki, J. March-Leuba, T. Downar, and A. Manera, "TRACE/PARCS Analysis of Out-of-Phase Power Oscillations With a Rotating Line of Symmetry," The 15th International Topical Meeting on Nuclear Reactor Thermal- Hydraulics, NURETH-15, paper 457, Pisa, Italy, May 12-17, 2013.

9. Y. M. Farawila and D. W. Pruitt, "A Study of Nonlinear Oscillation and Limit Cycles in Boiling Water Reactors – I: The Global Mode," Nuclear Science and Engineering: 154, 302-315 (2006).

10. Y. M. Farawila and D. W. Pruitt, "A Study of Nonlinear Oscillation and Limit Cycles in Boiling Water Reactors – II: The Regional Mode," Nuclear Science and Engineering: 154, 316-327 (2006).

11. Y. M. Farawila, "Application of Modal Neutron Kinetics to Boiling Water Reactor Oscillation Problems," Nuclear Science and Engineering: 129, 261 (1998).

12. Hideaki IKEDA et al., "Nonlinear Behavior under Regional Neutron Flux Oscillations in BWR Cores," J. Nuclear Science and Technology, Vol. 38, No. 5, p. 312-323 (May 2001).

13. D. W. Pruitt, D. R. Tinkler, and Y. M. Farawila, "Considerations for Bypass Boiling during BWR Power Oscillations," Trans. Am. Nucl. Soc., Vol. 99, pp. 739-740 (Nov. 2008).

14. D. W. Pruitt, K. R. Greene, F. Wehle, R. Velten, J. Kronenberg, A. Beisiegel, and Y. M. Farawila, " Stability and Void Fraction Measurements for the ATRIUM 10XM BWR Fuel Bundle," Proceedings of 2010 LWR Fuel Performance Top Fuel WRFPM, Orlando, Florida, Sept. 26-29, 2010.

15. F. Wehle, R. Velten, J. Kronenberg, A. Beisiegel, D. Pruitt, K. Greene, and Y. Farawila, "Full Scale Stability and Void Fraction Measurements for the ATRIUM 10XM BWR Fuel Bundle," 2011 Jahrestagung Kerntechnik, Berlin, Germany, May 17-19 2011.

16. [

]

17. W. Wulff, "Simulation of Two-Phase Flow in Complex Systems," Nuclear Technology Vol. 159, September 2007.

18. ANF-88-101(P), *MICROBURN-B: A Two-Group, Three Dimensional BWR Nodal Simulator Code*, Advanced Nuclear Fuels, July 1988.

19. Byung-Oh Cho, "Nodal Methods for Calculating Nuclear Reactor Transients, Control Rod Patterns, and Fuel Pin Powers," Ph.D. Thesis, Oregon State University, OSU-NE-9108, June 1991.

20. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.

21. EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2*, Siemens Power Corporation, August 2000.

22. EMF-3028P-A Volume 2 Revision 4, “RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2 -- Theory Manual,” AREVA NP Inc., March 2013.

23. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979.

24. W. Wulff, et al., "A Description and Assessment of RAMONA-3B Mod. 0 Cycle 4 A Computer Code with Three-Dimensional Neutron Kinetics for BWR System Transients," BNL-NUREG-51748, January 1984.

25. [ ]

26. Moody, L. F., “An Approximate Formula for Pipe Friction Factors,” *Mechanical Eng.* 69, pp 1005-1006, December 1947.

27. [ ]

28. [ ]

29. W. Wagner and H.-J. Kretzschmar, “International Steam Tables – Properties of Water and Steam Based on the Industrial Formulation IAPWS-IF97,” second edition 2008 Springer-Verlag Berlin Heidelberg.

30. ANP-10262PA Revision 0, “Enhanced Option III Long Term Stability Solution”, AREVA NP, May 2008.

31. NUREG/CR-6743, LA-UR-00-3122, “Phenomenon Identification and Ranking Tables (PIRTS) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel,” September 2001.

32. O. Nylund et al., “Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Non-Uniform Radial Heat Flux Distribution,” FRIGG-3, R-494/RL-1154, November 1969.

33. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

34. ANP-3274P Revision 3, *Analytical Methods for Monticello ATWS-I*, February 2017.

35. XN-NF-79-59(P)(A), *Methodology for Calculations of Pressure Drop in BWR Fuel Assemblies*, October 1983.

Appendix A Steady State Dryout Correlation CPROM

Abstract

A new dryout correlation is presented. This correlation, named Critical Power Reduced Order Model (CPROM), has been developed based on AREVA correlation development guidelines similar to dryout licensing correlations such as ACE. The CPROM correlation range of applicability is wide [

] making it well-suited to fitting into transient models of post-dryout that include cyclical dryout and rewetting with possible failure to rewet. CPROM is an integral part of the RAMONA5-FA transient model described in Section 5.5.5.

A.1 *Description of CPROM Correlation*

A dryout correlation of the critical heat flux type is developed. [

]

[

]

A.2 *CPROM Correlation for ATRIUM 10XM*

Table A-1: [ ]

[ ]

[ ]

[

]

[

]

Figure A-1: Calculated versus measured critical power,
[ ]

FigureA-2: [ ]

Figure A-3: [ ]

Figure A-4: [ ]

Figure A-5: [ ]

Figure A-6: [ ]

Table A-2: Statistics [ ]

[ ]

Table A-3: Statistics [ ]

[ ]

Table A-4: Statistics [ ]

[ ]

Table A-5: Statistics [ ]

[ ]

[ ]

[ ]

Figure A-7: Calculated versus measured critical power,
[ ]

[ ]

Figure A-8: [ ]

[ ]

Figure A-9: [ ]

Figure A-10: [ ]

Figure A-11: [ ]

[ ]

Figure A-12: [ ]

Table A-6: Statistics [ ]

[ ]

Table A-7: Statistics [ ]

Table A-8: Statistics [ ]

Table A-9: Statistics [ ]

The figure below shows comparison of the calculated and measured critical power for the combined data set. The mean critical power ratio is [ ] and the standard deviation of the calculated versus measured critical power for the entire database is [ ] and the number of data points is [ ].

[ ]

Figure A-13: Calculated versus measured critical power

A.3 *CPROM BOUNDS OF APPLICABILITY*

The bounds of applicability are determined by the data available to benchmark the correlation. For ATRIUM 10XM, the [

]

A.4 *CPROM COEFFICIENT GENERATION FOR NEW FUEL DESIGNS*

In order to properly generate CPROM coefficients for new fuel designs, the following requirements must be met.

[ ]

Appendix B Heat Transfer Data from KATHY Loop Stability Testing of ATRIUM 10XM

Abstract

Essential elements of the RAMONA5-FA model pertaining to the heat transfer coefficient behavior under wetted and dry conditions are extracted from measured data. The data set used for this purpose is the stability testing of the ATRIUM 10XM BWR bundle represented in full scale electrically heated module in the KATHY test facility. The testing conditions include steady state where the flow rate fluctuates only at the noise level. They also include very large unstable oscillations where significant inlet flow reversal occurs. The power level was kept constant under manual control for some tests. For other test runs, a feedback loop (SINAN module) determined the power input to the bundle and the resulting coherent power and flow oscillations of large amplitude provided a close simulation of the realistic conditions under ATWS-I transient.

The following sections provide the heat transfer parameters determining the wet and dry conditions and [

]

B.1 *Summary of Heat Transfer Coefficient Data and Observations*

The needed measured data include the test section power, pressure, inlet flow rate, inlet subcooling and the temperature of the heater rods. [

]

B.2 *Heat Transfer Coefficient under Wetted Conditions*

The heat transfer extracted at the initial testing time, where wet conditions are guaranteed, are fit to the [ ].

B.3 [ ]

[ ]

Figure B-1: [ ]

B.4 *References*

B.1. [

]

B.2. [

]

B.3. [

]

Appendix C Investigation of the Dynamic Effects of Interfacial Friction

[ ]

Figure C-1: Oskarshamn Core Power, Interfacial Friction Sensitivity

[ ]

Figure C-2: BWR A Core Power, Interfacial Friction Sensitivity

Appendix D Chromia Fuel Rod Properties

New RODEX4 based fuel rod properties models are described in Section 5.2.5. These models are intended for use with standard UO_2 pellets. For Chromia doped pellets, modifications to the fuel thermal conductivity and [ ] models will be needed to account for the effects of the Chromia doping. The Chromia doped pellet specific models presented here will be based on the RODEX4 Chromia models described in Reference D.1.

D.1 FUEL CONDUCTIVITY

The fuel thermal conductivity model for Chromia doped fuel is given in Section 7.1.1 of Reference D.1. [

] As such, this correlation forms the basis of the fit for the model shown.

- Fuel conductivity (W/m-K)

[

]

where

[ ]

D.2 [ ]

[ ]

where

[ ]

D.3 REFERENCES

D.1 ANP-10340P-A Revision 0, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," May 2018.

AREVA

December 15, 2017
NRC:17:055

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Request for Review and Approval of ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA"

AREVA Inc. (AREVA) requests NRC review and approval of Topical Report ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA" for referencing in licensing actions. This Topical Report describes a method for performing Anticipated Transient Without Scram with Instability (ATWS-I) analyses for Boiling Water Reactors (BWR).

AREVA considers some of the material contained in the enclosed document to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the report are enclosed.

In support of the Office of Nuclear Reactor Regulation's prioritization efforts, the prioritization scheme matrix is included in Enclosure A.

There are no commitments within this letter or its enclosures.

If you have any questions related to this information, please contact Mr. Alan Meginnis by telephone at (509) 375-8266, or by e-mail at alan.meginnis@areva.com

Sincerely,

Gary Peters, Director
Licensing & Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures:

A. NRC Prioritization Matrix
B. A proprietary copy of topical report ANP-10346, Revision 0
C. A non-proprietary copy of topical report ANP-10346, Revision 0
D. Notarized Affidavit

bcc: NRC:17:055
T Point: T4.12.2

M. Byram
R. Desteese
G. F. Elliott
N. E. Hottle
D. P. Jordheim
R. J. Land
S. L. McFaden
A. B. Meginnis
J. Morris
R. M. Pederson
G. A. Peters
K. S. Quick
R. R. Schnepp
D. Tinkler
L. Tupper
T. S. Wilkerson

Enclosure A:
NRC Prioritization Matrix

TR Prioritization Scheme Matrix for Metric and Resources			
Title: ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA"			
Expect submitting FY	**TAC**	**PM**	**Today's Date: 12/15/2017**
Technical Review Division(s)	**Technical Review Branch(s)**		
Factors	**Select the Criteria That the TR Satisfies**	**Points can be Assigned for Each Criteria**	**Assigned Points**
TR Classification (Select one only)	Resolve Generic Safety Issue (GSI).	6	2
	Emergent NRC Technical Issue.	3	
	New technology improves safety.	2	
	TR Revision reflecting current requirements or analytical methods.	2	
	Standard TR.	1	
TR Applicability (Select one only)	Potential industry-wide applications.	3	2
	Potentially applicable to entire groups of licensees.	2	
	Intended for only partial groups of licensees.	1	
TR Implementation Certainty (Select one only)	Industry-wide Implementation expected.	3	1
	Expected implementation by an entire group of licensees (BWROG, PWROG, BWRVIP, etc.) who sponsored the TR.	2	
	Docketed intent by U.S. plant(s) but no formal LAR schedule yet.	1	
	No U.S. plant(s) have indicated strong intent on docket to implement yet.	0	
Tie to a LAR (Select if applicable)	A SE is requested by a certain date (less than two years) to support a licensing activity or renewal date (note it in Comments).	3	0
Review Progress (Points are cumulative as applicable)	Accepted for review.	0.3	0
	RAI issued.	0.5	0
	RAI responded.	1.2	0
	SE drafted.	2.0	0
Management (LT/ET) discretion adjustment		-3 to +3	
Total Points (Add the total points from each factor and total here):			5
Comments: The need for this method is a result of NRC requirement for ATWS-I analyses for plants that operate in the MELLLA+ or Extended Flow Window.			

NRC-IC-18-039
T4.12.1
Rec'd 10/02/18 HHE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 21, 2018

Mr. Gary Peters, Director
Licensing and Regulatory Affairs
Framatome Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING FRAMATOME INC. TOPICAL REPORT ANP-10346P, REVISION 0, "ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA" (EPID: L-2017-TOP-0067)

Dear Mr. Peters:

By letter dated December 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17355A231), Framatome Inc. (Framatome, formerly AREVA Inc.) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval Topical Report ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On August 27, 2018, Alan Meginnis, Framatome Product Licensing Manager, and I agreed that the NRC staff will receive the response to the enclosed Request for Additional Information (RAI) questions within 30 days from the date of this letter.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-4053.

Sincerely,

Jonathan Rowley

Jonathan G. Rowley, Project Manager
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Project No. 728
Docket No. 99902041

Enclosures:
1. RAI Questions (Proprietary)
2. RAI Questions (Non-Proprietary)

The Enclosure transmitted herewith contains Official Use Only - Proprietary Information. When separated from the Enclosure, this transmittal document is decontrolled.

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO TOPICAL REPORT ANP-10346P, REVISION 0,

"ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA"

FRAMATOME INC.

(EPID: L-2017-TOP-0067)

BACKGROUND

The specific regulatory requirements associated with anticipated transient without scram (ATWS) events are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," and 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as well as General Design Criteria (GDC) 12, "Suppression of Reactor Power Oscillations," 14, "Reactor Coolant Pressure Boundary," 16, "Containment Design," 35, "Emergency Core Cooling," 38, "Containment Heat Removal," and 50, "Containment Design Basis," which are contained in Appendix A to 10 CFR 50.

Of those requirements, GDCs 12 and 35 are the most relevant, in that the intent of the ATWS with-instability (ATWS-I) analyses is to demonstrate that: (1) power oscillations that arise due to instability in the core are adequately mitigated by appropriate operator actions and/or automatic system responses, and (2) adequate cooling of the fuel is maintained, such that the maximum cladding temperature does not reach thresholds where significant fuel/cladding damage or metal-water reactions are expected to occur. The NRC guidance related to the boiling water reactor (BWR) ATWS-I event is presented in Standard Review Plan Section 15.8. During review of Topical Report (TR), ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," the NRC staff identified some information that would be necessary to establish adequate technical bases to make a safety determination based on the above regulatory requirements. In particular, this information directly affects the calculation of the three figures of merit (FoMs) that the demonstration of regulatory compliance are based on—the oscillation inception (i.e., how much time operators have to act), limit cycle amplitude (i.e., worst case oscillation), and post-dryout (i.e., peak cladding temperature (PCT)).

REQUEST FOR ADDITIONAL INFORMATION (RAI) QUESTIONS

The Framatome 3-dimensional core physics simulator code, MICROBURN-B2, is used to produce data transfer files which pass condensed versions of relevant information for use in the RAMONA5-FA neutron kinetics solution. [

] Therefore, the NRC staff requests the following information:

RAI-1

What [ ] information is passed from MICROBURN-B2 to the RAMONA5-FA ATWS-I calculation? What process and/or criteria, if any, are used to ensure that [ ]?

The RAMONA5-FA code incorporates several models to capture specific phenomena of interest for the ATWS-I event. These models are typically constructed from empirical correlations developed based on a combination of theoretical principles and experimental data analysis. In order for the models to capture the phenomena of interest throughout the ranges of interest for key analysis parameters, the appropriate dependencies must be accurately captured in the models. To do so, the models must consider all relevant parameters, and the empirical correlations must be based on an appropriate analysis of the available data (including any gaps or limitations). Therefore, the NRC staff requests the following information:

RAI-2

The NRC staff has the following questions regarding the fitting of model parameters to measured data:

a. For models such as the dryout-rewet model and gap conductance model which contain multiple fitting parameters, how were values for these parameters inferred in cases where direct experimental validation for each parameter is not possible or not available?

b. For the gap conductance model, the TR states that "[ ]". Describe this approach in additional detail, including how this adjustment was performed and how these values compare to similar values used in other Framatome methodologies.

RAI-3

Justify the [ ] used in the TR methodology. Include data for the [ ], where available.

RAI-4

Since the [ ] was not part of the KATHY dryout-rewet experimental validation, justify that the models are a reasonable and accurate representation of [ ] during ATWS-I.

RAI-5

The [

] How is the model ensured to give reasonable and accurate behavior under such conditions during ATWS-I?

Once all of the models and coupling equations were combined into an unified analysis methodology within RAMONA5-FA to calculate the thermal hydraulic and neutron kinetics response during an ATWS-I event, Framatome validated the overall methodology by comparing calculational results to independent benchmarks. By demonstrating that RAMONA5-FA can independently reproduce key parameters for applicable benchmarks, reasonable assurance is provided that RAMONA5-FA will reproduce the same parameters for a postulated ATWS-I event. The key comparison results are presented in the TR, but some additional detail is needed to confirm that the benchmarks, and information used in the benchmarking calculations, are applicable to the intended use of RAMONA5-FA. Therefore, the NRC staff requests the following information:

RAI-6

The NRC staff has the following questions regarding the linear stability benchmarks provided in the TR:

a. Provide a table showing the following operating conditions and calculated conditions for each linear stability benchmark case: core power, core flow rate, core inlet subcooling, axial peaking factor, peak axial power location, and radial peaking factor.

b. What fuel type(s) were present in the core for each of the linear stability benchmarks? Were all data and specifications available for these fuel types? What data and specifications required by RAMONA5-FA ATWS-I were not available, if any?

c. What neutronic and thermal hydraulic data were used from each plant in these benchmarks?

RAI-7

The NRC staff has the following questions regarding the nonlinear stability benchmarks provided in the TR:

a. What fuel type(s) were present in the core for each of the nonlinear stability benchmarks? Were all data and specifications available for these fuel types? What data and specifications required by RAMONA5-FA ATWS-I were not available, if any?

b. What deviations, if any, were made in the boundary conditions or other modeling assumptions for these cases relative to measured data and/or available benchmark specifications?

The TR includes guidance for nodalization of the plant models used in executing the RAMONA5-FA ATWS-I calculations. The nodalization selected in a model is generally a balance between managing the time required to complete a calculation, maintaining calculational stability, and resolving the time/spatial distribution of parameters to a sufficiently fine level for accuracy. In general, the testing performed by Framatome can be expected to ensure that the computational time and stability are acceptable, but the NRC staff needs to verify that the nodalization recommendations are adequate to provide reasonable accuracy in the calculations. Therefore, the NRC staff requests the following information:

RAI-8

Justify that the RAMONA5-FA ATWS-I axial nodalization scheme in the core region provides sufficient numerical fidelity for the ATWS-I calculations, including considerations of numerical diffusion and resolution of the axial void distribution. In particular, prior studies by the NRC staff and contractors have shown that the axial void distribution may need to be captured at a sufficiently high resolution to result in an accurate calculation of the axial power profile, and thus accurate calculation of stability behavior (e.g. decay ratio), in some codes. This effect has been shown to be separate from that of numerical diffusion, so provide a discussion regarding whether the average void fraction for each node is accurate enough to correctly capture the axial power distribution for stability calculations. For example, would an increase in the number of thermal hydraulic nodes lead to a significant change in the locally-averaged void fraction across the coupled neutronic nodes due to the higher resolution of the void fraction distribution, and would this significantly affect the RAMONA5-FA ATWS-I calculations?

RAI-9

Justify that the vessel nodalization used for RAMONA5-FA ATWS-I is sufficient to provide a reasonable and accurate prediction of PCT during ATWS-I events.

RAI-10

Provide an example(s) of the calculated time-dependent behavior of the [ ] during large-amplitude oscillations with flow reversal. In the cases presented in the TR, did sufficient flow reversal occur such that [ ]? If such a circumstance occurs, justify that the RAMONA5-FA ATWS-I methodology treats this circumstance in a reasonable and/or conservative way, with respect to the ATWS acceptance criteria.

Several inputs to the ATWS-I calculation are described in the TR, with specific recommendations provided. In some cases, the parameters of interest may be determined to be insensitive to specific inputs based on engineering judgment or sensitivity studies. In other cases, the parameters of interest are adjusted to achieve desired results. In all cases, the recommendations must ensure that the results from the ATWS-I calculations are accurate or conservative. In order to verify this, the NRC staff requests the following information:

RAI-11

Provide sensitivity results for one or more linear stability benchmark cases and a simulated ATWS-I event (either a nonlinear benchmark problem or a sample full-core case) by adjusting the gap conductance values. Show time-dependent results for power, PCT, and other relevant results.

RAI-12

The NRC staff has the following questions regarding time step control:

a. What input parameters are provided by RAMONA5-FA ATWS-I to control the timestep size? What are the recommended values for use?

b. What values for these parameters were used for the nonlinear stability benchmarks and the sample problem provided in the TR?

c. Provide a set of sensitivity results for timestep size, similar to the sensitivity study provided for RAI 11.

RAI-13

What spatial distribution is used for the [ ]? Justify that the modeling approach for [ ].

RAI-14

Justify that the [ ]

The TR provides a brief procedure that would be used to perform the ATWS-I analysis and determine whether acceptance criteria are met. [ ] Therefore, the NRC staff requests the following information:

RAI-15

[ ].

RAI-16

For the process described in RAI 15, discuss how the various modeling and input assumptions remain appropriate when considering their effect on the time of oscillation onset.

framatome

ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA Response to NRC Request for Additional Information

ANP-10346Q1NP
Revision 0

March 2019

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	New Document

Contents

BACKGROUND

This document comprises Framatome's response to the Nuclear Regulatory Commission (NRC's) Request for Additional Information (RAIs) for the Licensing Topical Report (LTR) ANP-10346, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA." The RAIs were transmitted in Reference 9.

The specific regulatory requirements associated with anticipated transient without scram (ATWS) events are contained in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," and 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as well as General Design Criteria (GDC) 12, "Suppression of Reactor Power Oscillations," 14, "Reactor Coolant Pressure Boundary," 16, "Containment Design," 35, "Emergency Core Cooling," 38, "Containment Heat Removal," and 50, "Containment Design Basis," which are contained in Appendix A to 10 CFR 50.

Of those requirements, GDCs 12 and 35 are the most relevant, in that the intent of the ATWS with-instability (ATWS-I) analyses is to demonstrate that: (1) power oscillations that arise due to instability in the core are adequately mitigated by appropriate operator actions and/or automatic system responses, and (2) adequate cooling of the fuel is maintained, such that the maximum cladding temperature does not reach thresholds where significant fuel/cladding damage or metal-water reactions are expected to occur. The NRC guidance related to the boiling water reactor (BWR) ATWS-I event is presented in the Standard Review Plan, Section 15.8. During review of Topical Report (TR), ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," the NRC staff identified some information that would be necessary to establish adequate technical bases to make a safety determination based on the above

regulatory requirements. In particular, this information directly affects the calculation of the three figures of merit (FoMs) that the demonstration of regulatory compliance are based on-the oscillation inception (i.e., how much time operators have to act), limit cycle amplitude (i.e., worst case oscillation), and post-dryout (i.e., peak cladding temperature (PCT)).

RAI-1

The Framatome 3-dimensional core physics simulator code, MICROBURN-B2, is used to produce data transfer files which pass condensed versions of relevant information for use in the RAMONA5-FA neutron kinetics solution [

]. Therefore, the NRC staff requests the following information:

1. *What [ ] information is passed from MICROBURN-B2 to the RAMONA5-FA ATWS-I calculation? What process and/or criteria, if any, is used to ensure that [*

   ]?

Framatome Response RAI-1:

[

]. These include:

[ ]

The RAMONA5-FA code incorporates several models to capture specific phenomena of interest for the ATWS-I event. These models are typically constructed from empirical correlations developed based on a combination of theoretical principles and experimental data analysis. In order for the models to capture the phenomena of interest throughout the ranges of interest for key analysis parameters, the appropriate dependencies must be accurately captured in the models. To do so, the models must consider all relevant parameters, and the empirical correlations must be based on an appropriate analysis of the available data (including any gaps or limitations). Therefore, the NRC staff requests the following information

RAI-2

The NRC staff has the following questions regarding the fitting of model parameters to measured data:

a. *For models such as the dryout-rewet model and gap conductance model which contain multiple fitting parameters, how were values for these parameters inferred in cases where direct experimental validation for each parameter is not possible or not available?*

b. *For the gap conductance model, the TR states that [*

]. Describe this approach in additional detail, including how this adjustment was performed and how these values compare to similar values used in other Framatome methodologies.

Framatome Response RAI-2–a:

The fitting of coefficients depends not only on experimental data but also on assumptions that are based on first principles and reasonable engineering judgment. These must be applied on a case by case basis.

[

]

Framatome Response RAI-2–b:

RAI-3

Justify the [ ] used in the TR methodology. Include data for the [ ].

Framatome Response RAI-3:

[ ]

Figure 1 [

]

Figure 2 [

]

[ ]

Figure 3 [

]

Figure 4 [

]

[

]

Figure 5 [

]

[

]

Figure 6 [

]

Figure 7 [

]

Figure 8 Dry [ ]

RAI-4

Since the [ ] was not part of the KATHY dryout-rewet experimental validation, justify that the models are a reasonable and accurate representation of [ ] during ATWS-I.

Framatome Response RAI-4:

[

]

RAI-5

The [ ]. How is the model ensured to give reasonable and accurate behavior under such conditions during ATWS-I?

Framatome Response RAI-5:

[

]

Figure 9 Illustration of [

]

[

]

Figure 10 Illustration of [ ]

Figure 11 Illustration of [ ]

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Figure 12 Illustration of [

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Figure 13 Illustration of the [ ]

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[

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Figure 14 Illustration of [

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[

]

Figure 15 Illustration of [ ]

Once all of the models and coupling equations were combined into an unified analysis methodology within RAMONA5-FA to calculate the thermal hydraulic and neutron kinetics response during an ATWS-I event, Framatome validated the overall methodology by comparing calculational results to independent benchmarks. By demonstrating that RAMONA5-FA can independently reproduce key parameters for applicable benchmarks, reasonable assurance is provided that RAMONA5-FA will reproduce the same parameters for a postulated ATWS-I event. The key comparison results are presented in the TR, but some additional detail is needed to confirm that the benchmarks, and information used in the benchmarking calculations, are applicable to the intended use of RAMONA5-FA. Therefore, the NRC staff requests the following information:

RAI-6

The NRC staff has the following questions regarding the linear stability benchmarks provided in the TR:

a. *Provide a table showing the following operating conditions and calculated conditions for each linear stability benchmark case: core power, core flow rate, core inlet subcooling, axial peaking factor, peak axial power location, and radial peaking factor.*

b. *What fuel type(s) were present in the core for each of the linear stability benchmarks? Were all data and specifications available for these fuel types? What data and specifications required by RAMONA5-FA ATWS-I were not available, if any?*

c. *What neutronic and thermal hydraulic data were used from each plant in these benchmarks?*

Framatome Response RAI-6–a:

The range of operating conditions for the various linear stability reactor benchmarks are shown in Table 1.

Table 1 Operating conditions of the linear reactor benchmarks

[

]

Framatome Response RAI-6–b:

The linear stability benchmarks were all taken from the approved STAIF (Reference 5) and RAMONA5-FA (Reference 6) benchmarking suites. For these benchmarks the majority of fuel inputs were already available. In some cases, [

].

[

].

Framatome Response RAI-6–c:

The neutronics and thermal-hydraulics data for these benchmarks was taken directly from the approved STAIF (Reference 5) and RAMONA5-FA (Reference 6) benchmarking suite. No additional neutronics or thermal-hydraulics data was required.

RAI-7

The NRC staff has the following questions regarding the nonlinear stability benchmarks provided in the TR:

a. *What fuel type(s) were present in the core for each of the nonlinear stability benchmarks? Were all data and specifications available for these fuel types? What data and specifications required by RAMONA5-FA ATWS-I were not available, if any?*

b. *What deviations, if any, were made in the boundary conditions or other modeling assumptions for these cases relative to measured data and/or available benchmark specifications?*

Framatome Response RAI-7–a:

The Oskarshamn-2 benchmark consisted of [

].

Framatome Response RAI-7–b:

For the Oskarshamn-2 benchmark, the initial power, core flow and inlet subcooling were all input as the initial measured conditions. The feedwater flow versus time was decreased from the measured data to ensure reasonable water level calculation. This was necessary to restore correct mass balance as the calculated steam flow underpredicted the measured steam flow which would have led to an upward drift in water level. Modifying the feedwater flow versus time to better control water level is also consistent with the application of the method which utilizes a feedwater control system for the same purpose. The feedwater temperature versus time was taken directly from the benchmark specifications and includes the adjustment to the temperature function that accounts for instrumentation delay due to pipe heat

conduction. The pressure was set to the measured value versus time. The control rod insertions were modeled as they occurred in the event. Two sets of runs were made for pump speed. In one case, the measured pump speed versus time was prescribed. Reviewing the output of this run, it was observed that the core flow just prior to oscillation inception is overpredicted when compared to measurements. To investigate this effect, a second run was made with a pump speed that was modified to provide a core flow close to the measured values. Both cases were presented in the topical report.

The TR includes guidance for nodalization of the plant models used in executing the RAMONA5-FA ATWS-I calculations. The nodalization selected in a model is generally a balance between managing the time required to complete a calculation, maintaining calculational stability, and resolving the time/spatial distribution of parameters to a sufficiently fine level for accuracy. In general, the testing performed by Framatome can be expected to ensure that the computational time and stability are acceptable, but the NRC staff needs to verify that the nodalization recommendations are adequate to provide reasonable accuracy in the calculations. Therefore, the NRC staff requests the following information:

RAI-8

Justify that the RAMONA5-FA ATWS-I axial nodalization scheme in the core region provides sufficient numerical fidelity for the ATWS-I calculations, including considerations of numerical diffusion and resolution of the axial void distribution. In particular, prior studies by the NRC staff and contractors have shown that the axial void distribution may need to be captured at a sufficiently high resolution to result in an accurate calculation of the axial power profile, and thus accurate calculation of stability behavior (e.g. decay ratio), in some codes. This effect has been shown to be separate from that of numerical diffusion, so provide a discussion regarding whether the average void fraction for each node is accurate enough to correctly capture the axial power distribution for stability calculations. For example, would an increase in the number of thermal hydraulic nodes lead to a significant change in the locally-averaged void fraction across the coupled neutronic nodes due to the higher resolution of the void fraction distribution, and would this significantly affect the RAMONA5-FA ATWSI calculations?

Framatome Response RAI-8:

A theoretical discussion of the numerical fidelity of the algorithms used for BWR stability must come with a disclaimer pointing to the wide opinions promulgated in the nuclear industry where no studies can claim completeness or generality of applicability to all codes. While a discussion is provided here, the ultimate support for the RAMONA5-FA code numerical fidelity rests on the empirical proof afforded by good agreement with a wide variety of experimental data forming a benchmarking database, and stressing the code with numerical experiments and sensitivity studies.

Numerical diffusion and damping

Numerical diffusion results from the residuals of finite differencing (or finite volume) approximations of transport differential equations. Neglected higher order terms resemble diffusion operators acting on the transported property, e.g. temperature moving in a 1-D pipe. In the ideal example of a step change in temperature of a fluid moving through a pipe, the exact solution maintains the same step function shape which moves along the pipe at the flow speed. Physical or numerical diffusion causes the sharp step change to diffuse and the temperature distribution spreads increasingly with time. This numerical diffusion is a kinematic aspect as it does not involve forces or momenta. The effect of diffusion (spread) of various flow variables on the momentum is not discussed in the available literature beyond numerical experiments demonstrating a damping effect of the perturbations as functions of time; the connection between the spatial diffusion and temporal damping is implied but only as a qualitative trend without proof. An attempt to make this dynamic connection in the context of density waves is presented here. [

] Another possible effect of diffusion is increasing the attenuation of the flow perturbations as they travel from the inlet to higher elevations. In this case, the magnitude of the pressure drop near the exit would be reduced relative to that of the inlet which has a dampening effect. A third possible effect is related to how the momentum balance equations are formulated where either nodal momentum equations are used or a single integrated momentum equation is formulated per boiling channel.

The effect of increased attenuation of the flow perturbations as they travel up the boiling channel has been shown [

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ratio increase is not large, the benchmarking becomes biased. For the large oscillation magnitude cases, the increased nodalization also produced larger oscillations that [

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Figure 16 Calculated Versus Measured Decay Ratio for [

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Figure 17 Calculated Versus Measured Frequency for [
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Table 2 Linear reactor benchmarks [

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Figure 18 [ ]

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Figure 19 [
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Figure 20 [

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Figure 21 [

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Figure 22 [

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Figure 23 [

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Table 3 provides a comparison between the results of the Brunswick ATWS-I sample base case with standard nodalization and the doubled core nodalization case.

Table 3 Comparison between Brunswick sample case results with standard and doubled core nodalization

Parameter	Base Case	Doubled Nodalization
Time of oscillation inception (sec)	113	92
Time of failure to rewet excursion (sec)	117	97
Time of peak bundle power (sec)	117.2	143.3
Peak bundle power (MW)	5.74	5.69
Time of maximum core inlet subcooling (sec)	147.1	151.5
Maximum core inlet subcooling (kJ/kg)	195.8	205.5
Time of peak clad temperature (sec)	140.7	143.5
Peak clad temperature (°C)	656	677

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RAI-9

Justify that the vessel nodalization used for RAMONA5-FA ATWS-I is sufficient to provide a reasonable and accurate prediction of PCT during ATWS-I events.

Framatome Response RAI-9:

The Brunswick test case for ATWS-I was rerun with the nodalization in the vessel

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[

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Figure 24 Brunswick sample problem showing effect on [
] due to vessel nodalization

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Figure 25 Brunswick sample problem showing effect on [ ] due to vessel nodalization

[

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Figure 26 Brunswick sample problem showing effect on [ ] due to vessel nodalization

Figure 27 Brunswick sample problem showing effect on [ ] due to vessel nodalization

For the same calculation where the operator actions are omitted (unmitigated case), the core inlet subcooling varies towards the end of the run due to the effect of the power and flow oscillations in the core which become highly nonlinear and chaotic fluctuations are expected. [

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Table 4 Unmitigated Brunswick sample problem PCT results with different vessel nodalization

[ ]

Figure 28 Unmitigated Brunswick sample problem showing effect on [ ] due to vessel nodalization

[

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Figure 29 Unmitigated Brunswick sample problem showing effect on [ ] due to vessel nodalization

RAI-10

Provide an example(s) of the calculated time-dependent behavior of the [ ] during large-amplitude oscillations with flow reversal. In the cases presented in the TR, did sufficient flow reversal occur such that [ ]? If such a circumstance occurs, justify that the RAMONA5-FA ATWS-I methodology treats this circumstance in a reasonable and/or conservative way, with respect to the ATWS acceptance criteria.

Framatome Response RAI-10:

[ ]

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Figure 30 [

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Several inputs to the ATWS-I calculation are described in the TR, with specific recommendations provided. In some cases, the parameters of interest may be determined to be insensitive to specific inputs based on engineering judgment or sensitivity studies. In other cases, the parameters of interest are adjusted to achieve desired results. In all cases, the recommendations must ensure that the results from the ATWS-I calculations are accurate or conservative. In order to verify this, the NRC staff requests the following information:

RAI-11

Provide sensitivity results for one or more linear stability benchmark cases and a simulated ATWS-I event (either a nonlinear benchmark problem or a sample full-core case) by adjusting the gap conductance values. Show time-dependent results for power, PCT, and other relevant results.

Framatome Response RAI-11:

The effect of gap conductance on the reactor benchmarks is given in Table 5, [

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Table 5 Effect of gap conductance [ ] linear reactor stability benchmarks

In conclusion, it is observed that the best estimate gap conductance is the best practice for performing stability calculations in particular for ATWS-I, where biasing its value in one direction may not be always conservative.

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Figure 31 [ ]

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Figure 32 [

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Figure 33 [

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Figure 34 [

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RAI-12

The NRC staff has the following questions regarding time step control:

a. *What input parameters are provided by RAMONA5-FA ATWS-I to control the timestep size? What are the recommended values for use?*

b. *What values for these parameters were used for the nonlinear stability benchmarks and the sample problem provided in the TR?*

c. *Provide a set of sensitivity results for timestep size, similar to the sensitivity study provided for RAI11.*

Framatome Response RAI-12–a:

[ ]

Framatome Response RAI-12–b :

The values are provided in Table 6 and are common between the benchmarks and the sample problem.

Table 6 Time Step Criteria

[ ]

Framatome Response RAI-12–c:

Time step control parameters were varied to generate the time step sensitivity for several cases including the reactor benchmarks and the sample Brunswick ATWS-I run. [ ]. An example of the time step variation will be provided.

Table 7 Control parameters for time step sensitivity

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The reactor benchmarking results using these time step control parameters are provided in Table 8, [ ].

Table 8 Linear reactor benchmark results with variations of time step controls listed in Table 7

[ ]

The results of the Brunswick sample run for different time step controls are plotted together. [ ].

Table 9 Peak clad temperature results for the Brunswick sample case with varied time step control parameters

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Figure 35 Brunswick sample problem with varied time step control parameters showing [ ]

Figure 36 Brunswick sample problem with varied time step control parameters showing [ ]

Figure 37 Brunswick sample problem with varied time step control parameters showing [ ]

Figure 38 Brunswick sample problem with varied time step control parameters showing [ ]

Figure 39 Brunswick sample problem with varied time step control parameters showing [ ]

RAI-13

What spatial distribution is used for the [ ]? Justify that the modeling approach for [ ].

Framatome Response RAI-13 :

[ ]

Figure 40 Oskarshamn Core Power Results, [ ]

Figure 41 Oskarshamn Core Power Zoomed Results, [
]

RAI-14

Justify that the [

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Framatome Response RAI-14:

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The TR provides a brief procedure that would be used to perform the ATWS-I analysis and determine whether acceptance criteria are met. [

]. Therefore, the NRC staff requests the following information:

RAI-15

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Framatome Response RAI-15:

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RAI-16

For the process described in RAI 15, discuss how the various modeling and input assumptions remain appropriate when considering their effect on the time of oscillation onset.

Framatome Response RAI-16:

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Additional Information

In the topical report ANP-10346P, Section 8.0 (Calculation Procedure), Steps 3a and 3b are inconsistent.

Step 3a contains the following statement:

[ ]

A corrected page 8-2 has been added to Appendix A to reflect this change.

References

1. Y. M. Farawila and M. R. Billaux, “XEDOR -- Reduced Order Stress Model for Interactive Maneuvering of Boiling Water Reactors,” Proceedings of the International LWR Fuel Performance Top Fuel Meeting, San Francisco, September 30 - October 3 2007, Paper 1059.

2. Y. M. Farawila, K. Wei, and R. G. Grummer, "XEDOR Evaluation of PCI Risk due to Loss of Feedwater Heating in Boiling Water Reactors," Proceedings of the 2008 Water Reactor Fuel Performance Meeting, Seoul Korea, October 19-23, Paper 8142.

3. J. March-Leuba, C. G. Thurston and T. L. Huang, "Time-space nodalization issues in BWR stability calculations," in NURETH-15, Pisa, Italy, 2013.

4. Aaron J. Wysocki, “Investigation of Limit Cycle Behavior in BWRs with Time-Domain Analysis,” PhD Thesis, University of Michigan, 2015.

5. EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2*, August 2000.

6. BAW-10255PA Revision 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code*, May 2008.

7. P. Yarsky, “Sensitivity to Tmin In Trace/Parcs Analysis of ATWS with Instability,” NURETH-16, Chicago, IL, August 30-September 4, 2015

8. Yousef M. Farawila, “Fuel Design Concept to Stabilize Boiling Water Reactors,” Top Fuel 2016, Boise, ID, September 11-15, 2016

9. Letter, Jonathan G. Rowley (NRC) to Gary Peters (Framatome), “Request for Additional Information Regarding Framatome Inc. Topical Report ANP-10346P Revision 0, “ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA (EPID: L-2017-TOP-0067),” September 21, 2018.

Appendix A Revised Topical Pages

The following Appendix contains revised topical pages as discussed during the audit held at the Framatome Richland facility on June 11-13, 2018. A revised page is also included to reflect the correction shown in the Additional Information portion of this document. It is intended that these pages replace the current LTR pages in the final approved document.

6.2 *Benchmarking to Void Fraction Tests*

The FRIGG experiments, Reference 32 have been included in the validating database because of the broad industry use of these experiments in benchmarking activities, including TRAC, RETRAN, and S-RELAP5. The experiments include a wide range of pressure, subcooling, and quality from which to validate the general applicability of a void correlation. However, the experiments do not contain features found in modern rod bundles such as part length fuel rods and mixing vane grids. The lack of such features makes the data less useful in validating correlations for modern fuel designs.

Because of its prototypical geometry, the ATRIUM-10 and ATRIUM 10XM void data collected at KATHY was useful in validating void correlation performance in modern rod bundles that include part length fuel rods, mixing vane grids, and prototypic axial/radial power distributions. The characteristics of the void fraction validation database are listed in Table 6-1.

Figure 6-1 provides comparisons of predicted versus measured void fractions for the AREVA multi-rod void fraction validation database using the [ ] correlation. This figure shows reasonable predictions versus the measured data. The mean predictions fall within [ ] (predicted – measured) error with a mean error of [ ] and a standard deviation of [ ].

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6.6.3 [ ]

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6.6.4 [ ]

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6.7 *Benchmarking to Non-Linear Reactor Benchmarks*

In order to validate the functions and additions to the RAMONA5-FA ATWS-I code, several sets of non-linear stability reactor benchmarks were performed. These benchmarks include:

- Oskarshamn Non-Linear Instability
- BWR A Feedwater Transient with Non-Linear Instability

A description of the benchmark analyses is given in the following sections along with the RAMONA5-FA ATWS-I calculated decay ratios and frequencies.

6.7.1 Oskarshamn Nonlinear Instability

On February 25, 1999 as maintenance work was being performed on a switchyard outside of Oskarshamn Unit 2, the power supply to a bus bar was interrupted for 150 ms. A load rejection signal was sent to the main breaker that connects the unit to the grid. This signal caused the turbine to trip, however due to a failure in the relay circuit, this signal was not transmitted to the reactor protection system. The system continued at power until operators initiated a partial scram. Following the partial scram, power and flow were decreased. The feedwater temperature continued to decrease causing reactor power to further increase until the system crossed the stability boundary. The reactor continued to oscillate until a high power scram was reached.

The Oskarshamn-2 non-linear stability was analyzed with RAMONA5-FA ATWS-I. Two separate scenarios were evaluated. In the first scenario, the measured pump speed was imposed and the recirculation flow was calculated based on the specified pump performance curves. The plots of relevant parameters to actual measured data for this scenario are given in Figure 6-24, Figure 6-25, and Figure 6-26. The results show good agreement with measured values with just a slight underestimation of the event growth rate. However, Figure 6-26 shows that the code overestimates the final recirculation flow during the final portion of the event. A second scenario was then run in which the pump speed versus time was modified to produce a recirculation flow approximately equal to the measured value. The results of this evaluation are given in Figure 6-27, Figure 6-28, and Figure 6-29. The results of this scenario show a very good agreement with the measured power results.

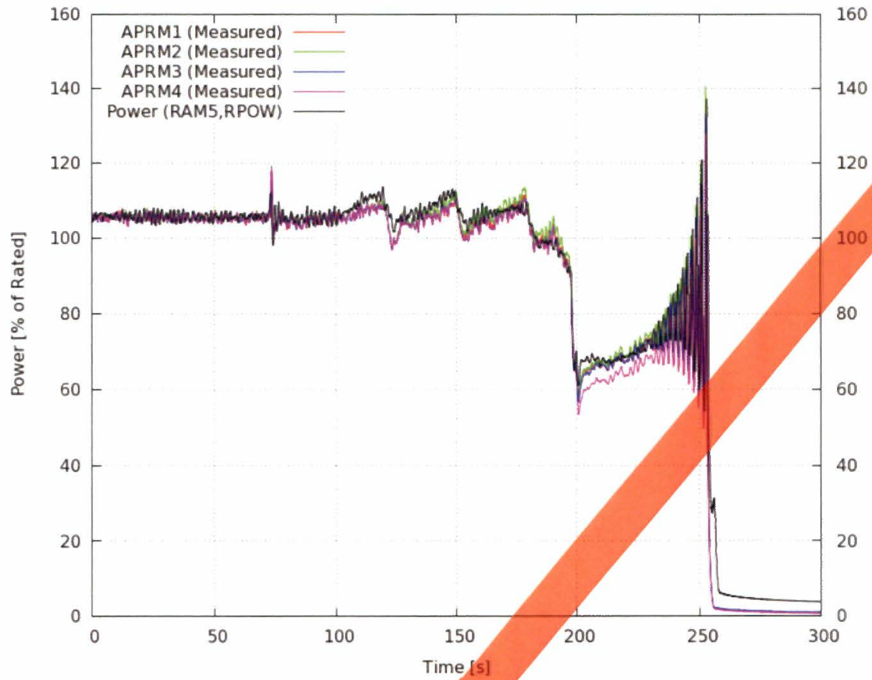

Figure 6-24: Oskarshamn-2 Non-Linear Stability Simulation Core Power, Imposed Pump Speed

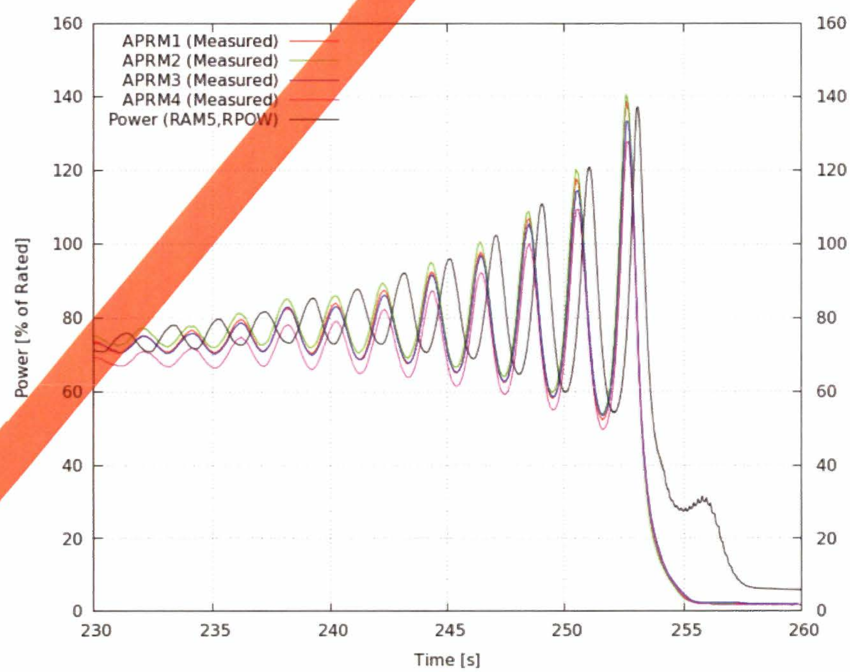

Figure 6-25: Oskarshamn-2 Non-Linear Stability Simulation Core Power - Zoom, Imposed Pump Speed

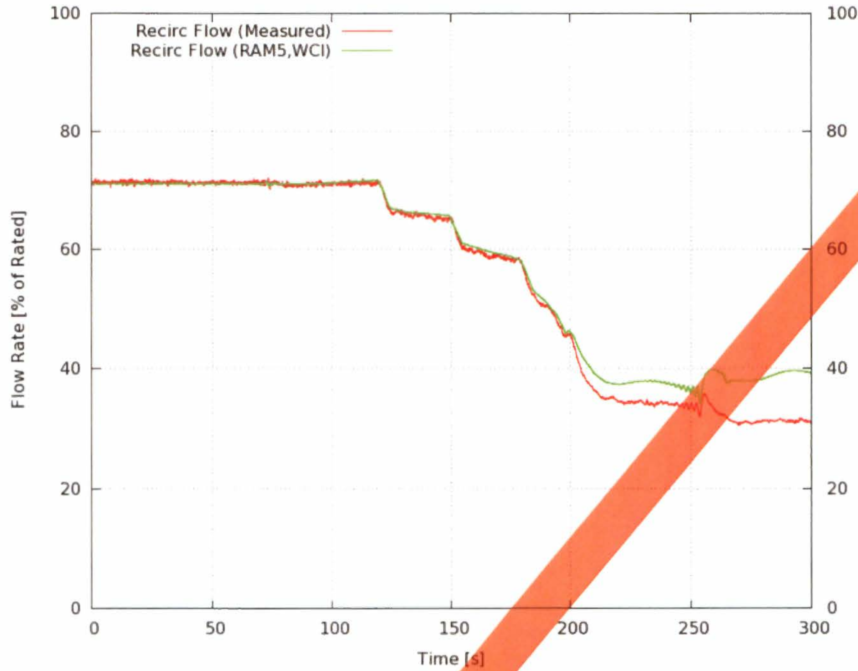

Figure 6-26: Oskarshamn-2 Non-Linear Stability Simulation Recirculation Flow, Imposed Pump Speed

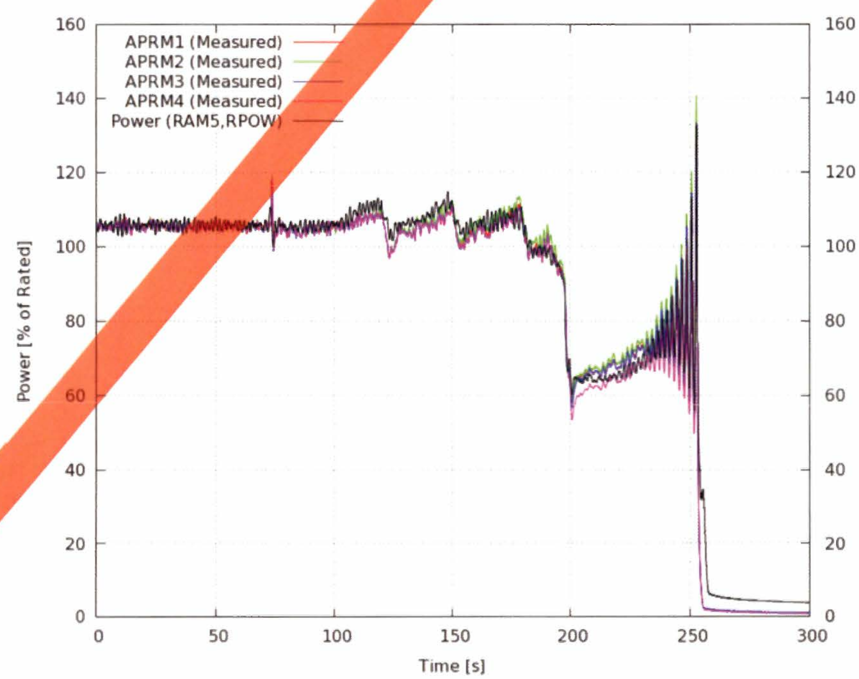

Figure 6-27: Oskarshamn-2 Non-Linear Stability Simulation Core Power, Match Recirculation Flow

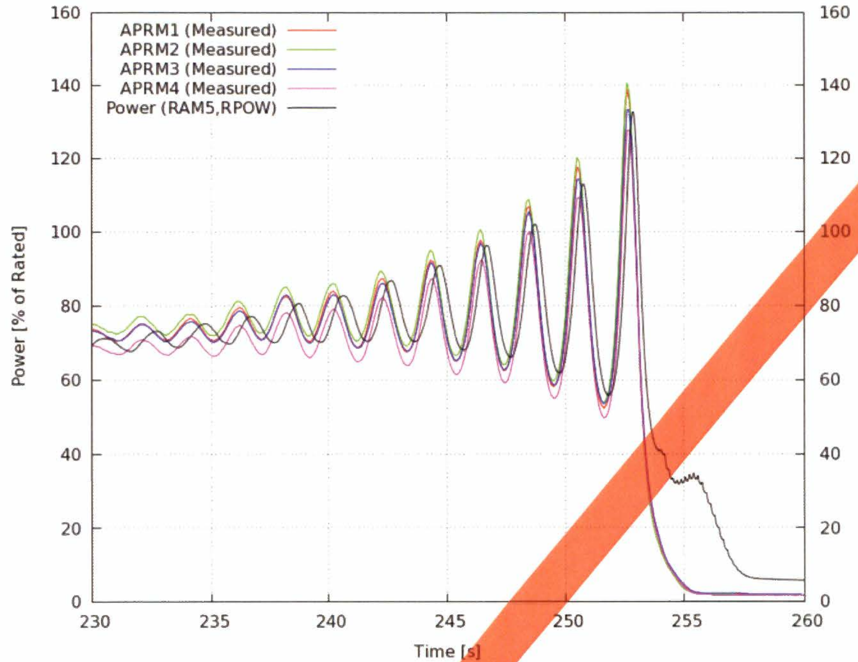

Figure 6-28: Oskarshamn-2 Non-Linear Stability Simulation Core Power - Zoom, Match Recirculation Flow

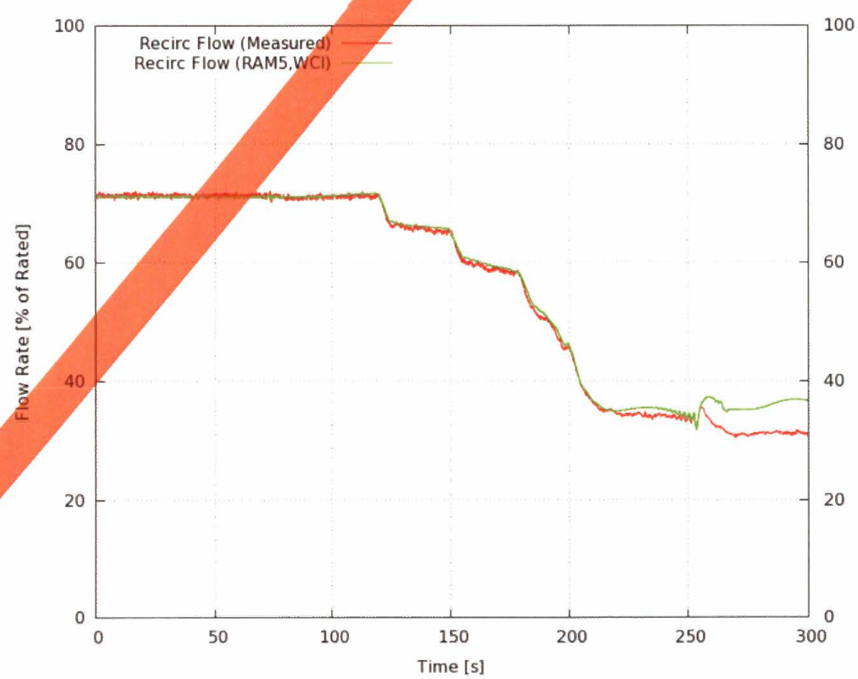

Figure 6-29: Oskarshamn-2 Non-Linear Stability Simulation Recirculation Flow, Match Recirculation Flow

Figure 6-30. The results show very good agreement between the measured and calculated [ ]

[ ]

Figure 6-30: BWR [ ] Transient Non-Linear Stability Simulation

Table 7-1: TTWB Sensitivity Study Results

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Figure 7-1: TTWB Base Case Core Power

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Figure 7-2: TTWB Base Case Core Inlet Flow

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Figure 7-3: TTWB Base Case Core Inlet Subcooling

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Figure 7-4: TTWB Base Case Limiting Channel Inlet Flow

Figure 7-5: TTWB Base Case Vessel Water Level

Figure 7-6: TTWB Base Case Limiting Channel PCT

Table 7-2: 2RPT Sensitivity Study Results

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Figure 7-7: 2RPT Base Case Core Power

Figure 7-8: 2RPT Base Case Core Inlet Flow

Figure 7-9: 2RPT Base Case Limiting Channel Inlet Flow

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Figure 7-10: 2RPT Base Case Core Inlet Subcooling

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Figure 7-11: 2RPT Base Case Vessel Water Level

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Figure 7-12: 2RPT Base Case Limiting Channel PCT

Figure C-14: Oskarshamn Core Power, Interfacial Friction Sensitivity

Figure C-15: BWR A Core Power, Interfacial Friction Sensitivity