FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE REVIEW OF TOPICAL REPORT: ANP-10300P, REVISION 0,

"AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS;

APPLICATION TO TRANSIENT AND ACCIDENT SCENARIOS"

AREVA NP, INC.

PROJECT NO. 728/DOCKET NO. 99902041

Enclosure

Table of Contents

List of Tables

1.0 INTRODUCTION

By letter dated December 23, 2009 (Reference 1), AREVA NP, Inc. (AREVA) submitted Topical Report (TR) ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios" (Reference 2), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. In June 2010, the NRC staff completed an acceptance review of the TR and found additional information was necessary (Reference 3) before a formal review effort could begin. The necessary supplemental information was submitted by AREVA in August 2011 (Reference 4). The NRC staff's review also relies substantially upon information submitted by AREVA in June 2016 in response to request for additional information (RAI) questions from the NRC staff (Reference 15). Furthermore, AREVA revised these RAI responses to address significant issues raised by the NRC staff during the review and submitted updated versions to the NRC (Reference 52). This SE is based upon all of the above submittals, including the revised RAI responses from AREVA.

TR ANP-10300P (AURORA-B) is a multi-physics, multi-code package developed for predicting the dynamic response of boiling water reactors (BWRs) during a variety of transient and accident scenarios. In conjunction with the application methodology described in TR ANP-10300P, AREVA has proposed to apply AURORA-B to a large set of transient and accident events described in Chapter 15 of the NRC's Standard Review Plan (SRP) (Reference 16) (with the exception of loss-of-coolant accidents (LOCAs), instability events, control rod withdrawal error events, control rod drop accidents, and the later stages of anticipated transient without scram (ATWS) scenarios). AREVA refers to the collection of codes within AURORA-B and the manner of their application as an "evaluation model (EM)." AURORA-B is described as being applicable to transient and accident scenarios. The TR, with the exception of the ATWS event and recirculation pump rotor seizure or shaft break accident, focuses on a specific category of transients known as anticipated operation occurrences (AOOs). AREVA intends to expand AURORA-B's analysis capabilities to include additional postulated design-basis and beyond-design-basis accident scenarios in future TR submittals.

AREVA is seeking NRC approval of ANP-10300P for all forced circulation BWR plant types over the full domain of operating conditions. This includes BWR product lines 2-6 (BWR/2-6) and the Advanced Boiling Water Reactor (ABWR), from low power conditions at which power operations commence up to and including operation at extended power uprate (EPU) conditions or expanded power and flow window (EFW) conditions (e.g., maximum extended load line limit analysis plus (MELLLA+)). This SE, however, addresses the applicability of ANP-10300P to BWRs/2-6 only. The submitted TR presents the general structure of the AURORA-B EM and its qualification for BWR analyses. In addition, an application methodology is presented that is specifically focused on the analysis of selected transient and accident scenarios. The presented EM, application methodology, and benchmarking and validation cases for the intended application range of AURORA-B for transients and accidents is evaluated in this SE.

1.1 Background

As mentioned above, AURORA-B is a multi-physics, multi-code system based on three computer codes, referred to individually as "component calculational devices" (CCDs), that have been coupled together. These CCDs are the S-RELAP5 thermal-hydraulic system code

(References 5-8), the MB2-K neutron kinetics code (Reference 9) (which incorporates input from the MICROBURN-B2 core simulator (Reference 10) to such an extent that MICROBURN-B2 is also considered a CCD of the EM), and the RODEX4 fuel thermal-mechanical code (References 11-12). Although these CCDs are codes that have individually received prior review and approval by the NRC, the NRC staff identified several areas where modifications to the codes and/or expansions to the ranges of applicability were made to accommodate the AURORA-B EM. As such, the NRC staff's review of AURORA-B was not limited to the manner in which the various codes respond when coupled, but included a focus on the areas of these changes. With regard to the individual CCDs, the NRC staff identified the following areas as needing additional review to support the present application:

S-RELAP5:

S-RELAP5 was developed as a thermal-hydraulic system code for pressurized water reactor (PWR) transient and LOCA analyses (References 5-8). Having originated as a PWR code, it was necessary for AREVA to add new models to the code and improve existing ones in order to extend its applicability to BWRs. The additions to the code include a jet pump model, pressure drop models for BWR fuel assemblies, and the installation of BWR critical power correlations. Improvements to existing models include the interfacial drag model, interfacial heat and mass transfer models, and the mechanistic BWR steam separator model. Additionally, the NRC staff identified that it would be necessary to examine many of the existing thermal-hydraulic models for applicability to BWRs, as prior review efforts had focused on the code's qualification for specific PWR applications.

MB2-K:

As the kinetics version of the MICROBURN-B2 steady-state neutronics/core simulator code, MB2-K receives a significant portion of its input from MICROBURN-B2. While MICROBURN-B2 has received prior NRC approval, MB2-K has not. Therefore, it was necessary for the NRC staff to examine the physical modeling, qualification results, and model uncertainties of the MB2-K code.

MICROBURN-B2:

The NRC staff's prior review of MICROBURN-B2 (Reference 10) did not include application to EPU and EFW conditions. Therefore, the NRC staff found it necessary to assess the code's applicability in these areas. During the acceptance review, the NRC staff identified that supplemental information (Reference 3) would be needed in order to adequately perform this assessment. AREVA supplied the necessary information in the August 2011 supplement submittal (Reference 4). AREVA also supplied additional data applicable to EFW conditions in the updated revision to this submittal (Reference 52).

RODEX4:

A subset of the routines from the previously approved RODEX4 steady-state fuel performance code (References 11-12) has been incorporated into S-RELAP5 to support evaluation of the transient thermal-mechanical fuel rod properties. AREVA refers to this subset of routines as the “RODEX4 kernel.” The RODEX4 kernel obtains a large amount of input that is used to define fuel rod initial conditions from the full RODEX4 code. For this reason, AREVA also considers the full RODEX4 code an integral component of the AURORA-B EM. Therefore, the NRC staff examined the subset of routines that comprise the RODEX4 kernel, their incorporation into S-RELAP5, and the nature of their communication with the full RODEX4 code.

In addition to the aforementioned focus areas, this SE also documents the following physical modeling areas of AURORA-B in general:

- BWR core and vessel thermal hydraulics,
- BWR core neutronics,
- BWR primary system modeling, and
- BWR fuel thermal-mechanical performance.

Pacific Northwest National Laboratory (PNNL) acted as a consultant to the NRC staff in this review, providing input for the development of this SE via a Technical Evaluation Report (Reference 13). Prior to engaging PNNL’s assistance, an acceptance review by the NRC staff resulted in a set of RAI questions that were sent by NRC to AREVA in March 2010 (Reference 3). AREVA provided a response to these RAI questions in August 2011 (Reference 4). These acceptance review RAI questions are referred to as “ARAI-##” when referenced in this SE. A second round of RAI questions were issued based on review of the subject TR and AREVA’s responses to the acceptance-review RAI questions. These RAI questions were transmitted to AREVA in February 2015 (Reference 14). AREVA provided a series of phased responses to this second round of RAI questions that concluded in May 2016 (Reference 15). These RAI questions are referred to as “RAI-##” when referenced in this SE. The ARAI questions, the RAI questions, and the responses are discussed at appropriate points throughout this SE. As noted above, AREVA intends to formally submit an updated version of both of these submittals (ARAI and RAI responses), and incorporate corresponding revisions into the approved version of TR ANP-10300P to reflect the final resolution of several outstanding issues identified during the review.

2.0 REGULATORY EVALUATION

Regulations that are applicable to the AOO and accident analysis methods presented in TR ANP-10300P, Revision 0, are found in the following sections of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities:”

- 10 CFR 50.34, “Contents of Applications; Technical Information,” which provides the requirements for the Final Safety Analysis Report required for each plant, and includes the requirements for licensees to perform analysis of transients and postulated accidents (PAs) to demonstrate safety of their facilities.

- 10 CFR Part 50, Appendix A, “General Design Criteria,” which establishes the minimum requirements for principal design criteria for the facility. In particular, General Design Criterion (GDC) 10, “Reactor Design,” requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.

- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” which requires that transient and accident analysis methods that are important to the safety of nuclear power plants be maintained under a quality assurance program.

When performing safety analyses in accordance with 10 CFR 50.34, a licensee may use a variety of methods to evaluate the AOOs and accidents postulated in the licensing basis for its nuclear power plant. The NRC staff reviews these methods to ensure that they provide a realistic or conservative result and that they adhere to applicable regulatory requirements. Events categorized as AOOs include conditions of normal operation and anticipated transients which are expected to occur one or more times during the life of the plant. Examples include, but are not limited to, tripping of the turbine generator, isolation of the main condenser, and loss-of-offsite power. A PA is a potentially limiting event which is not expected to occur during the life of the nuclear power plant, but must still be examined because of its potential to release significant amounts of radiation to the public. Examples of PAs include a spectrum of pipe ruptures in the primary system, seizure of a recirculation pump, and the dropping of a control rod blade. AREVA has submitted TR ANP-10300P, Revision 0, for review by the NRC so that licensees may reference its methods in safety analyses performed to support licensing without incurring additional NRC review of the methods.

Licensees perform simulations to demonstrate that the applicable regulatory criteria have been met, and as part of its regulatory oversight, the NRC staff has the responsibility to review these simulations. To ensure the quality and uniformity of NRC staff reviews, the NRC created NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (Reference 16) to guide the NRC staff in performing their reviews. Guidance for the NRC staff in its review of EMs like AURORA-B is provided in Chapter 15.0.2 of the SRP, “Review of Transient and Accident Analysis Methods.” Similar guidance is also set forth for the industry in Regulatory Guide (RG) 1.203, “Transient and Accident Analysis Methods” (Reference 17).

The areas of review for transient and accident analysis methods and their associated acceptance criteria are defined in Chapter 15.0.2 of the SRP. Satisfying these acceptance criteria ensures the EM satisfies the GDCs and other applicable regulatory requirements. Specifically, SRP Chapter 15.0.2 Section II indicates the acceptance criteria for the areas of review are:

(1) Documentation:

The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The submittal must also contain a complete description of the code assessment, theory manuals describing field equations and closure relationships, and user manuals discussing code limitations.

(2) Evaluation Model:

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.

(3) Accident Scenario Identification Process:

This process should identify and rank the reactor component and physical phenomena modeling requirements based on their importance to the modeling of the scenario and their impact on the figures of merit for the calculation. It must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process.

(4) Code Assessment:

Assessments against both separate effects tests and integral effects tests must be performed. All models need to be assessed over the entire range of conditions encountered in the transient or accident scenarios. Assessments must also compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models.

(5) Uncertainty Analysis:

The analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident scenario identification process.

(6) Quality Assurance Plan:

The code must be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

Generally speaking, AURORA-B represents a new method for analyzing certain BWR transient and accident scenarios. However, as explained above in Section 1.1, AURORA-B is composed of a number of CCDs, several of which have been previously reviewed on an individual basis by the NRC staff for specific applications that at least partially overlap with the application presently under review. In this sense, specific aspects of the NRC staff's review of AURORA-B can be thought of as involving small changes to existing evaluation models. As such, additional guidance for the evaluation of these specific aspects of AURORA-B may be found in SRP Chapter 15.0.2, Section III.5, which includes provisions for the review of submittals that constitute small changes to existing approved EMs. When only small changes are made, the following attributes of the EM should be considered when determining the extent to which the full review process may be applied:

(1) Novelty of the revised EM compared to the currently acceptable model: Small changes to an EM component may only need evaluation against a small subset of the entire code assessment matrix to adequately test the phenomena that are affected rather than a full review of the entire EM.

(2) The complexity of the event being analyzed: The level of effort involved in the review process should be commensurate with the complexity of the EM.

(3) The degree of conservatism in the evaluation model: The review process may be simplified if there is a large documented degree of conservatism in the evaluation model.

(4) The extent of any plant design changes or operational changes that would require a reanalysis: If the changes to plant design or operations are small, then the required scope of the review process is also likely to be small. This is because most of the changes to plant equipment or operations do not cause the plant to operate outside the range of validity of the EM.

Both SRP Chapter 15.0.2 and RG 1.203 were used by AREVA in generating the TR and by the NRC staff in reviewing the TR.

In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 and RG 1.203 in conducting its review of the AURORA-B methodology described in ANP-10300P. The review covered the areas of (1) documentation, (2) the EM, (3) the accident scenario identification process, (4) code assessment, (5) uncertainty analysis, and (6) the quality assurance plan. Additionally, where applicable as per SRP Chapter 15.0.2, Section III.5, the level of effort applied to the review of these areas was commensurate with the magnitude of the changes made to certain AURORA-B CCDs in support of the present application relative to the previous individual approvals for these CCDs and their impact on relevant phenomena.

3.0 TECHNICAL EVALUATION

The AURORA-B methodology described in ANP-10300P is intended to be a comprehensive EM for predicting the dynamic response of BWRs during certain transient, PA, and beyond-design-basis accident (e.g., ATWS) scenarios. ANP-10300P focuses largely on the qualification and applicability of AURORA-B to AOOs. However, the TR is also intended for application to the recirculation pump rotor seizure or shaft break PA and certain short-term aspects of the beyond-design-basis ATWS event. The EM contains a multi-physics, multi-code system that is intended to have the flexibility to incorporate all the necessary elements for analysis of the full range of BWR events that are postulated to affect the nuclear steam supply system of the BWR plant. Given the size and complexity of the AURORA-B EM and the wide variety of transients in ANP-10300P to which it is intended to be applied, the NRC staff first determined from a high-level the overall range of code applicability. All subsequent in-depth review efforts were then informed by this range.

3.1 Code Applicability

The AURORA-B EM presented in the TR is intended for application to domestically licensed BWR reactors up to and including operation at EPU conditions with expanded power and flow windows. The proposed range of power plant classes includes all forced circulation BWR plant types, including BWRs equipped with external recirculation pump systems (BWR/2 plants), jet-pump recirculation systems (BWR/3 through BWR/6 plants), and internal recirculation pump systems (ABWR plants). However, application to the ABWR design falls outside the scope of the present review, and, as specified in Section 5.1 of this SE, is not approved as part of this SE.

As documented in the TR, the intended range of scenarios for which the AURORA-B EM is to be applied includes AOOs, the recirculation pump rotor seizure and shaft break accidents (SRP Sections 15.3.3 and 15.3.4), and ATWS overpressurization (only up to the time of boron injection).

The applicable initiating events, as per the SRP, are as follows:

(1) SRP 15.1 Cool Down Events:

   a. SRP 15.1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature
   b. SRP 15.1.2 Feedwater system malfunctions that result in an increase in feedwater flow
   c. SRP 15.1.3 Steam pressure regulator malfunctions or failures that result in increased steam flow

(2) SRP 15.2 Heat Up Events:

   a. SRP 15.2.1 Loss of external load (generator load rejection)
   b. SRP 15.2.2 Turbine trip
   c. SRP 15.2.3 Loss of condenser vacuum
   d. SRP 15.2.4 Closure of main steam isolation valve (MSIV)
   e. SRP 15.2.5 Steam pressure regulator failure (closed)

f. SRP 15.2.6 Loss of non-emergency ac power to the station auxiliaries
g. SRP 15.2.7 Loss of normal feedwater flow

(3) SRP 15.3 Loss of Coolant Flow Events:

a. SRP 15.3.1 Recirculation pump trip
b. SRP 15.3.2 Recirculation flow controller malfunction (decreasing flow)
c. SRP 15.3.3 Reactor coolant pump rotor seizure
d. SRP 15.3.4 Reactor coolant pump shaft break

(4) SRP 15.4 Reactivity Events:

a. SRP 15.4.4 Startup of an idle recirculation loop
b. SRP 15.4.5 Recirculation flow controller malfunction which results in increased core flow rate

(5) SRP 15.5 Increasing Inventory Events:

a. SRP 15.5.1 Inadvertent operation of an emergency core cooling system (ECCS) that increases reactor coolant inventory, including high pressure core spray (HPCS), high pressure coolant injection (HPCI), or reactor core isolation system

(6) SRP 15.6 Decreasing Inventory Events:

a. SRP 15.6.1 Inadvertent opening of a pressure relief valve

(7) SRP 15.8 Anticipated Transients Without Scram:

a. Protection of the reactor pressure vessel and associated piping from failure due to over pressurization (prior to the time at which boron begins arriving at the core) by comparison to American Society of Mechanical Engineers (ASME) Service Level C limits.

b. Demonstration that fuel integrity is maintained (prior to the time at which boron begins arriving at the core)

During the review, the NRC staff generated several RAI questions that sought further description on the ranges of applicability for the AURORA-B EM that the TR did not specifically address and that are pertinent to the scenarios listed above. In the first of these RAI questions, ARAI-2, the NRC staff requested clarification of the applicability of AURORA-B to hypothetical BWR/2 operation at EPU or EFW conditions, as there are currently no EPU or EFW BWR/2 plants. AREVA responded that there are no current plans to apply the AURORA-B EM to a BWR/2 at EPU or EFW conditions. However, should the occasion arise, a review of the application of the EM to the new conditions would be performed to determine if new or expanded models are needed. If the review determines that new modeling is needed, that is not approved, the changes would be submitted either in a TR submittal or licensing amendment request. The NRC staff finds this response acceptable and has included a limitation and condition in Section 5.1 of the SE to capture this issue.

In the second of the RAI questions regarding range of applicability, ARAI-3, the NRC staff requested additional information regarding slow valve closure events. In many AOOs that are pressurization events, the more rapid the valve closure (e.g., turbine control valve), the more severe the event. However, there are some instances where slow valve closures result in a more severe transient. AREVA responded that an example of this situation is the load rejection without turbine bypass operation event at off-rated power with the power load unbalance device in the electro-hydraulic control system assumed out of service. In this event, scram on turbine control valve fast valve closure does not occur, as the valves close in the slower "servo" mode, and scram occurs on high reactor pressure. For this case, the longer time until scram occurs can result in a larger heat flux excursion and a more severe event. AREVA responded that the AURORA-B EM is designed to accommodate such scenarios as described above by utilization of the valve operational control features modeled in the S-RELAP5 CCD (e.g., as identified in the TR, Table 5-1: "Code Structure"). The NRC staff finds this response acceptable.

The AURORA-B application presented in ANP-10300P is intended to analyze ATWS scenarios, but only up until the time boron injection reaches the core. Part of the basis for determining this time frame is the assumption that maximum over-pressurization will in all cases occur in the time interval before boron reaches the core. In the third of the NRC staff's RAI questions concerning applicability, RAI-7, the NRC staff requested additional justification to support this position. AREVA responded with a detailed description of the sequence of events and assumed failures for MSIV closure without scram, which AREVA considers to be "the limiting or near limiting event for peak pressure during an ATWS." In this scenario, the peak pressure occurs **[ ]**, well before the standby liquid control system would begin injecting boron into the core. Furthermore, the AREVA response states that, were boron to be injected at the beginning of the event, the effect would be to mitigate the power excursion that drives the pressurization, thus making the event less severe. It is therefore conservative to neglect the potential effect of boron injection at any time in this scenario, with respect to the potential effect on the pressure response. The NRC staff finds this response acceptable.

In another instance, ARAI-50, the NRC staff noted the TR did not explicitly address whether the AURORA-B EM was applicable to mixed cores and, if so, how it would be applied. AREVA responded that the AURORA-B EM is indeed applicable to mixed core configurations consisting of multiple AREVA fuel types as well as AREVA and non-AREVA fuel types. When performing mixed core analyses, AREVA indicated the mixed core methodology has several principles: first, the hydraulic, neutronic and thermal-mechanical performance of each fuel type in the core is explicitly modeled; second, only like fuel assemblies are grouped together for the system model; third, critical power correlation applicability for each fuel type is explicitly demonstrated; and fourth, hot channels are explicitly modeled for each fuel type to determine cladding temperature and oxidation. Since the multiple fuel types present in a mixed core will each be explicitly modeled according to an approved methodology for mixed cores, and the parameters of interest determined for each, the NRC staff finds the response acceptable.

In RAI-84, the NRC staff requested clarification on whether or not any of the AOO events in the TR involve single loop operation (SLO). Domestic BWRs may be analyzed for SLO, which would allow continuous operation with only one of the recirculation loops forcing flow through the reactor core. AREVA responded that none of the demonstration analyses presented in the TR explicitly considered SLO, but that SLO is a valid operating mode that can be analyzed with AURORA-B, both as an initial condition and as a terminal event from normal operation. In addition, AREVA noted that the primary physical difference associated with SLO is the reverse flow condition that can exist initially in the idle loop. The reverse flow qualification of the jet pump model is included in Section 6.5 of the TR, and the NRC staff's review of the jet pump model may be found in Section 3.3.4.3.1 of this SE. The NRC staff finds AREVA's response acceptable, with the further observation that, as discussed further in Section 3.5.2.7, nodalization modifications to support modeling of SLO can significantly impact **[**

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Lastly, with respect to the application of AURORA-B in general, the NRC staff noted that Section 3.1.2 of the TR states, "Provided that the licensing basis of the plant does not significantly depart from the SRP bases, the AURORA-B EM supports the licensing basis of each plant to which it is applied, consistent with the criteria defined in the licensing basis documents for the plant." However, the TR does not address the applicability of AURORA-B to a plant with a licensing basis that *does* "significantly depart" from the SRP bases. Elsewhere in the TR, it is stated that AURORA-B is applicable to "all BWRs equipped with forced recirculation systems." This is potentially a broader class than explicitly defined in Section 3.1.2. Therefore, the NRC staff requested additional information in RAI-6 explaining 1) what constitutes a "significant departure" from the SRP bases, and 2) what procedure or evaluation steps are followed to verify that the AURORA-B EM is applicable to a given plant, with respect to the plant's particular licensing basis. AREVA responded that, for each potentially limiting selected event and operating condition, an assessment is made as to whether it can be adequately simulated by AURORA-B. If new modeling is needed that is not approved, the results of the review would be provided to the affected plant licensee(s) as part of the process for making code and methodology modifications. These changes would be submitted for NRC review and approval either in a separate TR or a plant-specific methodology submittal in support of the fuel transition license amendment request (LAR). This staff finds this response acceptable and considers it consistent with the limitations and conditions on code updates and changes specified in Section 5.2 of this SE.

Given the clarifications above, the NRC staff reviewed the AURORA-B EM for application to domestically licensed BWRs, up to and including operation at EPU or EFW conditions (with the exception of BWR/2s, per Section 5.1 of this SE), for use in performing safety analyses for the transient and accident scenarios listed above. The NRC staff notes that the TR makes mention of "further capabilities of AURORA-B," but these are considered beyond the scope of the present evaluation; the NRC staff has only reviewed the capabilities of AURORA-B for the requested applications.

3.2 Documentation

The first of the review areas discussed in SRP Chapter 15.0.2 for transient and accident analysis methods is documentation. The associated acceptance criteria indicate that the submittal must identify the specific accident scenarios and plant configurations for which the code will be used. The submittal must also contain a complete description of the code assessment, theory manuals describing field equations and closure relationships, and user manuals discussing code limitations.

The TR does not have a stand-alone theory manual. Instead, the theory discussion is spread among several documents, some of which have been previously reviewed and others that have not been reviewed. These documents cover the four main phenomenological areas encompassed in AURORA-B: thermal hydraulics, neutronics, systems modeling, and thermal-mechanical fuel performance. As a result, it became necessary for the review to encompass a large number of additional documents supporting and supplementing the TR.

The principal documents where the theory is discussed, including field equations and closure relationships, are found in References 5-7, 9-11, and 18-20. The NRC staff consulted each of these documents throughout the review of AURORA-B and concluded they contained sufficient information regarding the field equations and closure relationships of the models contained within each of AURORA-B's CCDs. The NRC staff found that many of these documents also discuss limitations in the CCDs, either self-identified by AREVA or identified within the associated NRC staff SEs.

In addition to the principal documents referenced above, a number of background documents were consulted to obtain a complete picture of the AURORA-B EM and its verification. These documents supplement the information provided in the principal documents and are found in References 8, 12, and 21-28.

The NRC staff finds that the documentation discussed above sufficiently describes the AURORA-B EM assessment, supplies the field equation and closure relationships within each of the CCDs, and discusses code limitations. The identification of applicable accident scenarios and plant configurations for the AURORA-B EM is primarily presented in the TR and is discussed in Section 3.1 of this SE.

3.3 Evaluation Model

Following the review guidance provided in Chapter 15.0.2 of the SRP, the second area of review for transient and accident analysis methods focuses on the EM. 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 SRP 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.

As discussed earlier, AURORA-B is a multi-physics, multi-code system based on four computer codes, referred to as CCDs, packaged together for the purposes of predicting the dynamic response of BWRs during transient and accident scenarios. These CCDs are the thermal-hydraulic code S-RELAP5, the neutron kinetics code MB2-K (which incorporates input from the MICROBURN-B2 core simulator to such a large extent that MICROBURN-B2 is also considered a CCD of the EM), and the fuel thermal-mechanical code RODEX4.

Summarizing each of the CCDs:

- S-RELAP5 serves as the backbone of the EM. S-RELAP5 provides the steady-state and transient thermal-hydraulic, thermal conduction, control systems, and special process model capabilities (e.g., valves, jet-pumps, steam separator, critical power correlations, etc.) necessary for the thermal-hydraulic analysis of a BWR plant. The starting point for the S-RELAP5 CCD used in AURORA-B was the S-RELAP5 code version associated with the PWR Realistic Large Break Loss-Of-Coolant Accident (RLBLOCA) methodology (Reference 7). Therefore, it was necessary for AREVA to make extensions and improvements to this code version for applicability to BWRs (Section 1.1).

- MB2-K uses advanced nodal expansion methods to solve the three-dimensional (3D), two-group, neutron kinetics equations. The MB2-K code is consistent with the MICROBURN-B2 steady state core simulator (Reference 10) and receives a significant portion of its input from the steady state core simulator via data transfer files. Because of this, although it is not directly coupled to S-RELAP5, MICROBURN-B2 is also considered a CCD of the AURORA-B EM. **[**

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- MICROBURN-B2 is a steady state core simulator code that solves the two-group neutron diffusion equation based on the interface current method (employing the advanced nodal expansion method), and calculates nuclide density distributions (heavy nuclides and burnable poisons). It also determines the nodal power and pin power distributions, bundle flow, and void distributions. For a given set of initial conditions and cycle exposure, MICROBURN-B2 supplies the steady-state core parameters that are necessary for the simulation of a transient or accident scenario.

- RODEX4 is a steady-state fuel performance code (Reference 12). A RODEX4 kernel consisting of a subset of routines from the full code is incorporated into S-RELAP5 and is used to evaluate the transient thermal-mechanical fuel rod (including fuel-clad gap conductance) properties as a function of temperature, rod internal pressure, etc. These fuel rod properties are used by S-RELAP5 when solving the transient thermal conduction equations in lieu of standard S-RELAP5 material property tables. The RODEX4 kernel receives a significant portion of its input from the full RODEX4 steady state fuel performance code via data transfer files. **[**

] consistent with the S-RELAP5 heat structure conditions as a calculation evolves.

Together, these CCDs model the four phenomenological areas of the AURORA-B EM: BWR core and vessel thermal-hydraulics, BWR core neutronics, BWR primary system modeling, and thermal-mechanical performance for BWR fuel. The NRC staff reviewed each of the main modeling areas while maintaining a focus on the specific areas identified as needing review within each of the individual CCDs (Section 1.1). Each of the modeling areas is discussed in turn within the subsections that follow.

3.3.1 BWR Core and Vessel Thermal-Hydraulic Models

The thermal-hydraulic models used in the AURORA-B EM are contained in the S-RELAP5 CCD. As described in Section 1.0 of the TR, S-RELAP5 "provides the transient thermal-hydraulic, thermal conduction, control systems, and special process capabilities (e.g., valves, jet pumps, steam separators, critical power correlations, etc.) necessary to simulate a BWR plant." Many of the models and correlations in S-RELAP5 are not the subject of the current review as these have been approved in previous submittals for sufficiently similar applications (References 5-8). The exceptions to this, which are primarily related to BWR-specific system components (e.g., steam separators and jet-pumps), are noted in Section 3.3.4. In this review of the AURORA-B EM for BWRs, the thermal-hydraulic models of S-RELAP5 are evaluated only insofar as to examine their appropriateness for assessment of the application of the AURORA-B EM to BWR plants for the set of events within the scope of this TR (described in Section 3.1).

The capabilities of the EM that are relevant to thermal-hydraulic modeling are identified in Section 3.3 of the TR. **[**

]

The assessment base developed for the evaluation of the EM is described in Section 4 of the TR. The stated objective of the assessment base is to select experimental data to assess whether the EM possesses the capabilities specified in Section 3 of the TR (Section 3.1 of this

SE). For this purpose, BWR components, processes, and phenomena to be assessed were identified and ranked according to their expected importance, relative to defined figures of merit that are described further in Section 3.4.3. Table 4.1 (on page 4-3 of the TR) shows the EM assessment matrix for the identified highly ranked phenomena. Specific items in this table that are relevant to the necessary thermal-hydraulic modeling capabilities identified above are described in Table 3.1 below. Items from the EM assessment matrix in Table 4.1 of the TR related to the recirculation system, steam separators and dryers, and other components above the core or within the primary system pipeline, are discussed in review of the EM integral assessment (see Section 3.5 of this SE). Also note that items **[** **]** in Table 4.1 of the TR are omitted in Table 3.1 below. These items are **[** **]** and are discussed in Table 3.2 of Section 3.6.1 in this SE.

Section 5 of the TR describes the overall structure of the EM and how it was developed. The review of topics relevant to this section is documented in Sections 3.3.4 and 3.5 of this SE. The assessment of the EM adequacy, as defined by model performance in comparison to experimental data, is presented in Section 6 of the TR. Therefore, the information presented in Section 6 of the TR is the main focus of the thermal-hydraulic portion of this review.

Table 3.1 Summary of Assessment Basis Identified for Thermal-Hydraulic Modeling

[ ]

The following subsections provide details on the general assessment of thermal-hydraulic capability (Section 3.3.1.1); assessment of void fraction prediction capability in rod arrays, herein referred to as combined effects tests (Section 3.3.1.2), and in simplified geometries, herein referred to as separate effects tests (Section 3.3.1.3); assessment of rod bundle pressure drop behavior prediction capability (Section 3.3.1.4); and assessment of boiling transition prediction (Section 3.3.1.5).

3.3.1.1 General Assessment of Thermal-Hydraulic Capability

The data sources used in the assessments documented in Section 6 of the TR are in many cases vetted and respected. There is no question about the integrity of the work; much of it constitutes the foundation of what is known today about two-phase flow and heat transfer behavior in general and in rod arrays in particular. It is appropriate for AREVA to have relied on it for a significant part of the assessment of the AURORA-B EM. However, there is no getting around the fact that much of this data is dated. In many of the primary documents, the work reported is specifically described as preliminary, particularly in regard to general conclusions about two-phase flow behavior in geometries other than the limited configuration(s) tested. The authors in many cases express the confident expectation that additional work would be undertaken in the future to explore questions that the existing work was unable to address or that were generated in the process of evaluating the results of the testing. They did not, in general, expect their work to be complete, but rather hoped that it would be expanded upon, increasing the range and scope of understanding of the complexity of two-phase flow thermal-hydraulics, particularly in regard to behavior in rod arrays and steam separators.

There are two main issues with reliance on "old data" in the assessment of the EM as presented in Section 6 of the TR. The first is the limited range and scope of the experimental testing, particularly in regard to geometry, for both separate effects tests in simplified single channels and integral testing in rod arrays. The second is the limited attention generally given at the time of testing to evaluation of overall experimental uncertainty and potential biases. Many of the older data sets present meticulous evaluations of measurement uncertainty with regard to the instrumentation used in the tests, but seldom provide information that could be used to assess overall experimental uncertainty. At the time much of this work was done, instrumentation uncertainty was considered to be the most significant source of error in testing. Replicate points were seldom captured, generally to maximize the range of testing, and the development of statistical analysis to capture overall experimental uncertainty had not even begun.

The review of the treatment of uncertainties in results obtained from application of the EM is documented in Section 3.6 of this SE. Therefore, this discussion of the assessment of the core and vessel thermal-hydraulic models, as presented in Section 6 of the TR, focuses on the first of the two issues discussed above: the range and type of data provided to perform the assessments. The reliance in the TR for verification of models on the basis of comparisons to older data sets with limited ranges of geometry and other test conditions is concerning in light of the increasing geometric complexity of fuel assemblies (due to the increasing size of the rod array), the increasing variation in **[**

]. A major effect of many of these changes in design is **[**

] in some assembly designs for ostensibly the same average conditions across the assembly at a given location.

In applications of the EM, the basic thermal-hydraulic modeling approach in S-RELAP5 represents a fuel assembly [

] However, this does not increase confidence in the void fraction models themselves, [

]

Assessments relevant to the thermal-hydraulic modeling capability of the AURORA-B EM are presented mainly in Sections 6.2, 6.5, and 6.6.2 of the TR. In general, the descriptions of the experimental data sources and presentation of data comparisons to support the reported assessments in these subsections are sparse and limited in scope. As a result, the NRC staff found the information in the TR insufficient to permit appropriate review of AREVA's thermal-hydraulic modeling capability assessments and the conclusions drawn therefrom in the TR. Therefore, the NRC staff requested that AREVA supply the primary references discussed in Section 10 of the TR (specifically, TR References 24, 26 through 31, 33, 34, 43, 44, and 47 through 52), in order to support appropriate, thorough technical review of this portion of the TR. In addition, some assessments were presented in the TR that did not reference the source of the experimental data they used. These references were requested in RAI-96, which also requested that the uncited references be added to the formal list of references in the TR.

In the response to RAI-96, AREVA stated, "It is not AREVA practice to cite internal calculation notebooks in TRs. Instead, AREVA maintains a Source Reference Record (SRR) associated with the TR that provides the cross-reference between the TR contents and the internal calculation notebooks. The internal calculation notebooks are available for audit." During an audit (Reference 29), AREVA provided the specific references noted above plus a number of proprietary calculation packages that contained documentation of the unreferenced data. The information obtained via the audit was sufficient to support the NRC staff's technical evaluation of the TR. Additionally, the NRC staff had the opportunity to examine the SRR and discuss the nature of the document with AREVA staff. To be specific, a separate SRR exists for each of AREVA's TRs, which provides a complete and accurate list of references for all data and contents presented in its associated TR. This includes calculation notebooks and proprietary data sources in addition to freely citable sources. Part of the NRC staff's impetus for asking RAI-96 was to ensure that an accurate paper trail would exist in support of the conclusions reached within the SE. The NRC staff finds that, having examined the SRR for ANP-10300P and documented the reviewed contents within an audit report, this objective has been met. Therefore, the NRC staff finds AREVA's response to RAI-96 acceptable.

The discussion of the approach used in the EM to determine the event minimum critical power ratio (MCPR) (the MCPR experienced during a particular transient event) in transient calculations, as presented in Section 5.2.9.1 of the TR, was found to be insufficient to perform an appropriate review. As discussed further in Section 3.4.3, the event MCPR is a significant figure of merit used for evaluating AOOs to ensure SAFDLs are not exceeded. Thus, the NRC staff issued RAI-9 and RAI-10 to request the omitted information. AREVA responded by providing an expanded and detailed discussion of the process and underlying methodology for determining the event MCPR, justifying the assertion that the methodology could be expected to reliably determine the potentially most limiting fuel assembly in the core during applicable transients. The NRC staff finds this response to be acceptable.

There is a similar ambiguity in the discussion presented in Section 9.3 of the TR, which defines an **[ ]** in three particular figures of merit (FoMs) (specifically, delta MCPR (ΔMCPR), peak system pressure, and maximum peak clad temperature, as discussed in Section 3.4.3 of this SE) due to a code change or input parameter changes in a sensitivity analysis. The TR stipulates that such analyses are **[ ]** but does not define how that might be determined, or **[ ]**. In response to RAI-18 on this issue, AREVA provided clarification indicating that **[ ]** AREVA further clarified in its response to RAI-18 that, **[ ]** While the NRC staff finds this approach reasonable, initial conditions should be set as close as possible to the expected limiting conditions for the event being analyzed. Deviations from expected limiting event conditions will be subject to review during the NRC staff's review of plant-specific applications.

Despite the issues discussed above that come with reliance on "old data," when this data was considered alongside the additional information and data supplied by AREVA in response to the NRC staff's questions, the NRC staff was able perform an informed assessment of the AURORA-B EM's thermal-hydraulic capabilities. Specific areas of the NRC staff's assessment and any applicable concerns stemming from the selected qualification data are discussed in the following sections.

3.3.1.2 Assessment of Void Fraction Prediction Capability: Rod Arrays

To address the assessment of the highly ranked Phenomenon Identification and Ranking Table (PIRT) (discussed in Section 3.4.2) phenomena related to **[** **]** that are identified in Table 3.1 above – specifically, **[** **]** – the TR presents comparisons of model predictions to measured void fraction data. The measured data selected for this evaluation consists of rod bundle testing performed at two experimental facilities:

- the FRIGG test facility in Sweden, from two test series (FRIGG-2 and FRIGG-3) using electrically heated rod arrays representing the Marviken reactor fuel assembly design (testing performed in 1968 and 1969), and

- the KATHY test facility in Germany, using an electrically heated 10×10 rod array designated ATRIUM-10A, representing a modern, AREVA-designed BWR assembly with part-length rods (date the actual testing was performed is not specifically documented, but it can be **[** **]**).

Limited description and background on these tests is presented in the TR (Section 6.2.1). The assessment is accompanied by only three plots, which compare measured and predicted void fractions (Figures 6-1, 6-2, and 6-3 in the TR) without connecting any particular measurement to any particular test. This presentation of results did not provide information that would permit evaluation of the data comparisons in terms of the range of BWR operating conditions (e.g., flow rate, pressure, and inlet subcooling) over which the EM will be applied, and it did not provide information for assessment of the capability of the EM to predict the axial distribution of void fraction along the length of a rod array. In addition, the TR did not address sources of uncertainty other than instrumentation uncertainty, even though it acknowledged that the instrumentation uncertainty does not capture the total uncertainty in the experimental data.

Therefore, the NRC staff issued RAI-29a, which specifically requested additional information in the form of calculated-to-measured comparisons of the axial distribution of void fraction for individual tests in the FRIGG and ATRIUM-10A databases. The comparisons were to span typical BWR operating ranges of flow rate, pressure, and inlet subcooling, insofar as possible within the range of the databases. In response to RAI-29a, AREVA provided axial void distribution plots comparing S-RELAP5 predictions to the measured data for a sampling **[** **]**.

The **[** **]** from the FRIGG-2 database are all at the same pressure (approximately (~) 725 pounds per square inch absolute (psia)), as this was the only pressure tested in this series. The reported mass flow rates for the **[** **]**. The range of inlet subcooling values for the **[** **]**. The maximum measured void fraction in these tests (presented in the response to RAI-29a in Figures 29-1 through 29-7) is shown to be **[** **]**. The predicted void fraction axial profiles for the selected tests from the FRIGG-2 database, as shown in Figures 29-1 through 29-7 of the response, follow the profiles indicated by the measured data **[** **]**. However, when taking into account the fact that the uncertainty bands on the measured data are likely somewhat understated, the agreement between the model results and the data **[** **]**. As such, the NRC staff considers the S-RELAP5 predictions to be in **[** **]** agreement with the measured data from the FRIGG-2 database.

The **[** **]** from the FRIGG-3 database include selections from the **[** **]** different pressures tested in this series. Of the selected tests, **[** **]**. The range on mass flow is **[** **]** of flow rates tested. The selected tests **[** **]** of inlet subcooling values tested, **[** **]**.

For the selected FRIGG-3 tests, the RAI-29a response presents plots (in Figures 29-8 through 29-18) showing void fraction predictions from S-RELAP5 compared to measured void fraction profiles reported for these selected tests. The maximum measured void fraction in these tests is shown to be **[** **]**. The agreement between measured and predicted void fraction axial profiles for the selected tests from the FRIGG-3 database is consistent with the agreement shown for the FRIGG-2 data: the uncertainty bands **[** **]**. The NRC staff therefore considers the S-RELAP5 predictions to be in reasonable to excellent agreement with the measured data from the FRIGG-3 database.

The comparison of measured-to-predicted void profiles shown in Figures 29-17 and 29-18 exhibit **[** **]** (the more recent S-RELAP5 code version, **[** **]**, was generated, in part, to address **[** **]**, and is discussed in Section 3.5.4.3 of this SE). **[** **]** However, the shift is relatively small, is quite likely within the actual uncertainty of the measured data, and **[**

] in this flow regime range.

In general, it can be concluded that the comparisons with the FRIGG-2 and FRIGG-3 databases show that the EM is capable of matching this void fraction data. This is not unexpected; these older datasets are part of the foundation that nearly all existing void models are built upon. However, these databases are from 1960s-vintage test sections with geometry and spacer grid designs being considered for the Marviken reactor. That is, circular tubes with 36 rods on a triangular pitch, tested at uniform axial power distribution only. These test sections are not prototypic of modern BWR square-array fuel assemblies. From the standpoint of assessing the suitability of the AURORA-B EM for analysis of currently operating BWRs loaded with modern fuel designs, the comparison to the test data from the ATRIUM-10A database is more relevant.

Compared to the FRIGG databases, the ATRIUM-10A database is **[** **]**. The ATRIUM-10A database consists of **[** **]**. The range of flow rates tested **[** **]**, and the range of measured void fractions **[** **]**. However, measurements of void fraction were **[** **]**

As a result, the NRC staff requested in RAI-29a additional calculated-to-measured comparisons of **[** **]** in the databases supporting the assessment of the EM as presented in the TR. AREVA's response provided point-by-point tabulated values **[** **]** in the database (see Table 29-4 in the response to RAI-29a). **[** **]**
Plots of measured-to-predicted values for this selected subset of the database (Figures 29-19 through 29-24 of the RAI responses (Reference 15)) show that the model predictions are in good agreement with the measured data and within reasonable estimates of the uncertainty.

The assessment presented in Section 6.2.1.2 of the TR (see Figure 6-3) appears to show that **[** **]** in the EM predictions for the ATRIUM-10A database in comparison to the results shown for the FRIGG databases (see Figures 6.1 and 6.2 of the TR). This suggests that the void models in the EM **[** **]**. In response to RAI-29c, requesting discussion of the implications **[** **]**. According to the response, the main reason for **[**

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The NRC staff also noted that, as part of the response to RAI-29e, AREVA discussed how, for Figures 6-1 through 6-3 in the TR (FRIGG2, FRIGG3, and ATRIUM-10A void fraction results), **[** **]**. The NRC staff does not agree with this interpretation. In particular, where the measured data points lie outside the prediction uncertainty bands, the NRC staff concludes these data are more likely to fall outside the prediction uncertainty bands, even when measurement uncertainty is applied. Additionally, the interpretation of **[** **]** allows for an exploitable interpretation of acceptably predicted data; a data result with a **[** **]** measurement uncertainty could be counted as acceptably predicted.

A plot of calculated-to-measured void fractions **[** **]** the ATRIUM-10A database (Figure 29-31 of RAI-29e) shows **[** **]**. The apparently **[** **]** for the ATRIUM-10A database **[** **]** is greatly reduced; **[** **]**. Since many of the data points from the original ATRIUM-10A database **[** **]**. There does, however, appear to be an **[** **]**. Whether this is an actual trend resulting from the models employed in S-RELAP5 or an artifact **[** **]**

The NRC staff concludes that the evaluations presented in the TR and the responses to RAI questions that expand the discussion of the data comparisons and evaluations show that the AURORA-B EM can be expected **[** **]** of void distribution in rod arrays. However, the assessment in the TR of the void fraction models in comparison to measured void fraction data from rod arrays **[** **]**. Given an apparently **[** **]** of the ATRIUM-10A database and **[** **]** used in the evaluations in the TR. In light of these points, the nature of AREVA's definition regarding an acceptably predicted datum point, and because **[** **]** highly ranked PIRT phenomenon (Section 3.6.4.15), the NRC staff concludes that before new fuel designs are modeled in licensing analyses using the AURORA-B EF, AREVA should justify that the AURORA-B EM can acceptably predict void fraction results for the new fuel designs **[** **]**.

The assessment of uncertainty in the predictions of the EM is addressed in detail in Section 3.6 of this SE.

3.3.1.3 Assessment of Void Fraction Prediction Capability: Simplified Geometries

In addition to the comparisons with void fraction data from testing in rod arrays (discussed above in Section 3.3.1.2), the TR presents comparisons of model predictions to data from a number of other geometries. These comparisons are also presented for the purpose of assessing the ability of the EM to appropriately model the highly ranked PIRT phenomena identified in Table 3.1, above. Specifically, the Christensen single-channel two-phase flow tests are used to assess **[** **]**, while the collection of Allis-Chalmers two-phase flow tests in large diameter pipes are used to assess the **[** **]**. Additionally, transient tests simulating downcomer level swell, performed by General Electric Company (GE) as part of the BWR Refill-Reflood program, are used to assess **[** **]**.

The Christensen void tests were designed as a preliminary investigation of a relatively new (in 1961) area of research, driven primarily by the need to develop a better theoretical and practical understanding of two-phase flow in BWRs. The description of the Christensen void tests in the TR (see Section 6.2.2) overstates the range and general applicability of this dataset. It consists of 7 tests performed in a non-prototypic geometry (single channel with rectangular cross-section) to investigate the potential effect of power oscillations on void formation and collapse.

The 7 tests were performed at three different pressures, two inlet flow rates, and inlet subcooling at approximately four different values, ranging from 3 degrees Celsius (°C) (5 °F) to 14 °C (25 °F). The expected two-phase flow regimes tested include bubbly flow, probably slug flow, and possibly churn-turbulent (but with void fractions less than 60%) and did not extend to annular flow. This testing range does not constitute a "wide range" of conditions covering "typical operating BWR conditions," as asserted in the TR.

The assessment in the TR makes use of **[**

] The code-data agreement captures all the major and minor trends of the data fairly well, thereby indicating S-RELAP5 can be expected to reasonably predict the phenomena associated with subcooled boiling.

While the TR makes use of only three of these seven tests for assessment of subcooled boiling, it does utilize all seven tests to show a predicted-to-measured plot for void fraction (Figure 6-4). This plot shows that the model results are **[** **]**. However, the plot also demonstrates that the results **[** **]** based solely on instrumentation uncertainty. The plot actually suggests that the prediction uncertainty **[**

]. The primary documentation suggests the experimental uncertainty is much larger than the instrumentation uncertainty. Because of difficulties in controlling the power oscillations, the uncertainty in the local surface heat flux is relatively large for any given test at any given point in the time period required to take the multiple gamma scans that constituted a given test. A significant conclusion reported in the primary documentation is that local void fraction seems to be relatively insensitive to power fluctuations, since there was no evidence of significant void collapse in the channel at the low point in the power oscillation. However, this may have been due in large part to testing with oscillations of relatively small magnitude, while maintaining a constant average power over the time interval of a given test.

The NRC staff finds that given **[** **]** between the Christensen void test data and the S-RELAP5 model results presented within the TR. As noted in regard to the FRIGG data sets discussed in Section 3.3.1.2, this is not an unexpected result. The Christensen void tests have become a touchstone within the nuclear industry for evaluation of the void modeling of two-phase thermal-hydraulic behavior in a boiling channel.

The assessments in the TR based on the Allis-Chalmers large diameter void tests (performed primarily in 1964) make use of data obtained primarily for industrial prototyping of steam separator performance and consist mainly of investigations of the flow of steam-water mixtures in large diameter piping. This work was funded by the U.S. Atomic Energy Commission (AEC) as part of the Euratom Program and was carried out by Allis-Chalmers, Atomic Energy Division. Some of the testing (in 2.9-inch diameter piping) used gamma densitometer measurements to determine void fraction, but most of the data, particularly in the larger diameter test vessels (18-inch and 36-inch diameters) did not involve measurement of the void fraction but instead inferred this quantity indirectly from pressure drop measurements using the homogeneous two-phase flow void model. The two-phase flow data obtained in these tests represents primarily the bubbly, slug, and (for some tests) the churn-turbulent flow regimes. None of the tests were at high enough void fraction to reach annular flow conditions.

The referenced documents for these tests (References 29, 30, and 31 from the TR) do not provide sufficient information to estimate the uncertainty in the void fraction data from these tests. The TR estimates the uncertainty as **[**

]. In response to RAI-31, requesting clarification on this **[** **]**, AREVA provided additional details on this **[** **]**.

Review of the responses to RAI-23 and RAI-31, and of the primary reference for the **[** **]**, shows that the approach used by AREVA for this assessment is somewhat circular and does not convincingly support the conclusion that the uncertainty in the measured data **[** **]**. The uncertainty of the **[**

]

The most that can be inferred from the **[** **]** is that the uncertainty in the Allis-Chalmers void fraction data **[** **]**, but the evaluation has no means of capturing any possible biases in this data. Since the void fraction data was calculated from pressure drop measurements using the homogeneous void model, it *ipso facto* incorporates the inherent bias of this relatively simplistic model of two phase flow. There is not sufficient information in the primary documentation (i.e., References 29, 30, and 31 in the TR) to quantitatively assess this bias, but at the relatively low void fractions tested, it is reasonable to assume that the bias is adequately captured in the overall **[** **]**. The highly ranked PIRT phenomena for which the Allis-Chalmers void fraction data is invoked for support are **[** **]**. The NRC staff evaluated AREVA's treatments of the prediction uncertainties for these two phenomena given a measurement uncertainty **[** **]** for the Allis-Chalmers void fraction data and found the approaches acceptable. The NRC staff's evaluations are found in Section 3.6.1 of this SE.

The assessment of **[** **]**, shows EM predictions for one test of seven reported for the smaller of two vessels in the BWR Refill-Reflood Program (see Reference 25 of the TR). This test was selected for presentation in the TR because it was used for code assessment in the RLBLOCA methodology assessment (see Reference 3 of the TR). Plots of predicted void fraction profiles at two different times in the transient (40 seconds and 100 seconds) are shown in Figures 6-9 and 6-10 of the TR. These plots show reasonable agreement with the measured data, and the TR infers from this that the S-RELAP5 modeling for interfacial heat transfer for depressurization phenomena has been "adequately validated" by this comparison.

As noted above in Section 3.3.1.2 for the evaluations in the TR with data from rod arrays, the NRC staff finds these separate effects tests show that the AURORA-B EM can be expected to give reasonable predictions of overall two-phase flow behavior in a BWR vessel for components other than the fuel assemblies. The assessment of uncertainty in the predictions of the EM is addressed in detail in Section 3.6.

3.3.1.4 Assessment of Rod Bundle Pressure Drop Prediction Capability

To address the assessment of the highly ranked PIRT phenomena **[ ]**, which is of primary importance in determining BWR core thermal-hydraulic behavior, the TR presents comparisons of model predictions to measured pressure drop data obtained for a range of BWR fuel designs, including 7×7, 8×8, 9×9, and 10×10 rod arrays. This data includes testing for two generic grid spacer designs and also spans a range of axial power shapes (see Section 6.5.1 of the TR). The majority of the data was obtained with **[ ]**. The measured data selected for this evaluation consisted of tests performed at two experimental facilities:

- **[ ]** and
- **[ ]**

The TR discusses this assessment in less than three pages (see Section 6.5.1 of the TR), presenting an evaluation **[ ]** in three plots (Figures 6-18 through 6-20), and one table **[ ]** (Table 6-10). No specific references are cited in the TR for the source(s) of this data, and the documentation in the TR was found to be insufficient to perform an appropriate technical review. Therefore, five RAI questions (specifically, RAI questions 32-36) were generated asking for expanded discussion and additional information on a number of points. RAI-32b was generated to address the impression that the plots in Figures 6-19 and 6-20 show that **[ ]**.

Additional information supporting the assessment and describing in greater detail the evaluations performed was provided by AREVA in the responses to these RAI questions. AREVA also provided proprietary calculation package documents that were not referenced in the TR. This allowed an appropriate technical review of the assessment of the capability of the EM to appropriately predict rod bundle pressure drop. In the response to RAI-32b, AREVA reported that a closer examination of **[**

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The additional information provided in the response to these RAI questions, and the expanded discussion of the approach used for **[ ]**, was sufficient for the technical review of this portion of the TR.

The NRC staff review concluded that the assessment (as expanded) does indeed show that the EM is capable of predicting rod bundle pressure drop in a BWR core. Assessment of the uncertainty in the EM predictions of bundle pressure drop relative to the highly ranked PIRTs is discussed further in Section 3.6.

3.3.1.5 Assessment of Boiling Transition Prediction

The assessment of the capability of the AURORA-B EM to appropriately predict critical heat flux in fuel assemblies within the core is presented in Section 6.5.4 of the TR. The TR identifies four highly ranked PIRT phenomena (see Table 3.1 of this SE) as being addressed in Section 6.5.4: **[**

]. Thermal-hydraulic behavior within a rod array relative to all four of these PIRT phenomena is highly dependent on fuel assembly design, and prediction of this behavior is highly dependent on the CPR correlation determined for that particular design. Although virtually any CPR correlation may be included in the AURORA-B EF, the licensing basis for a specific CPR correlation applicable to a specific fuel design is evaluated separately and is not part of the technical evaluation of the AURORA-B EM in this SE.

The focus of the assessment presented in Section 6.5.4 of the TR is on the capability of the EM to appropriately evaluate the ΔCPR response of fuel assemblies in transients; particularly, the Chapter 15 AOOs defined for BWRs. The TR presents assessments consisting only of examples illustrating performance of the AURORA-B EM in transient analyses for two specific fuel assembly designs and their associated CPR correlations. The assessments presented are for the SPCB CPR correlation used with the ATRIUM-10 fuel design and the ACE CPR correlation used with the ATRIUM-10XM fuel design.

The comparisons with transient boiling transition data presented for these two correlations in the assessment support the conclusion that the modeling approach yields conservative results in most cases. However, in some cases the modeling results shown are non-conservative; the model does not predict boiling transition for conditions where it was observed in the testing. This raised concerns on the conservatism of the EM for transient applications, and questions on the treatment of the determination of modeling uncertainty for ΔCPR was issued in RAI-37.

AREVA's responses in RAI-37a and RAI-37d provided greatly expanded descriptions of how transient evaluations are conducted with the EM to determine ΔCPR when applied to BWR thermal-hydraulic analysis with licensed CPR correlations for specific fuel assemblies. Specifically, for those cases where boiling transition was **[** **]**. In light of this observation, AREVA concluded that any non-conservatively predicted **[** **]**. AREVA further argued that, because these non-conservative cases have transient MCPRs **[** **]** Nevertheless, it appears reasonable to expect a good deal of overlap between the relevant uncertainty bands, and hence, that AREVA's approximation would not introduce a significant source of error. The NRC staff therefore concludes that the methodology described in the TR for determining boiling transition in AOOs is acceptable. The statistical treatment of the uncertainty in the ΔCPR is discussed in Section 3.6.

However, as mentioned above, the behavior within a rod array relative to the determination of CHF **[** **]** is highly dependent on fuel assembly design and the CPR correlation associated with that particular design. This is because each fuel assembly design is typically assigned a CPR correlation that has its parameters uniquely tailored to the empirical test results for that assembly type. Additionally, the transient performance of a given CPR correlation that has been incorporated into multiple thermal-hydraulic codes may differ from code-to-code depending on the specific thermal-hydraulic equations used in each of the codes. Therefore, based on the assessments presented in the TR, the NRC staff finds the AURORA-B EM can acceptably predict the time to boiling transition for the two CPR correlations used to generate the assessment tests: the SPCB CPR correlation used with the ATRIUM-10 fuel design and the ACE CPR correlation used with the ATRIUM-10XM fuel design. Other CPR correlations (existing and new) would need assessment via analyses performed using the approved version of the AURORA-B EM to confirm the predicted time to boiling transition is under-predicted and therefore conservative for the particular fuel design to which they would be applied.

For mixed core applications, CPR correlations must be developed using the AURORA-B methodology with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 50). If experimental data for transient boiling transition is available for other vendors' fuel, an assessment should be performed for this fuel in a manner similar to that discussed above for AREVA's own fuel. However, for proprietary reasons, such data is generally not expected to be available to AREVA. In this regard, the NRC staff notes that, as stated in EMF-2245(P)(A), the MCPR safety limit is primarily controlled by first-cycle fuel and that end-of-cycle conditions are normally limiting with respect to the determination of ΔCPR. Furthermore, the NRC staff recognizes that steady-state CPR correlations are generally expected to conservatively underpredict critical power under transient conditions (e.g., References 50 and 51). As a result, the unavailability of transient data for other vendors' fuels would not reduce the NRC staff's confidence in the capability of the AURORA-B EM to predict boiling transition.

3.3.2 BWR Core Neutronics Models

The BWR core neutronics models incorporated into the AURORA-B EM consist of the MICRBURN-B2 steady state neutronics/core simulator code (which includes the CASMO-4 lattice physics code) and the MB2-K neutron kinetics code. The version of the MB2-K neutron kinetics code submitted in the TR for application to transients and accidents has been derived from MICROBURN-B2, and it receives a significant portion of its input from MICROBURN-B2 via data transfer files. Because of this, the adequacy of the MICROBURN-B2 code for the present application is also included within the NRC staff's review. **[**

]

3.3.2.1 Previously Reviewed and Accepted Codes and Models

The MICROBURN-B2 steady-state neutronics code for BWR application has previously been reviewed and approved by the NRC (Reference 10). However, this prior review and approval did not include within its scope the code's applicability to EPU and EFW conditions. Therefore, the NRC staff found it necessary to assess the code's applicability in these areas. During the acceptance review, the NRC staff identified that supplemental information would be needed in order to adequately perform this assessment (Reference 3). AREVA supplied the necessary information in the August 2011 supplemental submittal (Reference 4). AREVA also supplied additional data applicable to EFW conditions in the updated revision to this submittal (Reference 52). Portions of this supplemental information are discussed below and specifically in Section 3.3.2.4.

3.3.2.2 Physical Modeling

The steady-state MICROBURN-B2 neutronics code for BWR applications produces nodal flux distributions, reconstructs pin powers, and simulates fuel depletion. MB2-K is a derivative of MICROBURN-B2 that includes kinetics parameters and delayed neutrons. The NRC staff examined the MB2-K modeling of the kinetics parameters and delayed neutrons and found that it is consistent with industry standard practice. The NRC staff therefore finds the modeling of the kinetics parameters and delayed neutrons acceptable.

Since MB2-K is derived from MICROBURN-B2, the two codes should produce similar results for steady-state calculations. To assess the consistency between the two codes, RAI-21 specifically requested a comparison of assembly powers and pin power reconstruction between MICROBURN-B2 and MB2-K. The AREVA response demonstrated that **[**

] and how one is derived from the other, the NRC staff observed that, **[**

] outside of the accepted MICROBURN-B2 limitations, then these limitations associated with MICROBURN-B2 should also be sufficient for MB2-K.

Although MB2-K is derived from the steady-state MICROBURN-B2 code, its primary employ in AURORA-B is for transient analyses. As a result, certain aspects of the code were specifically reviewed with transients in mind. One of the review areas concerned the coefficients used for the reflector response. However, the discussions presented in the TR regarding these coefficients did not directly address their applicability to transients. Therefore, the NRC staff inquired in RAI-81 whether the reflector response coefficients in MB2-K were not sensitive to changes in core conditions and therefore appropriate for rapid transients. AREVA responded that, [

] The NRC staff finds this response acceptable.

During the course of the review, the NRC staff observed that the discussion regarding the MICROBURN-B2 methodology for assembly power and pin power reconstruction did not address whether the methodology could be acceptably applied across large power differences between adjacent assemblies. In the responses to RAI-22 and RAI-83, AREVA indicated that [

] This assures that the reviewed and approved ranges of applicability for the two codes are met. Furthermore, AREVA stated that while there is no restriction [

] The NRC staff finds this response acceptable; it ensures proper application of MICROBURN-B2 within its approved range of applicability and appropriate penalties are applied as necessary.

As a result of the above responses and review efforts, the NRC staff finds that the neutronic modeling approaches from MICROBURN-B2 and MB2-K that have been incorporated within AURORA-B to be acceptable for the present application.

3.3.2.3 Validity of Calculated Results and Limitations

With the exception of the kinetics model in MB2-K, the rest of the physics is taken from MICROBURN-B2, a previously approved analysis method. Therefore, the assessment of the validity of MB2-K is limited to the kinetics model and analysis. Determining the validity of the kinetics' implementation in MB2-K is a challenge, since there is significant reactivity and power feedback from changes in coolant density and fuel temperature. Ideally, a comparison to a real reactor transient that has essentially no change in coolant temperature over the course of the transient would provide a good demonstration that the kinetics were properly implemented. This is because such a transient would effectively decouple the kinetics response from fuel temperature and thermal-hydraulic feedback effects. Lacking this ideal comparison, an alternate approach would be to determine that the calculated results are conservative relative to a reactor transient experiment. These results would provide evidence that the kinetics implementation is acceptable. Based on the review of the TR submission and several RAI questions, the validity of the MB2-K implementation was evaluated.

In an effort to approach the ideal comparison mentioned above, RAI-82 specifically requested test results that had little thermal feedback. AREVA **[**

] However, there is a set of numerical benchmarks that is often used by the industry to help demonstrate neutron kinetics equations are properly implemented. These numerical benchmarks are often called "industry-standard problems" (ISPs), and they are specifically designed to test spatial and temporal equations' numerical solutions. The numerical benchmark ISPs are the two-dimensional (2D) TWIGL seed-blanket reactor problem (Reference 30), the LMW 3D PWR delayed critical transient problem (Reference 31), and the LRA 2D and 3D BWR control blade drop (Reference 32) transients. The MB2-K results for these ISPs are presented in the AURORA-B TR.

In general, numerical benchmarks provide a useful test of equations and numerical solutions because numerical benchmarks are mathematical problems whose solutions are well known. Because the solutions are accepted as well-known, there often results within the literature a lot of code-to-code comparisons instead of code-to-benchmark solution comparisons (that is, the solutions as generated by the original benchmark codes). The aforementioned ISPs presented within AURORA-B are no exception. Indeed, in Section 6.4 of the AURORA-B TR, comparisons between the results for MB2-K and up to 9 other codes are presented. In all benchmarks, MB2-K yields a high degree of alignment with a number of other vetted codes, and this would suggest that the neutron kinetics equations in MB2-K are properly implemented. However, without code-to-benchmark solution comparisons, the accuracy of MB2-K's results is somewhat obfuscated. In an effort to examine code-to-benchmark solution comparisons, the NRC staff made several observations regarding the numerical benchmarks and the MB2-K results.

First, it was observed in the source documentation for the 2D TWIGL benchmark (Reference 30) that an analytical solution (that is, a closed-form solution result that demonstrates how a model will behave under any circumstances) or a verified numerical solution to the sample problem is not presented. Instead, what is presented is the "accepted solution" of the sample problem in the form of output generated from a code called "TWIGL." The "accepted solution" of the problem, according to the reference, is found by decreasing the time step in the TWIGL code until the flux response does not change out to four significant figures from one time step to the next. The reference document indicates that these tests that have since become known as the "2D TWIGL benchmarks" were intended to investigate code convergence/stability and precision (i.e., how the TWIGL code behaves). The NRC staff noted that MB2-K is not directly compared to the TWIGL code results, but to a variety of other codes that, in their own reference documents, are in turn compared to the original TWIGL code results. However, the high degree of agreement with these other vetted codes with respect to the TWIGL benchmark provides support that MB2-K's numerical methods possess an acceptable amount of stability and precision.

Secondly, with regard to the LMW 3D PWR Delayed Critical Transient reference document (Reference 31), a similar situation as the TWIGL tests exists. The LMW code is presented to determine "the ability of efficient time integration in the operational transient range," and the reference solution is again taken to be the results of the LMW code when utilizing the smallest time step. The value of this test lies in its ability to exercise the time integration behavior of codes. Again, MB2-K demonstrates a high degree of agreement with the benchmark solution results and other compared codes, which supports that MB2-K's time-integration techniques are sound.

Lastly, the NRC staff examined the LRA 2D and 3D BWR control blade drop transient reference document, Benchmark #14 of ANL-7416 (Reference 32). This is the Argonne Code Center Benchmark problem book, and the NRC staff observed that the benchmarks provided in this book possess peer-reviewed analytical solutions or very accurate, peer-reviewed numerical approximations (Reference 33). Benchmark #14 has three different versions: one-quarter-core 2D, one-quarter-core 3D, and full-core 3D. The last version is defined with asymmetric movement of a single control rod and is the most difficult of the tests because it involves severe radial flux asymmetry. The NRC staff compared the results of MB2-K for these tests to the solution provided in the ANL-7416 and found a high degree of agreement. Additionally, a high degree of agreement exists between the MB2-K results and the other code comparison results presented in the AURORA-B TR for this benchmark. This further bolsters the NRC staff's confidence that MB2-K's results for all the numerical benchmarks are both precise and accurate.

Moving beyond the numerical benchmarks, RAI-26 requested an analysis of how the delayed neutron fraction assumed in the AURORA-B calculation of the Peach Bottom Turbine Trip Tests compared to actual LPRM transient measurements. AREVA provided the calculated sensitivity to peak and integral power resulting from an assumed **[ ]** change in delayed neutron fraction for each of the three tests. The calculated result behaved as expected and showed that the delayed neutron fraction assumed in the TR results in **[ ]**. Therefore, the NRC staff found this response acceptable.

With the numerical benchmark results taken as a whole, and with consideration of the above RAI responses, the NRC staff finds that MB2-K adequately implements the neutron kinetics equations, and that MB2-K is acceptably implemented within the AURORA-B EM.

3.3.2.4 Qualification of CASMO-4/MICROBURN-B2 to EPU and EFW Conditions

As mentioned in Section 3.3.2.1, MICROBURN-B2's prior review and approval under the TR process did not include within its scope the code's applicability to EPU and EFW conditions. In recent years, the NRC staff has performed assessments of MICROBURN-B2's qualification to these conditions, but on a plant-specific basis through LARs (e.g., References 34-36). In each plant-specific LAR, AREVA has expanded the database supporting MICROBURN-B2's performance. However, assessment of MICROBURN-B2's qualification under EPU and EFW conditions on a generic basis has not occurred. During the acceptance review, the NRC staff identified that supplemental information would be needed in order to adequately perform a generic assessment (Reference 3), and AREVA supplied the necessary information in August 2011 (Reference 4). AREVA also supplied additional data applicable to EFW conditions in an updated revision to this submittal (Reference 52).

It is not the NRC staff's intent within this SE to reproduce work already performed in the aforementioned plant-specific LAR reviews. Instead, the NRC staff's conclusions from these reviews that are pertinent to the present discussion will be referenced.

Assessment of MICROBURN-B2's qualification for EPU and EFW analyses begins with an understanding of the neutronic conditions present in these domains. EPU and EFW domains are characterized by higher core powers and lower minimum core flows, respectively. For EPU cores to sustain the higher core power through the same cycle duration as a pre-EPU core, a combination of different assembly enrichments and a larger fraction of fresh assemblies loaded each cycle is used. High-energy cores such as this require high burnable poison loadings to compensate for the additional excess reactivity necessary to sustain core criticality for the cycle duration at the higher thermal power. Additionally, EPU cores have a higher average bundle power relative to the minimum core flow due to the slope of the load lines on typical power-flow maps. The reduction in the minimum core flow due to operating in EFW domains has a similar, but larger effect with respect to this ratio. A higher power-to-flow ratio in fuel bundles will tend to result in a higher core average void fraction.

Higher loadings of neutron poisons and higher core average void fractions each harden the neutron spectrum during cycle exposure. Burnable poisons such as gadolinium preferentially absorb thermal neutrons, and a heaver poison loading will do so with greater affinity. This shifts the flux spectrum towards epithermal and fast neutron energies. Higher core average void fractions yield a reduction in the number of neutrons moderated to thermal energies. This also shifts the flux spectrum towards epithermal and fast neutron energies.

A harder neutron flux spectrum causes a shift in the fuel's isotopic concentration, resulting in a series of cascading cause-and-effect relationships that ultimately impact the determination of thermal safety limits. Specifically, a change occurs in the production and removal rates of several plutonium isotopes (e.g., Pu-239, Pu-240, and Pu-241). The ratio of fissile plutonium to fertile plutonium has a strong impact on pin peaking late in life when the reactor core power shifts towards the top of the core. This ratio obviously also affects the distribution of reactivity worth in the reactor core, which in turn affects core-wide power and void distributions. Particularly, when the core void distribution changes radially (i.e., is distributed differently among the fuel bundles in the core), corresponding changes occur in bundle flow rates. Such changes in void distribution and flow rate affect bundle power and linear heat generation rate (LHGR), which ultimately impact the determination of MCPR. Therefore, it is important that those phenomena and processes which affect the neutron flux spectrum and determination of bundle powers are adequately modeled in the core neutronic methods. The NRC staff identified that the most influential of these phenomena and processes are: the prediction of cross-sections, the effects of gadolinia burnup and self-shielding, the tracking of plutonium accrual, and the prediction of void fraction. The NRC staff's assessment of each of these is discussed in the following subsections.

3.3.2.4.1 Cross-Section Representation

MICROBURN-B2 utilizes a large library of tabulated nuclear data generated by a lattice physics code when determining nodal macroscopic cross-sections. In the case of the AURORA-B EF, the specific lattice physics code is CASMO-4. The multi-dimensional table of microscopic and macroscopic cross-sections that CASMO-4 generates encompasses instantaneous branch calculations at alternate conditions of void, burnup, and control state. However, the multi-dimensional table has a finite resolution to it in that not every conceivable burnup/void/control state can be tabulated. It is therefore up to the core/neutronics code to interpolate between the points provided in the multi-dimensional table to obtain the cross-sectional values attributable to the core state under analysis. MICROBURN-B2 is no exception in this regard.

At beginning of life (BOL) conditions, cross-sections are primarily a function of void fraction (moderator density) due to the change in the neutron flux spectrum caused by moderator density variations. At BOL, a quadratic fit of the CASMO-4 data points is used to represent the continuous cross-section over instantaneous variation of void fraction or moderator density. To demonstrate the adequacy of the quadratic fit for BOL conditions at high void fractions, the supplemental information provided comparisons of detailed CASMO-4 and MICROBURN-B2 cross-section predictions **[ ]**. The results show excellent agreement, and the NRC staff therefore finds this approach acceptable for BOL conditions.

However, the situation becomes more complicated as burnup progresses because the isotopic changes will also result in neutron flux spectral changes. This will in turn cause additional changes to the predicted cross-sections. Furthermore, self-shielding effects will vary with the isotopic concentration. In this case, a combination of piecewise linear interpolations over tabulated values at various exposures is performed in conjunction with quadratic interpolations over the remaining parameters **[ ]** branch calculations. The intersection of these three quadratic void fraction interpolations with that of the historical void fraction (the void fraction observed in the core up to the exposure point of interest) are then used to create a final quadratic fit. It is this final quadratic fit that is ultimately used to determine the cross-sections. In support of this approach at high void fractions, AREVA again supplied comparisons of detailed CASMO-4 and MICROBURN-B2 results. This time, the comparisons were for predictions of **[**

]. As before, the results show excellent agreement.

However, despite the excellent agreement, the NRC staff noted **[**

]. Figure A-15 of the August 2011 supplement (Reference 4) suggests that the fitting approach used in MICROBURN-B2 yields cross-sections that **[**
]. Since these void percentages are typical of extended power/flow windows, it was necessary to assess whether the MICROBURN-B2 cross-section predictions were conservative for the transients considered in the TR. Thus, RAI-20 and RAI-24 questioned how the cross sections were developed for MICROBURN-B2 for higher voids. AREVA's response indicated that the core response in a transient is based upon the entire core reactivity, since no region responds in isolation to its environment. With that said, comparisons were made between k-infinity values from MICROBURN-B2 and **[**

] The NRC staff finds this response acceptable.

The NRC staff acknowledges that reasonable to excellent agreement exists for the cross-section predictions between CASMO-4 and MICROBURN-B2 at high void fractions. Additionally, the NRC staff finds that AREVA's explanation regarding **[**

]. Therefore, the NRC staff finds that MICROBURN-B2's cross-section representation is applicable to the transient, infrequent event, and accident scenarios described in Section 3.1 at EPU and EFW conditions.

3.3.2.4.2 Burnable Poison: Gadolinia Pin Self-Shielding

At very high loadings, gadolinia pins become self-shielding. In effect, the gadolinium on the outer surface of a fuel pin absorbs neutrons to such an extent that thermal neutrons do not appreciably penetrate the fuel pin. This is referred to as the spatial self-shielding effect. Generally, gadolinium depletion is modeled to account for the resulting "onion-skin" effect, where gadolinium in the outer radial regions of the fuel pin depletes at a faster rate than in the inner regions of the pin.

These highly loaded gadolinia pins may reside in lattice locations, such as near unheated water channels, where the enhanced neutron thermalization due to the increase presence of the liquid phase influences the depletion rate of the gadolinium in the outer regions of the fuel pin. If the gadolinia loading is high, the required radial resolution to model the gadolinium depletion rate increases. Predicting higher or lower gadolinia depletion rates has the potential to result in miscalculation of the lattice pin peaking factor near peak reactivity conditions (where the nodal power is expected to be high). The combined effect of miscalculation of the pin peaking factors concurrent with anticipated high nodal power may influence the determination of the margin to thermal limits.

In the Susquehanna Steam Electric Station (Susquehanna) EPU LAR review (Reference 34), the NRC staff reviewed the technical basis for the acceptance of the CASMO-4/MICROBURN-B2 methodology to determine lattice peaking factors. The NRC staff's review investigated the CASMO-4 methodology (which is based on the collision probability technique) and its original qualification in TR EMF-2158(P)(A) (Reference 10) and found that there is reasonable assurance the neutronic methods will adequately predict pin peaking factors for the gadolinia pins under EPU conditions.

The NRC staff's conclusion is based on two important observations. First, the 2D lattice geometry in CASMO-4 is treated in sufficient detail to allow for robust modeling of the spatial self-shielding effects in the gadolinia pins. CASMO-4 solves the associated equations using a large number of energy groups [ ] and therefore captures the sharp change in burnable absorber cross-section with neutron energy.

Second, the original CASMO-4 qualification results, which utilize pin-by-pin gamma scan measurements from Quad Cities Nuclear Power Station, do not indicate any bias in pin peaking factor with changes in void fraction or proximity to water channels, demonstrating the collision probability technique is sufficiently robust for EPU conditions. Similar use was made of pin-by-pin gamma scan data in the NRC staff's review of the Monticello Nuclear Generating Plant (Monticello) EFW LAR (Reference 36). In this instance, pin-by-pin axial gamma scan measurements of AREVA ATRIUM-10 fuel assemblies in the reactor designated KWU-S were examined. The results were found to validate pin power predictions at EPU/EFW conditions.

The NRC staff finds that the assessment of the 2D lattice geometry in CASMO-4 remains applicable to EFW conditions. The NRC staff also finds that the level of detail in the lattice geometry and the number of energy groups used make the method sufficiently robust. Based on these observations and the above discussion, the NRC staff finds that the CASMO-4/MICROBURN-B2 methodology can acceptably model gadolinia pin self-shielding effects and gadolinium depletion in the EPU and EFW domains.

3.3.2.4.3 Plutonium Accrual

EPU/EFW cores will generate a greater amount of plutonium due to the harder neutron spectrum resulting from higher poisoning and/or void fractions. In this regard, the isotopes Pu-239, Pu-240, and Pu-241 are of primary interest. The relative production and destruction rates of these isotopes are driven, predominantly, by low-lying resonances. These are narrow peaks in the cross-section at very low epithermal energies, which are typical of many higher actinides. The Pu-240 resonance is particularly sensitive to the incident neutron energy spectrum. As noted above, the ratio of the fissile plutonium nuclides to the fertile nuclides has a strong impact on pin peaking, and it also affects the distribution of reactivity worth in the reactor core, and therefore the prediction of core power and void.

In the Susquehanna EPU LAR, the NRC staff reviewed the technical basis for the acceptance of the CASMO-4/MICROBURN-B2 methodology to adequately account for plutonium accrual at EPU conditions. The NRC staff's review investigated the original qualification for the methodology in TR EMF-2158(P)(A) and found that there is reasonable assurance that CASMO-4/MICROBURN-B2 can model the accrual of plutonium at EPU conditions.

The NRC staff's conclusion is based on two primary observations. First, the qualification in TR EMF-2158(P)(A) includes mixed oxide (MOX) pin gamma scans. These MOX data provide reasonable assurance that the energy resolution of the CASMO-4 **[ ]** transport calculation is sufficient to resolve the Pu-240 low-lying resonance capture cross-section. Since the low-lying resonance for Pu-240 is very strong, an error in collapsing the cross-section would result in a noticeable bias in the pin peaking factors for MOX pins under any of the void conditions presented in TR EMF-2158(P)(A), and a bias in pin peaking is not observed.

Second, MICROBURN-B2 models the depletion of the principle actinides using a microscopic tracking method that accounts for the production and destruction of these nuclides explicitly in the core model. The accounting methodology is based on the CASMO-4-generated collapsed microscopic cross-sections, and it explicitly tracks the accrual of plutonium for nodal-specific historical conditions. The robustness of this explicit technique is evidenced by gamma scan data from Quad Cities showing that axial power profiles (pin or bundle) show no bias with axial elevation, which would otherwise be expected given the tendency for plutonium to accrue preferentially in the upper portions of the reactor core where the void fraction is highest.

Because it considers nodal-specific historical conditions, the NRC staff finds that the explicit isotopic tracking method is sufficiently robust for application to EFW conditions. In the case of the low-lying resonance for Pu-240, an error in cross-section collapsing would be expected to result in a noticeable bias in the pin peaking factors for MOX pins. Since no bias is observed in any of the MOX gamma scan data for any of the void fraction cases applicable to EPU conditions, it is unlikely that any appreciable bias will be introduced by the void fractions present at EFW conditions. While EFW conditions contain a higher core average void fraction and an increased degree of spectral hardening compared to EPU conditions, the difference in magnitude is not sufficiently large as to suggest that an appreciable pin peaking bias due to an error in cross-section collapsing could exist under one set of conditions and not be manifest in the other. However, while no appreciable bias is expected, increased uncertainty in prediction of pin and bundle powers are possible. Therefore, the above discussion does not, in and of itself, serve to adequately justify that the uncertainties for pin and bundle powers previously established in TR EMF-2158(P)(A) remain applicable. The NRC staff's assessment of pin and bundle power uncertainty is discussed below in Section 3.3.2.4.5.

The NRC staff finds that the discussion above and the NRC staff's assessment of pin and bundle power uncertainty discussed in Section 3.3.2.4.5 support extension of the methods for the accrual and depletion of plutonium within CASMO-4/MICROBURN-B2 to EFW conditions.

3.3.2.4.4 Void-Quality Correlation

A key issue in the EPU and EFW domains is the higher void fractions during steady state operation as well as during transients, which may lead to an annular flow pattern in the upper portion of the reactor core. MICROBURN-B2 utilizes the **[ ]** correlation to relate the flow quality, which can be directly calculated, to the void fraction. This correlation is based on a two-fluid semi-empirical bubble rise model. Since the semi-empirical **[ ]** correlation formulation is not representative of the three-field (i.e., liquid film, vapor core, and entrained liquid droplets) flow phenomena for annular flow regimes, the NRC staff finds that this correlation's validation is limited to the range of its qualification database. Furthermore, the NRC staff finds that the database must be representative of the fuel bundle geometry to support the application of the correlation to high void fractions. Therefore, the NRC staff reviewed the applicability of this correlation's qualification database, with particular emphasis on the predictive capability for conditions with increased void fractions that occur under EPU and EFW conditions. The NRC staff also reviewed the qualification database to determine whether it is applicable to representative fuel bundle geometries.

In the Susquehanna EPU and Monticello EFW LARs, the NRC staff reviewed the technical basis for the application of the **[ ]** correlation for EPU conditions and EFW conditions, respectively. The NRC staff's review investigated the correlation's qualification database for the ATRIUM-10 and ATRIUM-10XM fuels and found that for these fuels, the **[ ]** correlation is applicable to EPU/EFW conditions.

The NRC staff's conclusions were primarily based on three observations. First, AREVA supplied void fraction measurement data from experiments performed at the KATHY test facility. **[**

] Additionally, hot channel exit void fraction in EPU conditions is not expected to exceed [ ].

Second, the void fraction data from the KATHY experiments was acquired using full-scale testing of ATRIUM-10 and ATRIUM-10XM assemblies. The coefficients of the [ ] correlation were not adjusted based on the data gathered in the tests; rather, the tests served to demonstrate the applicability of the correlation to the conditions expected during the operation of the ATRIUM-10 and ATRIUM-10 XM fuels.

Third, the use of the [ ] correlation at void fractions above those measured in the KATHY test facility was found to be warranted. Two points on the void-quality curve are taken for granted: the points that correspond to pure liquid and pure steam flow. This knowledge is built into all void-quality correlations; consequently, as the quality is increased, all correlations approach a void fraction of 1.0. Thus, as quality (and void fraction) increases, the inherent uncertainty in the correlation should decrease and become essentially zero at a quality of 1.0. Even though the amount of data above [ ] void fraction is limited, the uncertainties should not significantly increase.

In the first case above, the NRC staff expects that the biases are indicative of a correlation based on a semi-empirical bubble rise model. At higher void fractions, a slight bias is expected as the effect of interfacial shear along the liquid film drives vapor slip, and liquid water droplets in the annular core contribute to the total liquid mass flow rate. At lower void fractions, effects such as subcooled boiling are not fully captured in the formulation of the correlation. Therefore, small divergences are not unexpected.

Based on the above discussion, the NRC staff finds the [ ] correlation is applicable to EPU and EFW conditions for the ATRIUM-10 and ATRIUM-10XM fuels. Because of the geometry-specific nature of the correlation's validation at these conditions, AREVA must justify that the [ ] correlation is valid and applicable to EPU and EFW conditions for other fuel designs (e.g., ATRIUM-11, mixed-core co-resident fuel).

3.3.2.4.5 Pin Power and Bundle Power Uncertainties

The uncertainty in prediction of pin power and bundle power contributes to the uncertainty in safety limits. Axial pin-by-pin gamma scans and full bundle gamma scans provide data from which the uncertainties in pin power peaking and bundle power, respectively, may be quantified. The measured parameter in the gamma scan measurements is the gamma activity of the La-140 isotope. La-140 is a decay product of Ba-140, which is a direct fission product. Ba-140 is relatively short lived (12.8-day half-life), so its atom density is proportional to the fission rate immediately before measurement, and the activity distribution for the short-lived La-140 (1.68-day half-life) is representative of the Ba-140 atom density distribution. Thus, comparison of core physics models to gamma scan results is done by converting the pin power distribution to a Ba-140 density distribution.

Traversing in-core probes (TIPs) directly measure the local neutron flux from the four fuel assemblies surrounding a given location. Bundle power distribution is calculated based on TIP measurements. Thus, the uncertainty in the bundle power distribution will be closely related to the calculated TIP uncertainty. However, the bundle powers in the assemblies surrounding a TIP measurement location are not independent. If a bundle is higher in power, neutronic feedback increases the power in the nearby assemblies, thus producing a positive correlation between nearby bundles. Bundle gamma scan data provides the means to determine this correlation. These correlation coefficients and the TIP uncertainties are used to calculate the bundle power distribution uncertainty. A smaller correlation coefficient implies that there is less correlation between nearby bundle powers, and thus, a larger bundle power distribution uncertainty.

The original CASMO-4/MICROBURN-B2 qualification TR EMF-2158(P)(A) included pin-by-pin gamma scans for a once-burnt ATRIUM-10 fuel bundle at spectral conditions similar to EPU operation from the Gundremingen Unit B (GUN-B) plant (3840-megawatt (MWt) uprated) in Germany. During the Susquehanna EPU LAR review, the NRC staff observed that a comparison of the spectral index (the ratio of fast to thermal flux) showed the GUN-B spectral conditions for the ATRIUM-10 bundle remained softer than the spectral index anticipated for EPU cores. This is caused, in part by the lower gadolinia loading and enrichment. The NRC staff's conclusion was the gamma scans supported application of MICROBURN-B2 to EPU conditions, but the scans did not adequately justify the use of the previously established pin and bundle power uncertainties in TR EMF-2158(P)(A).

Based on the available gamma scan data, and in the absence of additional applicable gamma scan data, the NRC staff found during the Susquehanna EPU LAR review that pin and bundle power uncertainties were not expected to increase by more than **[ ]** of the established values for the harder spectral conditions present in the EPU domain. The impact of these uncertainties on the SLMCPR was evaluated in an analysis performed by the licensee. For the pin power, a sensitivity analysis was performed where the pin power uncertainty was increased by 50 percent. **[ ]**. For the bundle power, the correlation coefficient was reduced by **[ ]** (thereby increasing bundle power uncertainty). When combined with local power range monitor (LPRM) failures, TIP out of service, and LPRM calibration interval uncertainties, the resulting change in SLMCPR was **[ ]**, which the NRC considers to exceed the threshold of concern. Ultimately, the application of the **[ ]** pin and bundle power uncertainties was found sufficient to account for any other uncertainties attributable to the extension of MICROBURN-B2 to higher spectral indices.

This approach to accounting for methodology uncertainties at EPU conditions was repeated for other EPU LAR reviews such as Browns Ferry Nuclear Plant (Browns Ferry) (Reference 35). For EFW LARs such as Monticello, a SLMCPR penalty of 0.03 was imposed to further account for uncertainties in extending methodologies to EFW conditions where the effects of spectral hardening are exacerbated. This SLMCPR penalty has its basis in non-AREVA methods wherein an analysis to account for uncertainties attributable to extending methodologies to EPU and EFW conditions was employed (Reference 37) in a manner very similar to that of the Susquehanna EPU LAR review. This analysis resulted in a 0.02 change in SLMCPR at EPU conditions and a 0.03 change at EFW conditions.

At the core of both the EPU and EFW SLMCPR penalties is the lack, at the time, of applicable gamma scan data to quantify the pin and bundle power uncertainties for the EPU and EFW domains. As part of the Monticello EFW LAR review, AREVA supplied the results of a more recent gamma scan campaign comprised of 48 assemblies that included AREVA ATRIUM-10B and Westinghouse Electric Company SVEA-96 Optima 2 designs (Reference 38) from the GUN-B plant in Germany. While these data were supplied to support operation of Monticello at EFW conditions, GUN-B is a 3840-MWt uprated BWR with an asymmetric lattice design, making it representative of a D-lattice plant (i.e., characteristic of BWR/2-3 and some BWR/4 reactors) operating at EPU conditions. The results present the relative standard deviations for MICROBURN-B2 measured bundle and pin power distributions. These results meet the measured power distribution uncertainty criteria specified for D-lattice plants in Table 5.3 of the MICROBURN-B2 qualification report, TR EMF-2158(P)(A).

In addition to the gamma scan data, AREVA also supplied TIP statistical uncertainties for the EPU plants in their fleet as a function of core power-to-flow ratio. This data includes a comparison of TIP uncertainties at pre- and post-EPU operation. Both the pre- and post-EPU 3D TIP uncertainties presented in these results meet the TIP relative standard deviation acceptance requirements presented in Table 5.2 of the MICROBURN-B2 qualification report, TR EMF-2158(P)(A).

AREVA's intent with supplying 3D TIP data for pre- and post-EPU conditions was to demonstrate that the TIP uncertainties do not increase with 1) changes in operating domain (e.g., pre- to post-EPU) and 2) increasing power-to-flow ratios. As noted above, the uncertainty in the calculated bundle power distribution is closely related to the calculated TIP uncertainty. Therefore, while both TIP uncertainties and correlation coefficients are used to calculate the bundle power distribution uncertainty, it is expected that any change in TIP uncertainty would be indicative of a trend in bundle power distribution uncertainty. In other words, evaluating TIP uncertainty can qualitatively assess a change in bundle power distribution uncertainty. However, to quantify any change in the bundle power distribution uncertainty would require applicable gamma scan data for determination of the correlation coefficients.

Evaluation of the 3D TIP data supplied by AREVA for the Monticello EFW LAR shows no significant error trend with increased power-to-flow ratio or from pre- to post-EPU. This suggests there is no change in the bundle power distribution uncertainty. Additionally, several of the 3D TIP data points within the results were collected at power-to-flow ratios higher than 42 megawatts thermal per million pounds per hour (MWth/Mlb/hr), which is typical of operation in the EFW domain. However, the collection of data at these higher power-to-flow ratios was too limited to come to a meaningful conclusion regarding TIP uncertainty.

As a result, AREVA built upon this data by providing additional information in the August 2011 supplemental response (Reference 4). Since TIP data above 42 MWth/Mlb/hr is sparse, AREVA chose to utilize [ ]. The primary purpose of the TIP system is to provide data for the [ ]. AREVA concluded that the comparison of [ ]

While the NRC staff acknowledges that, [ ]

[ ]

[ ]

[ ]

Based on the discussions above, the NRC staff finds that for the power-to-flow ratios examined, it is unlikely that a bundle power uncertainty exceeding the acceptance criteria of TR EMF-2158(P)(A) will be encountered at EFW conditions. Hence, the uncertainties quantified for pin and bundle power distributions within TR EMF-2158(P)(A) remain applicable. In light of the information supplied by AREVA during the present review, **[ ]**, the NRC staff further concludes that imposition of a SLMCPR penalty for EPU conditions is not necessary. As explained above, however, this conclusion is contingent upon the use of MICROBURN-B2-based methods **[ ]**. However, given that the range of power-to-flow ratios examined may not be bounding for every corner of a plant's EFW domain, potential uncertainty impacts on SLMCPR should nevertheless be reviewed on a plant-specific basis.

Additionally, based on the discussions in the preceding sections, the NRC staff finds that the CASMO-4/MICROBURN-B2 code system is qualified for analyses at EPU and EFW conditions. This qualification applies only to fuel designs for which AREVA has justified that the **[ ]** void-quality correlation is valid at EPU and EFW conditions, as discussed in Section 3.3.2.4.4.

3.3.2.5 Specific Uncertainties in Model (Experimental, Measurement, and Model)

The model uncertainties that are used in the methodology to calculate the figures of merit for safety analyses are discussed in Section 3.6.

Given that MB2-K will be subject to the same limitations as MICROBURN-B2 (Section 5.0), and given that AURORA-B **[ ]**, the NRC staff finds that AURORA-B adequately predicts BWR core neutronics performance for the transient and accident scenarios described in Section 3.1.

3.3.3 Fuel Thermal Mechanical Models

The AURORA-B EM makes use of the RODEX4 steady-state fuel performance code to determine fuel thermal-mechanical properties. The code is actually incorporated into the EM in two different ways, and because each actualization of RODEX4 serves a different purpose, it is important to delineate the difference between the two. In the first case, the full, stand-alone RODEX4 code is used to determine initial conditions for the irradiated fuel **[ ]** at the desired time in cycle prior to performing the transient or accident analysis. In the second case, a subset of the routines from the full RODEX4 code, known as the "RODEX4 kernel," has been incorporated into S-RELAP5 to support evaluation of the fuel thermal-mechanical properties during transients. Specifically, the RODEX4 kernel supplies fuel rod properties used by S-RELAP5 when solving the transient thermal conduction equations in lieu of standard S-RELAP5 material property tables. The NRC staff examined the information used to initialize fuel conditions as well as the subset of routines that comprises the RODEX4 kernel. Review of the RODEX4 kernel's coupling with S-RELAP5 and the nature of their information exchange can be found in Section 3.3.5 of this SE.

3.3.3.1 Previously Reviewed and Accepted Codes and Models

The thermal-mechanical modeling in AURORA-B comes from RODEX4 models and subroutines. This code was approved in TR BAW-10247(P)(A) Revision 0, "Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors" (Reference 11).

3.3.3.2 Physical Modeling

The steady-state RODEX4 fuel performance code is used to determine the initial conditions for the irradiated fuel (e.g., material properties, dimensions, gas composition, fuel-clad gap size, rod pressures, etc.) prior to the transient or accident. The exact data transferred from RODEX4 to AURORA-B for initialization is not well defined in the TR. Therefore, the NRC staff issued RAI-95, which requested a detailed description of those data passed from RODEX4 for initialization of AURORA-B. AREVA responded that the following parameters are passed **[ ]**:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

Following initialization, as described in the response to ARAI-12, a subset of the routines from the full RODEX4 code (contained in the smaller RODEX4 kernel within S-RELAP5) is used to predict **[ ]**

As noted in the TR and mentioned above, a subset of models and subroutines from RODEX4 is included in S-RELAP5 to evaluate the transient thermal-mechanical performance. The TR states that **[**

], but it does not discuss the treatment of other specific models. RAI-15 requested that AREVA define those RODEX4 models specifically implemented in the RODEX4 kernel within S-RELAP5 and define the status of these models relative to the approved version of the TR for the full RODEX4 code. AREVA responded that the following RODEX4 models from the approved TR (Reference 12) are used in the AURORA-B EM:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

Based on this response, the NRC staff raised a concern that AURORA-B **[ ]**. RAI-16 requested that data and analyses be provided **[ ]** these models in applications of the AURORA-B EM for transient and accident analyses. AREVA responded that there are three reasons why **[ ]**

Given these considerations, the NRC staff finds that the occurrence of **[ ]**. The NRC staff therefore finds the response acceptable. However, the NRC staff observes that the reasons provided in AREVA's response are grounded within the range of AOOs for which AURORA-B is currently applicable. Any future expansion to AURORA-B's range of applicability (e.g., to RIAs or LOCAs) may require re-examining the acceptability of excluding the burst FGR and gaseous swelling models.

In ARAI-5, the NRC staff requested additional information relative to possible application of the AURORA-B EM or RODEX4 for evaluating cladding strain and centerline temperature. AREVA responded that the AURORA-B EM explicitly evaluates the cladding strain and fuel centerline temperature via the RODEX4 kernel for each node of each fuel group. However, these results are not directly used in demonstrating that thermal-mechanical criteria are satisfied. Rather, the full RODEX4 code fast transient methodology (Reference 12) is used to evaluate cladding strain and centerline temperature for safety analyses using AOO power levels calculated with AURORA-B. This is done because the fuel assemblies or assembly groups in the AURORA-B EM core model are not likely to be at the limiting exposure or peak LHGR conditions necessary for evaluating limiting transient cladding strain or peak fuel temperature.

Based on the above discussion, the NRC staff finds the modeling approach for the fuel thermal-mechanical models and the justification for the specific selection of models that are included in the RODEX4 kernel to be acceptable.

3.3.3.3 Range of Validity of Field Equations

The RODEX4 fuel thermal-mechanical code has received prior NRC review and approval (Reference 12). Based on this prior approval, and with consideration of the results presented in the AURORA-B TR, the RODEX4 code and its models within the AURORA-B EM are found to be acceptable for the following ranges of application. The code is acceptable to model BWR fuel rods with solid pellet (non-annular) uranium dioxide fuel and with cold-worked stress-relieved (CWSR) Zircaloy-2 cladding, up to a peak rod-average-burnup of 62 GWd/MTU. The code is also acceptable for modeling mixed urania and gadolinia fuel rods with up to 10 weight percent gadolinia, up to a peak rod-average-burnup of 62 GWd/MTU. These same approved ranges of applicability would apply to the subset of routines from RODEX4 that are incorporated within the RODEX4 kernel in S-RELAP5.

In addition to the ranges of approval discussed above, the following limitations were placed on the application of the RODEX4 code for licensing analyses in prior NRC reviews and therefore also apply to the subset of the routines from RODEX4 that is incorporated into the RODEX4 kernel in S-RELAP5:

- The RODEX4 code is only applicable to fuel **[** **]**.

- The FGR model is only applicable to fuel **[** **]**

In addition to these limitations, the NRC staff made two observations regarding fuel performance models with respect to the RODEX4 kernel and its function within the AURORA-B EM. First, the RODEX4 **[** **]**. Second, as discussed in Section 3.3.3.2 above, the RODEX4 kernel **[** **]**, future expansions to the

AURORA-B range of applicability, such as the inclusion of LOCA or RIA analyses, may need to estimate parameters determined by these models in order to meet NRC regulatory requirements. It is therefore noted here that re-examination of the acceptability **[** **]** may be required in future submittals because the RODEX4 kernel is responsible for calculating the transient fuel thermal-mechanical properties that the aforementioned analyses would utilize.

3.3.3.4 Specific Fuel Rod Uncertainties

TR ANP-10300P presented a sensitivity study-based uncertainty analysis methodology. In this methodology, the uncertainty in **[** **]**, and the fuel rod related sensitivity studies included sampling of both of these parameters. However, **[** **]** gap gas composition, gas pressure, and fuel-clad surface roughness (if fuel/cladding contact is present). Gap conductance is an important parameter in evaluation of hot channel ΔCPR to verify that the MCPR limit is met, and it is therefore important to adequately determine the associated uncertainty. Therefore in RAI-12, the NRC staff requested that AREVA provide an assessment of the uncertainty in gap conductance by performing sensitivity analyses **[** **]**. In RAI-49b, the NRC staff requested justification for AREVA's perturbation **[** **]**.

[**]**. In the response to RAI-49b, AREVA indicated that, as part of the updated uncertainty analysis methodology, it is better to sample on the primary fuel rod parameters that determine **[** **]**. The primary fuel rod parameters AREVA chose to sample include:

- **[** **]**
- **[** **]**
- **[** **]**
- **[** **]**
- **[** **]**

As provided in the RAI response, AREVA's justification for the selection of these parameters is that, **[** **]** Therefore, the NRC staff finds AREVA's response acceptable. The assessment of uncertainties and sampling for each of these parameters is discussed in Section 3.6.4 of this SE.

Based on the above discussions, and because the RODEX4 kernel is subject to the same limitations and conditions as the full RODEX4 code, the NRC staff concludes that the AURORA-B code adequately predicts thermal-mechanical fuel performance.

3.3.4 BWR Primary System Models

The CCDs comprising the AURORA-B EM model a variety of important phenomena, such as thermal-hydraulics, neutronics, and fuel thermal-mechanical performance. Equally important to the EM are the models describing the various plant hydraulic components, heat structures, and control systems comprising the BWR primary system. Modeling of these components and structures is a mathematical mapping from the physical system (i.e., the “real-world”) to the computational structure of the EM. In TR ANP-10300P, AREVA terms this modeling process “nodalization.”

Nodalization generally refers to the mathematical representation of spatial dimensions of the physical system (e.g., flow areas, lengths, and volumes); however within TR ANP-10300P, this definition has been expanded to include all computer code input necessary to represent any engineered features influencing plant performance (e.g., trips and control systems, pumps, and component performance). Model options and input for phenomenological code models are also included.

Different approaches to nodalization may yield different results. In order to minimize these differences and ensure consistency, the EM relies on a consistent approach to defining the nodalization of the BWR plant. This is accomplished through a framework of technical guidance for standardized input models that has been prepared for the AURORA-B EM. Specifically, the submitted TR summarized key aspects of the technical guidance for nodalization of hydrodynamic components and heat structures, as well as the modeling practices for key components and processes. Requirements for establishing control variables, trip definitions, and their use are provided in Reference 26. These parameters are discussed in greater detail in Section 3.3.4.2.

In addition to examining the aforementioned parameters, the NRC staff also examined the modifications made to S-RELAP5 to support the BWR primary system. As mentioned in Section 1.1, S-RELAP5 was originally developed for PWR transient and LOCA analyses. Hence, it was necessary for AREVA to extend the S-RELAP5 source code beyond its existing approval basis (References 5 - 8) in order to support BWR applications. Improvements to existing models were made and several new models were added. Specifically, the major changes made to S-RELAP5 for BWR applications are:

- A new jet pump model was added (Section 3.3.4.3.1)
- Interfacial drag and heat and mass transfer models were improved (Section 3.3.4.3.2)
- Improvements were made to the mechanistic BWR separator model (Section 3.3.4.3.3)

- New pressure drop models for BWR fuel assemblies (consistent with approved models in MICROBURN-B2) were added (Section 3.3.4.3.4)

- Previously approved BWR CPR correlations were added (Section 3.3.1.5)

These new models are discussed in greater detail in each of the subsections indicated above.

3.3.4.1 Previously Reviewed and Accepted Codes and Models

The thermal-hydraulic and thermal conduction field equations used within S-RELAP5 are the same as used in the RLBLOCA methodology described in TR EMF-2103(P)(A), Revision 0 (Reference 7) and other NRC-approved methodologies based on S-RELAP5. Technical detail related to the field equations is provided in the S-RELAP5 theoretical description (Reference 5). Therefore, the NRC staff did not review the existing formulation of the field equations in S-RELAP5. However, the NRC staff did review the applicability of the code to the defined benchmark cases (see Section 3.3.1).

The numerical solution techniques contained in S-RELAP5 for the thermal-hydraulic and thermal conduction equations are described in TR EMF-2100(P) Revision 14 (Reference 5), and are the same as used in other S-RELAP5 based methodologies. Therefore, the NRC staff did not review the existing numerical solution techniques. However, the NRC staff did review the numerical stability of the code to the defined benchmark cases (see Section 3.3.1). In ARAI-13, the NRC staff requested additional information on the default numerical solution technique and the user guidance that dictates when alternative numerical solution techniques may be applied. AREVA responded that AURORA-B uses the standard S-RELAP5 numerical solution scheme, which is described in detail in Section 2.6 of the S-RELAP5 models and correlations code manual (Reference 5). To summarize, the numerical solution technique is generally semi-implicit with consideration for both accuracy and computing speed. More specifically, mass inventory and total energy are conserved and convected from the same cell and evaluated at the same time. Implicit evaluation is used for velocity in mass and energy transport terms, the pressure gradient in the momentum equations, and the interphase mass and momentum exchange terms. This is done because these quantities are responsible for the sonic wave propagation time-step limit and for phenomena known to have small time constants. The system of differential equations is solved as a system of finite-difference equations that is semi-implicit in time. These are manipulated to produce an $N \times N$ matrix of simultaneous pressure equations with N control volumes. AREVA indicates that the well-posed structure of this numerical scheme (as well as its accuracy) has been demonstrated by extensive numerical testing during the development and subsequent applications of the RELAP5 family of codes. The techniques to effectively use the solution scheme (time-step control and node sizes) are provided in the S-RELAP5 user's manual (Reference 28). Given that this approach is the same as used in S-RELAP5 for the approved RLBLOCA methodology (Reference 7), the NRC staff finds this response acceptable.

In ARAI-14, the NRC staff requested additional information for time step algorithms related to the neutron kinetics and thermal-hydraulic solutions, the default ratio for neutron kinetics to thermal-hydraulic solution time step, and the user guidance used to establish an appropriate value for this ratio. AREVA responded that the time-step control scheme for S-RELAP5 is described in Section 2.6.7, of TR EMF-2100(P) (Reference 5). Sensitivity to time-step size is described in Section 6.8.2 of the TR. Guidance is provided in selecting the requested time-step sizes in the guidelines document (Reference 34). These documents indicate that the major criteria for determining time-step sizes are based on the material Courant limit and the measure of mass error. The measure of mass error determines whether the next time-step size is to be increased, reduced, or unchanged. The material Courant limit defines the maximum limit on the time-step size. Additional limits on the time-step size are established through user input, and if the code-calculated allowable time-step size is larger than the user-requested time-step size, then the code will adhere to the user-requested limit. This is standard practice and the NRC staff therefore finds the response acceptable.

3.3.4.2 Physical Modeling

As discussed in the opening to Section 3.3.4, nodalization within the AURORA-B EM is described as including all computer code input necessary to represent the physical plant and any engineered features influencing plant performance (e.g., trips and control systems, pumps, and component performance), and the EM relies on a consistent approach to defining the nodalization in order to reduce the potential for subjective nodalization and modeling decisions to influence the results of plant analyses. The technical guidance that accomplishes this objective establishes a consistent approach for the following parameters:

- Nodalization of both hydrodynamic components and heat structures (including their connections), plus flow and pressure boundary conditions
- Modeling practices for components and processes (including selection of phenomenological code models)
- Control variable and trip definition, and their use of material properties
- Initialization of the components and structures

Reviewing these parameters, the NRC staff identified in ARAI-38 that the AURORA-B TR did not specifically address the control system models, and that at a minimum it should address: (1) the control system model requirements from TR ANP-2830P, "Control System and Reactor Protection System Requirements for Modeling BWR Events" (Reference 26), (2) the control system model assessment base, and (3) proper code options, boundary conditions, and CCD interfaces for the control system model. AREVA's response indicated that the control system models are provided in the AURORA-B EM through arrangement of S-RELAP5 control system blocks. To be more specific, the S-RELAP5 code has a full complement of control system elements inherited from the RELAP5 family of codes. This means the control system can access information from the thermal-hydraulic components, heat structures, and reactor kinetics for use in computations. Additionally, outputs can be supplied to these same components for manipulating problem execution. For example, pressure, fluid temperature, and junction flow rates can be specified for time-dependent hydraulic components; heat flux and deposited power can be controlled for heat structures; and a reactor scram can be initiated by the S-RELAP5 trip system with the control rod positions being controlled through the reactor kinetics module. With proper arrangement, the control system blocks are capable of modeling responses to selected events when supplied with plant-specific input parameters.

AREVA's response continued to provide further detail regarding the nature of the control blocks and the specific components and associated component parameters that they adjust. From a very high level, the RELAP5-based control blocks connect to each other and share information via standard logic and mathematical relationships (e.g., sum-difference, multiplier, delay, constants, etc.). By utilizing these relationships, a system of control blocks can be configured for plant-specific applications to the level of detail necessary to simulate plant control system responses. The system of control blocks can therefore monitor parameters during code execution and, when necessary, simulate actuation of control valves, pump trips, etc. This is not an uncommon approach in modern system code packages. In fact, qualification exists for the individual S-RELAP5 control blocks in approved non-LOCA methodologies (Reference 8), albeit in the form of PWR control systems. Therefore, given the NRC staff's review of the information provided in the response and the aforementioned qualification, the NRC staff finds the use of S-RELAP5 control blocks to model BWR-specific control systems acceptable.

In RAI-4, the NRC staff requested additional information for how uncertainties are addressed relative to the control system modeling identified in Table 5-4 of the TR. AREVA responded that if the uncertainty or variability of the parameter **[**

]. In addition, the reactor protection system and its response are specified in a conservative manner; the key parameters are controlled by the plant technical specifications and/or analytical limits that have been defined by various setpoint methodology documents. **[**

] The NRC staff finds this response acceptable since it ensures a conservative approach is used if parameters have a significant impact.

Nodalization of both the model and validation cases is discussed in the TR. However, the NRC staff noted some nodalization schemes for specific applications of the EM will be either user-defined or are not prototypic between the validation cases and EM application. In RAI-5 and RAI-8, the NRC staff requested additional information related to **[**
] and differences between the plant model and validation cases. AREVA responded that modeling guidelines are established to provide a consistent approach to nodalizing the EM so that variation in results due to subjective nodalization and modeling decisions (i.e., the "user effect") is minimized. Assessment models used to simulate test facilities generally utilize the same modeling approach as the intended full-scope plant simulations, except where non-prototypical geometry of the test facility requires otherwise. Because both the plant models and validation cases generally utilize the same modeling approaches intended for actual plant simulations, thus minimizing the "user-effect," the NRC staff finds this response acceptable.

In addition to the information discussed above, the NRC staff also requested in RAI-8 additional information for what nodalization sensitivity studies are planned, or have been completed, to verify that guidance provided for the AURORA-B EM nodalization is adequate (with respect to convergence and accuracy) for the transients and accidents within the scope of TR ANP-10300P. AREVA responded that nodalization studies were performed for the main steam lines and reactor vessel. AREVA provided results of these nodalization studies in the response to RAI-8 and also referenced the response to ARAI-24 in support its conclusion. In summary, the BWR system nodalization was iteratively refined during the development process while maintaining a focus on accommodating geometric features, **[**

]. The nodalization study demonstrated the minimum number of nodes needed and the minimum time-step size required to obtain a converged solution for individual and composite steam line models based on simulation of the Peach Bottom Turbine Trip Test 2. Of specific note is that, **[**

], these have specific, conservative nodalization requirements delineated in Section 5.2.8.1 of the TR, which for example, **[**

]. The NRC staff reviewed the nodalization scheme proposed by AREVA and concluded that this scheme would likely **[** **]** under consideration. In light of the reasons discussed above, AREVA **[**

]. Because these results demonstrate a methodical approach towards determining a nodalization scheme and time-step size that ensures a converged solution, the NRC staff finds AREVA's responses acceptable.

Based on the discussions above and the acceptability of the new models implemented to support the application of AURORA-B to BWR analyses (see Section 3.3.4.3, below), the NRC staff concludes that the AURORA-B EM incorporates adequate models for simulating the BWR primary system.

3.3.4.3 Validity of New Models and Field Equations

The S-RELAP5 computer code has been previously approved by the NRC for PWR large- and small-break LOCA analysis and PWR non-LOCA transient analysis (References 7, 6, and 8). A summary of updates made to S-RELAP5 since approval of the RLBLOCA methodology for PWRs was provided in the TR. AREVA identified that the RLBLOCA methodology was chosen as the reference point for this effort because it contains the comprehensive code assessment and documentation, and because the methodology development followed the code, scaling, applicability, and uncertainty process (Reference 23).

In order to support BWR analyses, the S-RELAP5 code and its verification were extended beyond what was described in the RLBLOCA methodology documentation (Reference 21). As mentioned in Section 3.3.4 of this SE, the major changes made to S-RELAP5 for BWR applications and presented in the TR are:

- A new jet pump model
- Improved interfacial drag and heat and mass transfer models
- Improved mechanistic BWR separator model
- New pressure drop models for BWR fuel assemblies
- Inclusion of BWR CPR correlations

In the TR, AREVA identified that the S-RELAP5 code changes had been implemented without affecting the existing capabilities and approvals of the code. Application of the specific models associated with an approved methodology is controlled through the methodology option selected by the user. AREVA asserts that the same basic S-RELAP5 code version that supports AURORA-B has the capability to support all of the other approved methodologies described herein. Therefore, the S-RELAP5 code qualifications and vast experience from previous methodology applications is generally applicable to the AURORA-B EM, with the exception of the differences arising from improvements summarized in the TR for BWR application. In this regard, the NRC staff emphasizes that its present review applies only to the application of AURORA-B as described in TR ANP-10300P. The NRC staff did not review and does not approve any changes to AURORA-B or any of its components in the context of any other regulatory applications. Existing regulations (e.g., Appendix B to 10 CFR 50, 10 CFR 50.59) require that AREVA ensure that changes implemented to AURORA-B and its components to support one application have no adverse impacts on other approved applications.

With the exception of the CPR correlations, the NRC staff's assessment of the new models is discussed in the following subsections. The CPR correlations are discussed in Section 3.3.1.5 and Section 3.5.4.4 of this SE.

3.3.4.3.1 Jet Pump Model

The BWR jet-pump model used in S-RELAP5 assumes a jet pump can be treated **[**

]

Test data from **[** **]** was used to develop or assess the performance of the jet-pump model. The data was available from a mixture of sources: bench top tests for reduced scale and production jet-pumps, "component tests" performed within integral test facilities, and in-situ measurements within operating reactors. The assessment results show excellent agreement between the predicted and measured jet-pump performance over six different flow regimes that are defined based on the flow direction in each leg of the jet pump (i.e., drive, suction, discharge; referred to as 1+, 2+, 3+, 1-, 2-, and 3- of a Karman-Knapp diagram). In addition, transient data from the **[ ]**. Additional details and discussion of these tests are found in Section 3.5.4.2. Based on the agreement between the predicted and measured results, the NRC staff concludes that the jet-pump model within S-RELAP5 can adequately simulate jet-pump performance.

3.3.4.3.2 Interfacial Drag and Heat Transfer Model

In S-RELAP, the interphase coupling terms are formulated within the framework of the two-fluid model and are constructed from the formulations for the basic elements of flow patterns such as bubbles, droplets, vapor slugs (i.e., large bubbles), liquid slugs (i.e., large liquid drops or chunks of liquid), liquid film, and vapor film. Three principal flow-regime maps are used in S-RELAP5: (1) vertical flow map for elevation angles greater than 30 degrees, (2) horizontal flow map for elevation angle less than or equal to 30 degrees, and (3) high-mixing flow map for pumps.

Some flow regime transition criteria were modified to make them consistent with published data. Partition functions for combining different correlations and for transitions between flow regimes were developed based on physical reasoning and code-to-data comparisons. The vertical stratification model that was originally derived from ANF-RELAP was further improved. Also, the approximation to the Colebrook equation of wall friction factor from RELAP5/MOD2 was **[ ]** The use of these correlations and equations is standard within the industry.

Evaluation of the interfacial drag model included PIRT phenomena **[ ]** (discussed in Section 3.5.2.3). Validation of interfacial drag modeling **[ ]** (see Sections 3.5.2.1 and 3.5.2.2), and **[ ]** (see Section 3.5.5.1). Validation of the interfacial drag modeling **[ ]** (discussed in Section 3.5.2.3).

The details of these evaluations are more thoroughly discussed in each of the subsections indicated above. However, a brief summary of these evaluations is provided here. The core void distribution assessment database includes **[ ]**. These tests were performed for a wide range of flows and for system pressures (**[ ]**). The assessment results show excellent code-to-data agreement for these rod bundle void tests. The standpipes and steam separator and upper plenum assessment data base **[ ]**. The experiments were performed for a wide range of system pressures (**[ ]**)

and flow conditions that cover the operating conditions expected in the BWR steam separator standpipes and upper plenum. The assessment results show reasonable agreement between the predicted and measured void fractions. As a result, the NRC staff concluded that these models, as implemented in S-RELAP5, can adequately predict the interfacial drag phenomenon.

3.3.4.3.3 BWR Steam Separator Model

The S-RELAP5 model for BWR steam separators is based on the theory presented in NUREG/CR-2574 (Reference 24), as well as NUREG/CR-5535 (Reference 25). The separator is composed of three main parts: standpipe, separating barrel, and discharge passages. The two-phase mixture flows from the standpipe through a set of stationary swirl vanes to the separator barrel. The vanes generate a high rotational velocity component in the flow. The resultant centrifugal force separates the vapor-liquid (steam-water) mixture into a liquid layer on the inner wall of the separating barrel and a vapor core. The liquid layer on the wall is directed into the discharge passages through pick-off rings that are axially stacked into stages. Both two-stage and three-stage steam separators are implemented in the S-RELAP5 BWR steam separator model. Improved correlations for the radial void distribution are included in the latest version of the EM.

Experimental data from full-scale production steam separators has been used to develop and assess the performance of the steam separator model implemented in S-RELAP5. As mentioned above, both two-stage and three-stage separator designs are considered in S-RELAP5. The two-stage design is used in BWRs/2-5 while the three-stage design is used in BWR/6 plants. Evaluations of these models are discussed in detail in Section 3.5.4.3. To summarize them, there is reasonable code-to-data agreement for the carryunder and excellent agreement for the pressure drop. In addition, the results for carryover show reasonable to excellent agreement with the data. Based on this agreement and the observations that the physical models have their foundations in NUREG/CR-2574 and NUREG/CR-5535, the NRC staff concluded that the steam separator models are acceptably implemented within S-RELAP5 and can adequately predict steam separator performance.

3.3.4.3.4 BWR Fuel Bundle Pressure Drop Model

A new local form-loss correlation was developed for BWR fuel assembly spacer grids based on the pressure drop test data of **[** **]**. In addition to the local form losses from spacer grids, the pressure drop across the reactor core depends also on wall friction. **[** **]**.

The closure relations that define pressure drop within a rod bundle in S-RELAP5 have been assessed over a broad experimental database that includes the 7×7, 8×8, 9×9, and 10×10 fuel designs, the egg-crate and **[** **]** The results of these tests are discussed in more detail in Section 3.5.4.1. In summary, the pressure drop test data used to develop and assess the local form-loss correlation [

]. The results show reasonable to excellent code-to-data agreement. Therefore, the NRC staff concluded that the broad range of data and reasonable to excellent code-to-data comparisons demonstrate the models are acceptably implemented within S-RELAP5 and adequately predict bundle pressure drop.

3.3.5 Coupling of Component Calculational Devices

As discussed in Section 1.1 of this TR, the CCDs comprising the AURORA-B EM are S-RELAP5, MB2-K, MICROBURN-B2, and RODEX4. Together, these CCDs model the four phenomenological areas of the AURORA-B EM: BWR core and vessel thermal-hydraulics, BWR primary system modeling, BWR core neutronics, and thermal-mechanical fuel performance for BWR fuel. Each of these modeling areas relies on inputs from and is inherently dependent upon the other areas. This drives the need for the multi-physics, multi-code system to iterate between the modeling areas in order to calculate the parameters of interest. If not performed properly, the coupled nature required of the CCDs for the AURORA-B EM to iterate between modeling areas can be prone to inaccuracies in the calculated parameters at best and code instabilities at worst. Therefore, the NRC staff examined the nature of the CCD coupling and assessed its adequacy.

Of the four phenomenological areas, the BWR thermal-hydraulics and the primary system modeling are both contained within S-RELAP and utilize implicit and semi-implicit schemes operating at the same time steps. The remaining two areas, core neutronics and fuel thermal-mechanical performance, are modeled within MB2-K and RODEX4, respectively, and the NRC staff observed that the coupling of these two CCDs to S-RELAP5 is explicit in nature. Since explicit coupling can result in code instability and inaccuracy if not carefully executed, the NRC staff requested justification in RAI-63 that the chosen coupling frequency does not result in code instabilities and that the passing of any instantaneous rate parameters between the CCDs does not result in increased numerical inaccuracies.

AREVA's response to RAI-63 indicates that the time-step control algorithm for the coupling of MB2-K and S-RELAP5 is designed [

] The AURORA-B TR discusses this further in Section 5.2.7.2. [

]

With regard to RODEX4 and S-RELAP5, **[**

]

In both of the coupling cases discussed above, AREVA's response indicated that **[**

]. The NRC staff finds that this approach helps minimize the potential for code inaccuracy. Additionally, AREVA stated that **[** **]** coupling scheme between the point kinetics (MB2-K) and heat conduction (S-RELAP5) solutions has been empirically proven and has been in use for decades. While there is room for interpretation of the phrase "empirically proven," the NRC staff acknowledges that this coupling scheme and the RODEX4/S-RELAP5 coupling scheme are standard approaches that have been employed by the industry for quite some time and have, in the NRC staff's experience, produced acceptable results.

Furthermore, Section 6.8.2 of the TR discusses time step sensitivity studies of the code coupling schemes for the system scale and single channel models. Compared to the nominal time step size **[** **]**, these sensitivity studies demonstrate insignificant to minor impacts (**[** **]**) on the FoMs when time steps **[** **]** are used for both codes within the coupling schemes. Additionally, **[**

]. The only exception to this was an increase in ΔMCPR in some simulations for the MB2-K/S-RELAP5 coupling at time steps [ ]. Based on these studies, AREVA selected [ ] as the acceptable time step range for analyses. However, AREVA indicates in the TR that time-steps sizes larger than [ ] may be used for analyses if the sensitivity studies demonstrate the impact on the pertinent FoMs is conservative. Based on the minor and conservative impacts demonstrated by the sensitivity analyses, the NRC staff finds the time step ranges acceptable.

In conclusion, because the coupling schemes are implemented in such a fashion as to minimize code instabilities and inaccuracies, and because the coupling schemes are industry-standard approaches that are expected to have minor or conservative impacts on the FoMs with changes in time-step size, the NRC staff finds the coupling of the CCDs within AURORA-B to be acceptable. This acceptance is contingent upon AREVA's implementation of time-step sizes that have conservative impacts on the FoMs for a given analysis as discussed in Section 6.8.2 of the TR.

3.4 Accident Scenario Identification Process

Following the review guidance provided in Chapter 15.0.2 of the SRP, the third area of review for transient and accident analysis methods focuses on the accident scenario identification process. The associated acceptance criteria indicate that this process should identify and rank the reactor component and physical phenomena modeling requirements based on their importance to the modeling of the scenario and their impact on the FoMs for the calculation. It must include evaluation of physical phenomena to identify those that are important in determining the FoMs for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process.

Restated in terms of the review procedures provided in Section III of Chapter 15.0.2 of the SRP, it must be determined if the process of accident scenario identification for a given submittal is a structured process, and that the dominant physical phenomena influencing the outcome of a transient or accident scenario are correctly identified and ranked. The identification of transient and accident scenarios and the ranking of their associated physical phenomena support accurate prediction of the FoMs. Since it is the FoMs that a code is ultimately predicting for evaluation against acceptance criteria, ensuring that the dominant physical phenomena for a given scenario are correctly identified, ranked, and modeled with sufficient fidelity is vital.

3.4.1 Target Scenarios

As documented in the TR, AURORA-B is designed to be a comprehensive EM for analyzing a wide range of BWR events. The intended range of scenarios for which the AURORA-B EM is to be applied include AOOs, the recirculation pump rotor seizure and shaft break accidents (SRP 15.3.3 and 15.3.4), and ATWS overpressurization (only up to the time of boron injection). The range of target scenarios is discussed in greater detail in Section 3.1 of this SE.

Ultimately, AREVA envisions receiving NRC approval to use the AURORA-B code system in conjunction with an acceptable evaluation model for the analysis of a broad range of BWR analysis applications, including AOOs and infrequent events, RIAs, and LOCAs. Thus, AREVA ultimately intends to cover the vast majority of BWR AOOs and accidents with AURORA-B based methods. However, the breadth and depth of detail required for completing a methodology covering this range of events is such that the process has been broken into multiple development stages. It is therefore noted that the NRC staff's evaluation of the AURORA-B EM as documented in this SE is constrained to the range of scenarios identified in the current TR (see Section 3.1 of this SE). Specifically, the NRC staff assessed the adequacy of the phenomenon identification and ranking that AREVA performed with respect to these particular scenarios only. The identified phenomena and their ranking may not be adequate and are not applicable for other scenarios identified for possible future TR submittals.

3.4.2 Phenomenon Identification and Ranking

As part of the TR submittal, AREVA performed a phenomenon identification and ranking for the BWR target scenarios discussed above (Reference 27). AREVA's review identified the phenomena that must be addressed to analyze the BWR events within the scope of TR ANP-10300P using the AURORA-B EM and ranked their importance. Generated from this process and included in the TR was a PIRT, which defines the application of AURORA-B for analysis of AOOs and certain infrequent event and PA scenarios. The PIRT does not address other applications, such as RIAs, ATWS instability, and LOCAs.

The phenomena associated with the plant types of interest and applicable target scenarios are identified and described in TR ANP-2831P (Reference 27).

Section III of Chapter 15.0.2 of the SRP makes note that a PIRT is one example of an acceptable structured process for identifying and ranking phenomena applicable to a given target scenario. As such, the NRC staff finds AREVA's approach to a structured process acceptable.

However, in order to assess the adequacy of the PIRT, the NRC staff requested information regarding the manner in which it was vetted. Specifically, RAI-64 sought additional discussion regarding the PIRT committee make-up relative to the independence of the review committee and the development team. AREVA's response provided details about the PIRT evaluation process, indicating that it has evolved over a significant period of time involving several groups of contributing individuals. Succinctly, a preliminary PIRT development team comprising more than 175 years of total applicable experience generated the first iteration of the PIRT in 2004, while a second group with more than 150 total years of experience finalized it in 2009. In both the preliminary and final cases, the document was reviewed by a separate in-house committee of BWR application engineers, and updated to reflect their feedback. Given the diversity of the members of each of these groups and the iterative process involved in generating the PIRT, the NRC staff finds AREVA's response to be acceptable.

The NRC staff also performed a review of the PIRT and had questions regarding AREVA's choice in the identification and ranking for a number of phenomena. In particular, the NRC staff noted in ARAI-41 that several phenomena that may impact transient analyses were not addressed, including:

1) Countercurrent flow limiting breakdown
2) Core or bypass flashing
3) Jet pump flow reversal
4) Lower plenum flow stratification
5) Safety relief valve flow and critical flow
6) Natural circulation
7) Droplet field effects

AREVA's response provided justification for all these phenomena:

1) The Countercurrent flow limiting (CCFL) breakdown phenomenon can occur in a number of locations in the vessel after a LOCA, but it does not occur during any of the targeted non-LOCA transients.

2) Flashing of superheated liquid in the core is mainly controlled by **[**

   ]

3) BWR jet pumps **[** **]**. Jet pump flow reversal is one of the possible flow regimes that may result from the pressure difference between the legs of the jet pump, and **[** **]**.

4) Lower plenum flow stratification is considered in **[** **]**, which is highly ranked for the target scenarios of interest.

5) Safety relief valve flow **[** **]**. Additionally, the critical flow model implemented in S-RELAP5 has been validated against **[** **]**.

6) Natural circulation is an integral process that is defined by specific phenomena, each of which is addressed in the PIRT and each of which **[** **]**. It is controlled by a balance between buoyancy, friction, and form losses in all components within the pressure vessel. Friction and form losses depend on model inputs from plant data and are reflected as "delta pressure" for components in the PIRT. Buoyancy depends mostly on **[**

]

7) The droplet field effects are considered primarily with constitutive models for interfacial heat and mass transfer and interfacial drag. [ ]

With the exception of safety relief valve flow and critical flow, the identified phenomena are covered by existing [ ] PIRT phenomena, have been sufficient validated against test data, or are not applicable for the target scenarios of interest. Therefore, even though they were omitted from the PIRT, the NRC staff finds the response acceptable. However, the NRC staff does not agree with AREVA's response regarding safety relief valve flow and critical flow. In particular, this phenomenon has the potential to significantly impact the system response for certain transient classes, including pressurization events. [ ]. Therefore, the NRC staff finds it necessary for licensees implementing the method in TR ANP-10300P [ ]. In this regard, the NRC staff notes that safety/relief valve critical flow is typically modeled by tuning input parameters until the critical flow model in the thermal-hydraulic code reproduces a rated design flow rate at the specified design pressure. Subsequently, in the transient calculation, the critical flow model computes the critical flow based on the calculated development of system thermal-hydraulic conditions. As a result, it is apparent that the demonstration of conservatism on a plant-specific basis will in general rely upon both the specification of a conservative rated design flow and the accurate or conservative physical modeling of critical flow from safety/relief valves in the AURORA-B code model.

The NRC staff also sought justification in ARAI-44 for the [ ] for bypass boiling, which affects instrument response and rod-to-rod peaking. Additionally, it was postulated to be larger at EFW conditions than at EPU conditions. AREVA's response indicated that core bypass average exit void fraction is very small under normal EPU conditions and [ ]. Steady-state calculations performed by [ ] yielded values of [ ], respectively, for EPU and EFW conditions. Additionally, AREVA demonstrated that increasing the pre-transient core bypass average exit void fraction from [ ] had no effect on [ ] for limiting events. [ ] is the reason for the aforementioned phenomenon's ranking for all events [ ]. Based on the demonstrated [ ] for non-depressurization transients and the [ ] for depressurization transients, the NRC staff finds the response acceptable.

The NRC staff has assessed AREVA's phenomenon identification and ranking that has been performed and manifested in the PIRT. Based on the discussions above, which serve to satisfy the review criteria presented in SRP Chapter 15.0.2, the NRC staff finds the process is applicable to the target scenarios identified in the TR (as described in Section 3.1 of this SE). The NRC staff notes, however, that the identification and ranking is not approved for and may not be applicable to other scenarios. For example, the CCFL phenomenon is significant in LOCA scenarios, to which the AURORA-B EM described in TR ANP-10300P is not applicable. In light of this, future submittals that seek to expand the range of target scenarios that AURORA-B is applicable to will entail additional review effort in this area.

The result of the PIRT evaluation process is a ranking of each phenomenon as “high,” “medium,” “low,” or “N/A” when considering its impact on the FoMs for *all* scenarios. In the case of the AURORA-B EM, the PIRT presented in TR ANP-10300P is a “summary-level PIRT,” and it was created by accumulating the maximum ranking of each phenomenon from a series of individual event-specific PIRTs (called the “event PIRTs”). The event PIRTs are documented in the phenomenon identification and ranking reference, TR ANP-2831P (Reference 27). It is important to note that during the accumulation of phenomena from the individual event PIRTs, two exceptions were made to the manner in which phenomenon importance was assessed. First, phenomenon rankings for events considered to have *low* safety significance in determining the ΔMCPR FoM were reduced by one ranking level (e.g., phenomena ranked high for an event with low safety significance for the MCPR FoM were reduced to medium). Second, phenomenon rankings for events with *N/A* safety significance in determining the ΔMCPR FoM were reduced by two ranking levels (e.g., phenomena ranked high for an event with *N/A* safety significance for the ΔMCPR FoM were reduced to low). These two exceptions tend to focus the PIRT on those phenomena most important in terms of safety significance for the ΔMCPR FoM. They play a large role in the modeling of pressurization transient scenarios, which are generally limiting for MCPR. As a result, each of those phenomena from the PIRT that has been ranked as “high” for one or more scenarios will be included in the determination of upper bound FoMs for safety analysis. The NRC staff was not convinced of the acceptability of AREVA’s practice of reducing PIRT rankings based upon a generic, *a priori* expectation of event significance. However, AREVA stated, and the NRC staff confirmed, that this practice did not reduce the ranking of any phenomenon that would have otherwise been ranked high or medium in the summary-level PIRT. Additionally, several of those phenomena from the PIRT that have been ranked “medium” and found to be pertinent to non-pressurization transient scenarios will also be included in the determination of the upper bound FoMs for safety analysis. This is further discussed in Section 3.6.

3.4.3 Figures of Merit

As mentioned in Section 3.4, the identification and ranking of the dominant physical phenomena associated with each scenario (in this case, via the PIRT) determines which of the phenomena are expected to be the most influential for accurately predicting the FoMs. Having reviewed the PIRT, the NRC staff examined the chosen FoMs and assessed their relevancy with respect to the ranked phenomena. The FoMs that are calculated by AURORA-B are:

1. Transient change in MCPR (ΔMCPR), as quantified through the limiting ΔCPR response for the event.

2. Peak system pressure.

3. Time-dependent nodal power.

4. Peak cladding temperature.

5. Maximum local oxidation.

The first FoM is ΔMCPR, and it is used to determine the event MCPR. **[**

].

Evaluation of event MCPR is performed for two different classes of analyses. The first, which generally applies to AOO events, is for providing assurance that the event MCPR remains above the safety limit MCPR for each fuel design in the core. The second class, which generally applies to accident conditions, is used in conjunction with other methods to determine radiological consequences in cases where the event MCPR is below the safety limit, and thus fuel cladding integrity cannot be ensured.

The second FoM is peak system pressure. This FoM is used to provide assurance that the maximum pressure experienced during a given event is maintained below the applicable limit. Depending upon whether the event is categorized as an AOO or ATWS scenario, the applicable limits and analysis assumptions differ.

The third FoM is time-dependent nodal power. This FoM is specifically used with fuel thermal-mechanical methodologies to evaluate transient strain and fuel centerline melt under transient conditions. As background, note that it is primarily the fuel thermal-mechanical analysis methodology that is used to evaluate 1) the transient strain of the cladding, with assurance that it does not exceed one percent (1%), and 2) fuel peak pellet temperatures to assure they remain below the melting point. However, these acceptance criteria oblige consideration of transient conditions. Depending on the bases of the fuel thermal-mechanical analysis methodologies, boundary conditions from “fast transient” and “slow transient” methodologies (specifically, the time-dependent nodal power FoM) may be used in part, to demonstrate compliance with the applicable acceptance criteria. To this end, the AURORA-B EM is applied to predict time dependent nodal power boundary conditions for input to fuel thermal-mechanical codes, such as RODEX4, within the bounds of the approved fuel thermal-mechanical methodologies. As discussed in Section 3.3.3.2, this passing of time-dependent nodal power to an external fuel thermal-mechanical code, such as the full RODEX4 code, is done because the fuel assemblies or assembly groups in the AURORA-B EM core model are not likely to be at the limiting exposure or peak LHGR conditions necessary for evaluating limiting transient cladding strain or peak fuel temperature.

As discussed in Section 3.4.2, the summary-level PIRT in ANP-10300P derives from a more detailed PIRT study documented in the phenomenon identification and ranking reference document, TR ANP-2831P (Reference 27). The PIRT effort documented in TR ANP-2831P is based primarily on the ΔMCPR FoM, but was modified by additional iterative assessments that incorporated phenomenon rankings associated with the determination of (1) system peak pressure and (2) PCT during an ATWS (prior to boron injection). During the course of the review, the NRC staff questioned whether a PIRT developed primarily for phenomena affecting the ΔMCPR FoM **[**

] Thus, ARAI-40 requested the basis for using **[** **]**. AREVA's response stated that the normalized transient power deposited in the fuel rods for each axial node in an assembly is a key input to evaluation of **[**

] The NRC staff agrees with this assertion and therefore finds the PIRT ranking of phenomena for the ΔMCPR FoM **[** **]**.

The fourth and fifth FoMs concern fuel integrity and are intended to ensure adequate core cooling capability: peak cladding temperature and maximum local oxidation. The AURORA-B EM described in TR ANP-10300P will be used to evaluate these FoMs for infrequent events and PAs other than LOCA. As mentioned in Section 8.3.2 of the TR, these FoMs are also determined during the ATWS scenario. Depending on the purpose of the analysis, the cladding temperature and oxidation criteria may be the goal of the analysis, or the cladding temperature may contribute as an element of a radiological consequences analysis. In such a role, the cladding temperature and duration of the temperature exceeding some value may be used in conjunction with other methodologies to demonstrate that the radiological consequences of the event do not exceed applicable limits from 10 CFR 100 or 10 CFR 50.67. The cladding temperature (and duration) criteria are plant-specific and event-specific, and they are defined in the licensing documentation for each plant.

The oxidation model used by the AURORA-B EM when determining the maximum local oxidation FoM is the Cathcart-Pawel correlation. The NRC staff has concluded that when making use of this correlation, an oxidation limit of 13 percent should be used, as discussed in Section 3.4.3.1 below.

Each of the FoMs discussed above is compared to acceptance criteria in order to ensure reactor safety. The acceptance criteria that these FoMs are compared to are:

1. Evaluation of the event MCPR for AOOs to ensure that the event CPR remains above the safety limit.

2. Evaluation of the radiological consequences to ensure that the calculated release of radioactivity does not exceed applicable 10 CFR 100 limits for selected infrequent events and PAs. For some events (e.g., recirculation pump rotor seizure) the evaluation must show that the calculated release does not exceed a small fraction (typically 10 percent) of the applicable limits. For plants approved for alternate source term, the limits specified in 10 CFR 50.67, “Accident Source Term,” are applicable.

3. Evaluation of peak system pressure to ensure that peak pressure during AOO scenarios is maintained below 110 percent of design value, in compliance with ASME limits.

4. Evaluation of peak system pressure to ensure that peak pressure is limited such that the maximum primary stress within the reactor coolant pressure boundary does not exceed the ASME Service Level C limits during the ATWS scenario. For BWRs, this is generally a maximum reactor vessel pressure of 1500 psig.

5. Evaluation of the transient strain of the cladding to assure that it does not exceed one percent. Also, fuel peak pellet temperatures are evaluated to assure they are maintained below melting.

6. Adequate core cooling capability is ensured by meeting the cladding temperature and maximum local oxidation criteria utilized for LOCAs as specified in 10 CFR 50.46 (i.e., peak cladding temperature not exceeding 2,200 °F and the local oxidation of the cladding not exceeding 17 percent of the total cladding thickness. Because of the use of the Cathcart-Pawel oxidation correlation, a 13 percent oxidation criterion should be used in place of the 17 percent value, as discussed in Section 3.4.3.1.

Adherence to these acceptance criteria ensures that SAFDLs are met, that the reactor coolant system pressure boundary is not challenged, and that radiological releases for PAs do not exceed limits set forth 10 CFR Part 100 or 10 CFR 50.67, as applicable for the given event. Each of the FoMs serves to demonstrate compliance with one or more of these criteria. Additionally, each of the target scenarios for which AURORA-B is intended could potentially challenge the acceptance criteria for one or more FoMs. Therefore, based on this and the discussions found above and in Section 3.4.3.1 regarding their utilization, the NRC staff finds the choice of FoMs acceptable.

The NRC staff also assessed the phenomena contained within the PIRT and their rankings based on the target scenarios and determined they correctly represent the influence they exert on the FoMs. With regard to the fidelity of the models for simulating the ranked phenomena and predicting the FoMs, this is assessed in Section 3.5 of this SE.

3.4.3.1 Cathcart-Pawel Oxidation Correlation Relative to 10 CFR 50.46 Criteria

The requirements set forth in 10 CFR 50.46(b)(2) regarding adequate cooling capability impose a limit on cladding oxidation of 0.17 times the total cladding thickness before oxidation. The oxidation limit is usually considered as a percentage, and it can also be expressed as

Equivalent Cladding Reacted (ECR) (i.e., 17 percent ECR). The experimental studies supporting this limit evaluated cladding ductile performance and correlated it to the thickness of the differing layers (i.e., oxide, brittle zirconium, ductile zirconium) rather than to a measured ECR. The percentage values were calculated, based on the test conditions, using the Baker-Just correlation. Thus, the AEC noted that "the Regulatory Staff in their concluding statement compared various measures of oxidation (page 90) and concluded that a 17 percent total oxidation limit is satisfactory, *if calculated by the Baker-Just equation*" (6 AEC 1097).

The AURORA-B EM makes use of the Cathcart-Pawel correlation, which is also considered acceptable for best-estimate calculations of metal-water reactions, as documented in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis" (Reference 40) and RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling Performance" (Reference 41). In fact, both recommend the use of the Cathcart-Pawel correlation based on its superior accuracy when compared to Baker-Just. However, as noted in Research Information Letter (RIL) 02-02, Attachment 2 (Reference 42), the original and confirmatory ring compression tests upon which the 17 percent ECR criterion was based relied on an ECR value calculated using Baker-Just. Specifically, page 9 of RIL 02-02, Attachment 2, states "had the Cathcart-Pawel correlation – which did not exist at that time – been used, the cladding oxidation limit would have been about 13 percent. Therefore, the Baker-Just correlation must be used when comparing results with the old 17 percent limit."

As a result, the use of a 17 percent limit on ECR when applied to cladding oxidation values calculated using the Cathcart-Pawel correlation does not provide the same level of assurance of cladding ductility as the same limit when applied to a result calculated using the Baker-Just correlation. Therefore, the NRC staff is imposing a limitation specifying that the ECR results calculated using the Cathcart-Pawel correlation are considered acceptable and in conformance with 10 CFR 50.46(b)(2) if the ECR value is less than 13 percent, which is equivalent to 17 percent ECR if calculated using the Baker-Just equation.

3.5 Code Integral Assessment

Following the review guidance provided in Chapter 15.0.2 of the SRP, the fourth area of review for transient and accident analysis methods focuses on integral 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. Additionally, assessments must compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models.

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. In either case, some tests may not be full-scale, and, in demonstrating applicability to full-scale plant conditions, the tests may contain scaling distortions. These distortions can affect both local and overall elements. It is therefore necessary to examine the nature of the tests involved in the assessments.

Each of the CCDs within the AURORA-B EM has been assessed against integral and separate effect data and found to be acceptable for performing safety analyses during the review and approval of its individual TR. As a result, the majority of this section of the report will focus on the overall code assessment that was performed on AURORA-B to demonstrate that the overall EM provides adequate predictions of the phenomena of interest. However, some discussion of the qualification of these CCDs will be provided in this section, especially with regard to the new and modified models introduced within them to simulate BWR phenomena.

3.5.1 Range of Code Assessment

The AURORA-B EM and application methodology were developed in compliance with the Evaluation Model Development and Assessment Process (EMDAP) defined in RG 1.203 (Reference 17). RG 1.203 and the EMDAP are designed to be complementary to the guidance provided to NRC reviewers in SRP Chapter 15.0.2, and EMDAP has been defined by the NRC as an acceptable process for developing and assessing EMs that may be used to analyze the behavior of a nuclear power plant. Therefore, in order to facilitate meeting the review requirements put forth in the SRP (as discussed above), AREVA structured the assessment of the AURORA-B EM in the TR such that it presents four general areas that each correspond to an assessment step within the EMDAP. These general areas are: the model's fidelity and accuracy (EMDAP step 14), the model's ability to represent processes and phenomena (EMDAP step 16), the model's ability to accurately simulate system components (EMDAP step 17), and the assessment of system interactions and global capability (EMDAP step 18). The following subsections discuss, in turn, each of the four general areas of the EM assessment, and are based on the more detailed discussions in Section 3.3 of this SE.

As discussed above in Section 3.4.2, a PIRT analysis was performed to identify all the highly ranked phenomena that are modeled in the AURORA-B EM and their impact on the FoMs. These highly ranked phenomena are listed below in Table 3.2 in Section 3.6 of this SE. Each individual assessment that is performed in one of the four general areas serves to provide validation for AURORA-B's ability to model specific highly-ranked phenomena. The assessments described within the following subsections demonstrate that all the highly ranked phenomena listed in Table 3.2 are adequately modeled, with the exception of **[**

]

3.5.2 Model Fidelity and Accuracy

The first of the four areas presented for assessment of the AURORA-B EM is model fidelity and accuracy. A number of tests have been used to assess the performance of AURORA-B and its components. The results of these assessments and the evaluation of the AURORA-B EM are summarized in the following subsections.

3.5.2.1 Rod Bundle Void Tests

These tests address the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**
- **[ ]**

These tests and the comparisons of AURORA-B to these test results are discussed in Section 3.3.1.2. In summary, the evaluations presented in the TR, plus the responses to RAI questions that expand the discussion of the data comparisons and evaluations, show that the AURORA-B EM can be expected to give reasonable predictions of the void distribution in rod arrays.

3.5.2.2 Christensen Void Tests

These tests serve to assess the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**
- **[ ]**

These tests and the comparisons of AURORA-B to these test results are discussed in Section 3.3.1.3. In summary, the evaluations of these separate effects tests show that the AURORA-B EM can be expected to give reasonable predictions of the axial void distribution in a BWR vessel for components other than the fuel assemblies.

3.5.2.3 Allis-Chalmers Large Diameter Void Tests

These tests address the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**

[ ]. The selected tests for this assessment cover 3 different pipe diameters (2.9, 18, and 36 inches) and length/diameter ratios (16.5, 7.85, and 3.92). The experiments were performed for system pressures of 615 psia to 2,015 psia and flow conditions that cover the normal operating conditions expected in the BWR steam separator standpipes and upper plenum. The characteristics of the void fraction tests are summarized in Table 6-3 of the TR. The purpose of this assessment was to test the **[ ]** models in predicting void fraction distributions that occur with two-phase flow for conditions that encourage phase separation.

The S-RELAP5 models affecting these phenomena have been improved and the range of assessment has been expanded to cover these regions. The results presented from the Allis-Chalmers tests show **[**

] In light of this, the assessment results show reasonable agreement between the predicted and measured void fraction, suggesting the **[** **]** models possess adequate prediction capability. Evaluation of the uncertainty in predictions related to the highly ranked PIRT phenomena for which this data is invoked for support is addressed in Section 3.6 of this SE.

3.5.2.4 GE Level Swell Test

These tests address the following highly ranked PIRT phenomenon listed in Table 3.2:

- **[** **]**

[
] level swell is defined by the interactions of several lower-level phenomena. Thus, the purpose of this assessment was to test the two-fluid interfacial models in predicting the flow regimes and void fraction distributions that occur under level swell in depressurization conditions. The assessment also evaluates the interfacial drag and heat transfer models that contribute to predicting level swell. The key model affecting these assessments is the interfacial friction for the bubbly and slug flows. In order to better facilitate predicting these lower-level phenomena, the S-RELAP5 models affecting them have been improved within the AURORA-B EF.

The selected level swell test was essentially a small-break blowdown of a vertical vessel 14 foot (ft) high by 1 ft in diameter. The vessel was initially pressurized to 1,011 per square inch and filled with saturated water up to the 10.4 ft elevation. Comparisons between measured and calculated axial void fraction distributions at two different times (40 seconds and 100 seconds) show very good agreement; the calculated results fall within the experimental uncertainty. The calculated flow regimes are bubbly flow below the void fraction of 0.25, slug flow from the void fraction of 0.25 up to the two-phase mixture level position (between 0.3 to 0.6 void fraction), and annular-mist flow above the mixture level. The jump in void fraction from ~0.4 to ~0.99 that defines the location of the two phase mixture level is distinct with a smooth but sharp transition. To achieve this requires the interfacial friction models for slug flow, vertical stratification, and annular-mist flow all working in harmony. These assessment results indicate the lower-level phenomena are adequately modeled for the EM to acceptably predict the PIRT phenomenon. AREVA's statistical treatment of downcomer level swell is discussed further in Section 3.6.4.5 of this SE.

3.5.2.5 Summary of MB2-K and MICROBURN-B2 Qualification

Qualification of AURORA-B's neutronic CCDs addresses the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

The PIRT phenomena listed for MICROBURN-B2 and MB2-K **[ ]**. Based on the NRC staff's review of AREVA's responses to RAI questions related to MICROBURN-B2 and MB2-K, as discussed in Section 3.3.2.2, this is indeed the case. The MICROBURN-B2 code and the comparison to its assessment data has received prior review and was found to acceptably predict the above PIRT phenomena (Reference 10).

Based on the above discussion and an evaluation of MB2-K's performance with respect to numerical benchmarks (Section 3.3.2.3), the NRC staff assessed the adequacy of the neutronics CCDs to model the above PIRT phenomena within the AURORA-B EM and noted the following:

- **[ ]** show reasonable to excellent agreement with data and, for code-to-code comparisons, good agreement with higher order methods. Uncertainties in these parameters have been determined and are discussed further in Section 3.6.

- The assessment of **[ ]** has been made by comparison of calculated isotopic inventories to data. This is not impacted by the addition of kinetics modeling and is unchanged from the MICROBURN-B2 steady-state neutronics code for BWR application which has previously been approved (Reference 10).

- **[ ]** qualification of the CASMO-4/MICROBURN-B2 lattice physics code/steady state neutronics code system. **[** **]** the temperature of heavy nuclei in the fuel; an increase in thermal energy will result in broader neutron absorption resonances and a corresponding reduction in the number of thermal neutrons available for fission. CASMO-4 generates lattice neutronics data, such as temperature-dependent microscopic absorption cross-sections for the heavy nuclei in the fuel, and passes it to MICROBURN-B2. **[** **]**. CASMO-4 has been qualified for the generation of relevant cross-sections and effective resonance integrals, and MICROBURN-B2 has been qualified for the underlying neutronics models that combine all the various interaction probabilities into a reactivity impact (Reference 10). Additionally, MB2-K implements **[** **]** impacted by the addition of the MB2-K kinetics code because it is unchanged from the MICROBURN-B2 steady-state neutronics code for BWR application.

- **[** **]** are indirectly assessed by comparing code calculations to measured plant data. These data include TIP measurements of neutron flux performed periodically at BWRs supported by AREVA and eigenvalue trending where actual cycle operations are simulated to compare the calculated eigenvalue against the actual reactivity balance for operating BWRs. Comparisons indicate reasonable to excellent agreement of the calculated results with these "integral" data. Therefore, it is inferred that the MICROBURN-B2 core simulator code and underlying lattice physics method provide reasonable to excellent predictions of these phenomena.

The qualification information provided through prior approval (Reference 10) and the comparisons to numerical benchmarking (Section 3.3.2.3) demonstrate that MICROBURN-B2 and MB2-K perform acceptably over a wide range of conditions. The qualification information provided in AREVA's August 2011 submittal (Reference 4) and the updated revision of this submittal (Reference 52), as discussed in Section 3.3.2.4, demonstrates the codes' applicability range includes EPU and EFW conditions. On these bases, the NRC staff concludes the neutronic CCDs within the AURORA-B EM can acceptably predict the PIRT phenomenon listed above.

3.5.2.6 Summary of RODEX4 Qualification

Qualification of the RODEX4 CCD addresses the following highly ranked PIRT phenomenon listed in Table 3.2:

- **[** **]**

The full, stand-alone RODEX4 code and the comparison to its assessment data has already been reviewed and found to be acceptable. A subset of the routines from the full RODEX4 code, known as the "RODEX4 kernel," is contained within S-RELAP5 and is used to predict short-term transient fuel rod properties in a manner fully consistent with the stand-alone RODEX4 code (see Section 3.3.3), which provides initialization parameters. The RODEX4 kernel routines are mathematically equivalent to those of the full RODEX4 code and have been demonstrated to produce similar transient temperature results.

Based on this, it can be concluded that the full RODEX4 code, the RODEX4 kernel, and the models contained therein can perform best-estimate fuel performance predictions considering steady state operation and transient scenarios. The overall code performance shows excellent agreement with a broad database of fuel rod data, and is capable of accurately modeling heat release rates from the fuel rods to the coolant for the target scenarios identified in Section 3.1. Therefore, the NRC staff finds the RODEX4 CCD can acceptably model the highly ranked PIRT phenomenon listed above.

3.5.2.7 Transient Coolant Mixing

Transient coolant mixing is addressed through the following highly ranked PIRT phenomena listed in Table 3.2:

- **[** **]**
- **[** **]**

Transient coolant mixing **[**

]

The approach to transient mixing discussed above is highly dependent upon the proper selection **[**

]

AREVA responded in three parts. **[**

]

The NRC staff agrees with AREVA's first two responses. [

]

However, the NRC staff does not agree with AREVA's third response. [
], but the acceptability of any of these methods has not been determined by NRC review. Additionally, as highly ranked PIRT phenomena, the uncertainty associated with any degree [
] (further discussed in Section 3.6.1, Table 3.2 of this SE). The uncertainty is quite likely to vary from one [
] method to the next, and a generically applicable uncertainty has not been presented. As a result, the NRC staff is imposing a limitation and condition in Section 5 of this SE that the allowance of thermal mixing [
] on a plant-specific basis requires prior NRC review and approval.

Based on the discussion above, the NRC staff finds the approach to transient coolant mixing presented within TR ANP-10300P to be acceptable for modeling the highly ranked PIRT phenomena listed above. This acceptance is contingent upon the applicable limitation and condition.

3.5.3 Model Ability to Represent Processes and Phenomena

The second of the four areas presented for assessment of the AURORA-B EM is model ability to represent processes and phenomena. A number of tests and/or numerical benchmarks have been used to assess the performance of AURORA-B and its components. The results of these assessments and the evaluation of the AURORA-B EM are summarized in the following subsections.

3.5.3.1 TWIGL 2D Neutron Kinetics Cross Code Comparisons

The key point of interest in this cross-code comparison is that the neutron kinetics equations implementation in MB2-K is consistent with how other vetted codes implement similar equations. This suggests that AURORA-B can acceptably predict power changes in a manner similar to the compared codes. Additionally, as discussed in Section 3.3.2.3, the high degree of agreement with these other vetted codes with respect to the TWIGL benchmark provides support that MB2-K's numerical methods (and hence the AURORA-B EM) possess an acceptable amount of stability and precision in regards to neutronic phenomena.

3.5.3.2 LMW 3D Numerical Cross Code Comparison

The key item in this cross-code comparison is that the neutron kinetics equations implementation in MB2-K is consistent with how other vetted codes implement similar equations. In addition, the time step changes in MB2-K produce results similar to the compared codes, providing support that the time-integration techniques are sound (Section 3.3.2.3). This suggests that AURORA-B can predict power changes as well as similar codes.

3.5.3.3 LRA 2D and 3D BWR Control Blade Drop Transients Code Comparisons

This cross-code comparison supports that the nodal representation of power and the neutron kinetics equations implementation in MB2-K is consistent with how other vetted codes implement similar equations. Although the calculated powers between the codes are not consistent through the entire analysis, this set of comparisons indicates that the integrated response is similar between different codes. The three cross-code comparisons, described in Section 3.3.2.3, demonstrate that the kinetics implementation in MB2-K is similar to other well-known codes.

However, of prime importance is that this particular benchmark possesses an accurate peer-reviewed solution for the control blade drop transient, benchmark #14 of ANL-7416 (Reference 32). As discussed in Section 3.3.2.3, the results of MB2-K for these tests to the solution provided in the ANL-7416 possess a high degree of agreement, providing support that MB2-K's results are both precise and accurate.

3.5.4 Model Ability to Accurately Simulate System Components

The third of the four areas presented for assessment of the AURORA-B EM is model ability to accurately simulate system components. The fidelity and accuracy of these models was assessed within the TR largely through the use of component effects test data. Comparison to component effects test data helps demonstrate the capability of the AURORA-B EM to predict relevant characteristics of entire components and/or regions of the BWR plant (e.g., jet-pumps and steam separators). Each of the component effects tests selected also serves to address one or more highly ranked PIRT phenomena. The results of these assessments and the evaluation of the AURORA-B EM are summarized in the following subsections.

3.5.4.1 Rod Bundle Pressure Drop

The rod bundle pressure drop tests address the following highly ranked PIRT phenomenon listed in Table 3.2:

- **[** **]**

The selected tests for this assessment include the 7×7, 8×8, 9×9, and 10×10 fuel designs, and the egg-crate **[**

]
The purpose of this assessment was to evaluate the closure relations that define pressure drop within a rod bundle.

The assessment was performed by comparing calculated and measured pressure drop across prototypical fuel assemblies. The assessment results show excellent agreement between the predicted and measured pressure drop in two-phase flow **[** **]**. There is also excellent agreement between the predicted and measured exit void fractions, **[** **]**. The NRC staff therefore finds the AURORA-B EM provides acceptable predictions of the assembly pressure drop **[** **]**.

3.5.4.2 Jet-Pump Performance Tests

These tests address the following highly ranked PIRT phenomenon listed in Table 3.2:

- **[** **]**

The selected test data was taken from **[** **]** jet-pump assemblies. The data was derived from a mixture of sources; bench top tests for reduced scale and production jet pumps, "component tests" performed within integral test facilities, and in-situ measurements within operating reactors. The dimensional characteristics for the **[** **]** jet-pump assemblies cover the full spectrum of jet-pump designs found in operating jet-pump BWR plants, from BWR/3 through BWR/6, and **[** **]**.

[

]

Test data from **[** **]** jet-pump assemblies was used to develop and assess the performance of the jet-pump model implemented in S-RELAP5, **[** **]**. Use of these data to develop and assess the jet-pump model supports the scalability of the model from reduced scales, through a wide variety of full scale jet-pump designs. The assessment results show excellent agreement between predicted and measured jet pump performance; the model predictions **[**

], for which suction flow is entering the jet pump from the downcomer and drive flow is entering from the recirculation system. The

NRC staff therefore finds the AURORA-B EM provides acceptable predictions of the jet-pump delta pressure PIRT phenomenon.

3.5.4.3 Steam Separator Tests

These tests address the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**

The selected test data from full scale production steam separators was used to develop and assess the performance of the steam separator model implemented in S-RELAP5. Two different separator designs are considered in S-RELAP5: the "two stage" design used in BWR/2-5 plants, and the "three stage" design used in BWR/6 plants. The TR states that a description of the steam separator test facility from which these data were gathered can be found in Reference 43.

However, examination of this reference showed **[ ]**. In response to RAI-42a on this issue, AREVA stated that the data used in these assessments was actually from three proprietary reports from GE **[ ]** (Section 3.3.1.1). **[ ]**, they are available for review only during on-site audits. Examination of these documents allowed the NRC staff to verify the conditions under which the data was collected.

AREVA's assessment of the model results for carryunder and delta pressure, as initially presented in the TR, were "reasonable to excellent." The NRC staff's review of the results concluded there was indeed reasonable to excellent agreement for delta pressure predictions, **[ ]**

In order to better ascertain the nature of this behavior and discern the adequacy of the model results, the NRC staff requested in RAI-42b further information regarding how the separator models were constructed. During the course of the NRC staff's review, and in conjunction with developing a response to the RAI, AREVA **[ ]** and presented the results, along with model descriptions, in the RAI response (Reference 15). The updated assessment results demonstrate **[ ]** the results initially presented in the TR. For carryunder comparisons, the NRC staff found that the updated assessment results show reasonable to excellent agreement between the predicted and measured carryunder **[ ]**. The predicted results clearly and closely follow the measurement trends. AREVA imposed **[ ]** prediction error bounds on the results for the 2-stage and 3-stage separators, respectively. The NRC staff performed an analysis of the data that showed **[**

], which would correspond to a 2 sigma (σ) uncertainty for a normal distribution (the NRC staff analysis demonstrated the error distribution for each test could be reasonably approximated by a normal distribution). These results tend to corroborate AREVA's bounds. The NRC staff's analysis also found a **[**

] and serves to illustrate the precision with which the **[ ]** models predict carryunder.

Although not a highly ranked PIRT phenomenon, the separator carryover influences the predictive capability of the separator model. Therefore, carryover assessment results were also provided. The results for carryover also show reasonable to excellent agreement with the data. As with the carryunder results, the carryover results also showed **[**

]
assessment results, the NRC staff finds the AURORA-B EM provides acceptable predictions of the steam separator carryunder and delta pressure PIRT phenomena.

3.5.4.4 Critical Power Tests

The critical power tests address the following highly ranked PIRT phenomena from Table 3.2:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

In the TR, assessment for the PIRT phenomena **[**
] was performed by comparing the calculated and measured time to boiling transition for typical transient scenarios using the tested rod-to-rod relative peaking (as it defines the "F-effective" or "K-factor" input to the critical power correlation). This approach also allows for indirect assessment of the **[ ]**; the transient response of fluid quality is affected by the transfer of heat and mass between liquid and vapor phases, which is an input to the determination of CHF. The test measurements of time to boiling transition and the comparisons of these results with AURORA-B predictions are discussed in Section 3.3.1.5 of this SE. In summary, the assessment in the TR adequately demonstrates that the EM provides conservative evaluations relative to boiling transition in transients when using the SPCB and ACE CPR correlations. Therefore, the NRC staff finds the EM can acceptably predict the three PIRT phenomena discussed above when used in conjunction with these approved CPR correlations.

[

] AREVA's response to RAI-37 provides additional detail about [ ] used in the AURORA-B EM described in TR ANP-10300P. In particular, the RAI-37 response indicates that the standard S-RELAP5 [

] Based on this and the assessments presented in the TR, the NRC staff agrees with the physical models used in the AURORA-B EM to predict the Post-CHF Heat Transfer phenomenon. However, as discussed further in Sections 3.6 and 5.1 of this SE, AREVA did not adequately characterize the uncertainty of the post-CHF heat transfer package against test conditions [ ].

3.5.4.5 Peach Bottom Steam Line

These tests address the following highly ranked PIRT phenomenon listed in Table 3.2:

- [ ]

The selected test was a full-scale turbine trip of Peach Bottom Unit 2. The assessment focused on evaluating the code capability to predict [ ], and also addressed other important processes occurring in the steam line, [ ]. AREVA stated that this assessment was used to determine [

]. Measured flow rates were not available for comparison of the TBV performance. However, a calculated estimate of the flow rate through the TBVs was provided as part of an ISP that was based on Peach Bottom Turbine Trip Test 2. The value provided in the ISP was based on RETRAN analyses by the operator of the Peach Bottom plant (Reference 44).

The assessment was performed via measured and predicted pressure versus time at several locations in the steam line at a variety of code time-step sizes. The results show reasonable to excellent agreement; all major trends and pressure spikes in the measured data are captured by the predictions. [

]. This suggests the thermal-hydraulic field equations and numerical solutions can predict pressure wave propagation in steam flow. Therefore, the NRC staff finds the AURORA-B EM can adequately model the Pressure Wave Propagation phenomenon.

3.5.5 Assessment of the System Interaction and Global Capability

The last of the four areas presented for assessment of the AURORA-B EM is system interaction and global capability. This involves demonstrating AURORA-B's ability to predict multiple system interactions and demonstrating AURORA-B's global capability. Within the TR, the assessment of AURORA-B's ability in these areas was presented through comparisons of predictions to tests performed in an electrically heated test facility, through comparisons to three Peach Bottom Turbine Trip transients, and through analyses of several target scenarios for BWR/4 and BWR/6 plants. Each of the system interaction and global capability tests selected also serves to demonstrate AURORA-B's ability to predict one or more highly ranked PIRT phenomena. The results of the assessment tests and the evaluation of the AURORA-B EM are summarized in the following subsections.

3.5.5.1 Full Integral Simulation Test

Test results from the FIST facility serve to help address the capability of the AURORA-B EM (specifically, the S-RELAP5 CCD) in predicting the following highly ranked PIRT phenomena listed in Table 3.2:

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

Previous sections of this SE have already discussed the assessment of the EM's capability in predicting these phenomena, but only under steady state conditions. The FIST tests help assess AURORA-B's ability to predict the interaction of these phenomena during transient events. Additionally, the FIST test results are used within the TR for assessing the global capability of AURORA-B for predicting transient scenarios in which the downcomer water level decreases below typical operating range, and in predicting the dynamic system response during pressurization and depressurization scenarios.

The FIST facility was cosponsored by the NRC and industry, and the intent of its design was for simulating large- and small-break LOCAs and operational transients in jet-pump BWRs. As such, the three different FIST tests simulated with S-RELAP5 and presented in the TR are:

- a quasi-steady natural circulation test (6PNC2),
- a pressurization test (4PTT1) that utilized boundary conditions similar to those in the Peach Bottom Turbine Trip Test 3, and
- a depressurization test (6MSB1) that simulated a main steam line break event.

The nodalization and selected options used in S-RELAP5 for modeling FIST were consistent insofar as possible with the nodalization and model options used in plant analyses (summarized in Section 5.2.8 of the TR and discussed in Section 3.3.4 of this SE). However, revisions to the standard nodalization were necessary to accommodate the narrow geometry of the test facility and non-prototypical configuration of the downcomer and lower plenum regions.

The series of natural circulation tests run at the FIST facility measured the natural circulation core flow rate as a function of power and water level in the downcomer and bypass. The tests were performed under quasi-steady state conditions where power and system pressure were kept nearly constant and feedwater flow adjusted such that the level dropped at a rate of less than one inch per second. The downcomer was at saturated temperature, the recirculation pumps were disabled, and the loop isolation valves were closed to eliminate any forced or secondary circulation paths. The tests started with the water level above the nominal level and ended when it reached the top of the jet pumps. The initial water level in the downcomer was higher than the levels in the core, upper plenum, and standpipes. Thus, the difference in water levels caused a gravity induced (natural circulation) flow from the downcomer to the core. Test 6PNC2-1, which AREVA chose to analyze, possessed a 2.0 MW assembly power and had the slowest rate of level decrease. AREVA's goals with the analysis were to demonstrate the capability of the AURORA-B EM to predict natural circulation and the capability of S-RELAP to predict transient scenarios and system interactions.

In a quasi-steady natural circulation test, the key parameter is the downcomer water level, which drives the natural circulation flow through the core and core bypass. The results for Test 6PNC2-1 presented in the TR compare predicted and measured downcomer flow and core bypass flow as a function of downcomer level. The predictions reasonably follow the behavior of the data trends. The reasonable prediction of measurement data and the interplay between downcomer and bypass demonstrate that losses and interfacial drag phenomena are modeled correctly.

The turbine trip pressurization test 4PTT1 was designed to simulate Peach Bottom Turbine Trip Test 3. To simulate such a scenario within AURORA-B requires appropriately modeling the interactions of a number of BWR control systems and system components. The results compare predicted and measured data for a large number of parameters versus time: steam dome pressure, lower plenum pressure, assembly pressure, orifice flow, TBV flow, jet-pump exit flow, and assembly void at various heights. In all but one case, TBV flow, the predictions show reasonable to excellent agreement with the measurement data, as all the major trends and magnitudes of fluctuations are captured. In the exceptional case of TBV flow, a spike in the measured flow rate is not duplicated by the predicted data. According to AREVA, the measured data in the steam line is not accurate after two-phase flow reaches the line, and the sudden increase seen in the flow rate is a measurement anomaly. AREVA's response appears reasonable; the spike in measurement data appears to be an impulse response that is not typical of actual flow data under the applicable test conditions.

Test 6MSB1 was designed to simulate the response of a BWR/6 to a double-ended break of a steam line upstream of the flow limiter. This test thus serves to stress the EM's ability to model phenomena that occur during a depressurization event and to predict dynamic pressure and void fraction responses within the vessel. The predicted system pressure response for the steam dome and lower plenum show excellent agreement with the measurement data. The predicted break flow rate and jet-pump exit flow rate also show excellent agreement with the measurement data. Predicted downcomer flow rate and assembly void fraction at various heights show reasonable agreement with the measurement data. The void fraction response helps demonstrate the EM's ability to predict **[ ]**, while the pressure results help demonstrate the ability of the EM to predict dynamic pressure responses.

In summary, the assessment results show reasonable agreement between measured and predicted dynamic pressure in different regions of the vessel, void fractions in the simulated core and bypass regions, and incipience of flashing and level swell. The NRC staff therefore finds the results demonstrate that AURORA-B can adequately predict the dynamic system response and the interaction between different components (and the specified highly ranked PIRT phenomena) during pressurization and depressurization events.

3.5.5.2 Peach Bottom Turbine Trip Tests

These tests address the following highly ranked PIRT phenomena listed in Table 3.2:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

The Peach Bottom Turbine Trip Tests are a set of three plant transients that were performed at Peach Bottom Unit 2. The plant transients were turbine trip pressurization events, and their purpose was to expand the experimental database of neutron kinetics and thermal-hydraulics coupled behavior for qualification of BWR transient analysis codes. The three tests, known as Turbine Trip 1 (TT1), Turbine Trip 2 (TT2), and Turbine Trip 3 (TT3), respectively, were performed at reduced power with some control rods positioned within the core to maintain the desired power level. Design and operating characteristics of Peach Bottom Unit 2 at the time of the tests are summarized in Table 6-14 of the TR.

Input parameters for the thermal-hydraulic, thermal-mechanical, and neutron kinetics models were based on the same processes and technical guidance with which models for licensing analysis were developed (see Section 3.3.4 of this SE). Specifically, fuel parameters and neutron kinetics data (e.g., cross section) for the thermal-mechanical and neutron kinetics models are based on RODEX4 and MICROBURN-B2 and not data specified in the ISP. The nodalization description provided in Section 5.2 of the TR is directly applicable to the Peach Bottom plant model. **[**

].

The scram signal used in calculating the assessment results was derived from the Average Power Range Monitors (APRMs) and utilized best estimate values for the reactor protection system timing and the APRM flux setpoint. The assessment results show that the peak power, peak dome pressure, and integral power are all conservatively calculated. There is reasonable to excellent code-data comparisons of the pressure and LPRM responses. From this, the NRC staff concludes the EM makes acceptable predictions of the indicated PIRT phenomena. In addition, the global capability of the EM to predict the events is demonstrated.

3.6 Uncertainty Analysis

Following the review guidance provided in Chapter 15.0.2 of the SRP, the next area of review for transient and accident analysis methods discussed in this SE focuses on uncertainty analysis. The associated acceptance criteria indicate that the analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistently with the results of the accident scenario identification process.

According to Section 6.8 of the TR, there are three factors that contribute to biases and uncertainties of the EM. They are: the EM structure (e.g., as examined via nodalization and time-step size sensitivities), the selection of plant parameters and initial conditions, and biases and uncertainties associated with the modeling of highly ranked PIRT phenomena. The NRC staff agrees with this assessment, but finds that biases and uncertainties associated with the modeling of medium ranked PIRT phenomena should also be considered. Medium ranked phenomena have a moderate influence on the FoMs, and particularly in totality they may have an impact that cannot be neglected.

AREVA addresses the impact of the EM structure by defining nodalization and time-step parameters such that the “user-effect” is minimized and the FoMs experience an insignificant or conservative range of variation. Nodalization and time-step sensitivity studies are discussed in Sections 3.3.4.2 and 3.3.5 of this SE, respectively, and the NRC staff finds them acceptable.

With regard to the impact of plant parameters and initial conditions, AREVA indicated that these will be selected in such a manner as to ensure the predicted FoM is conservative. According to Section 6.8 of the TR, sensitivity analyses are performed to determine which plant parameters are important for a specific application and what value of each parameter is conservative. The selection of parameters to be perturbed in the sensitivity analyses is based on prior experience and engineering judgement. Section 8.0 of the TR provides an overview of the different plant parameters that are perturbed for transient, ATWS, and accident analyses (e.g., reactor protection system setpoints and response times, turbine stop valve closure time, etc.). The range over which the plant parameters are perturbed appears to be informed by plant technical specifications and the plant parameters data document, which is a plant-specific document jointly prepared by AREVA and the utility that defines nominal plant parameter values and their performance (i.e., allowable range of a parameter and measurement/monitoring uncertainty). AREVA expanded on this in the response to RAI-4, which indicates that when performing a sensitivity analysis to determine the value for a plant parameter, there are **[**

] Section 8.0 of the TR also indicates that sensitivity analyses are performed at various statepoints throughout the plant operating conditions envelope (i.e., the spectrum of possible initial operating conditions that is allowed during normal operation).

The NRC staff agrees with the overall approach described by AREVA, as it demonstrates AREVA's due diligence to determine plant parameter values and initial conditions that predict the most conservative FoMs throughout a plant's operating cycle. However, Section 8.0 of the TR does not discuss which plant parameters are used in a sensitivity analysis on an event-specific basis (opting instead to provide an overview) and whether the parameter values will be set high or low with respect to the nominal condition. For example, some transients produce more conservative results when turbine bypass valves respond slowly, while other transients produce more conservative results when turbine bypass valves respond quickly. As a result, while the NRC staff agrees with AREVA's overall approach for addressing uncertainties associated with plant parameters and initial conditions, it is the NRC staff's position that licensees will need to justify the key plant parameters chosen for the sensitivity analyses and the input values ultimately selected for these key plant parameters and initial conditions when adopting TR ANP-10300P. The NRC staff has added this as a limitation and condition in Section 5.1 of this SE

Concerning biases and uncertainties associated with the modeling of medium and highly ranked PIRT phenomena, several portions of the uncertainty analysis are not well described in the TR:

- The classification of the code/model goodness of fit is not well defined.

- The way uncertainty bounds were calculated is not well defined, and thus the TR does not demonstrate that bias and random uncertainties are accounted.

- The TR does not explain how both measurement and experimental uncertainties are considered.

- The NRC staff had concerns regarding the **[** **]**.

- No uncertainties were presented for medium ranked PIRT phenomena.

As a result of these shortcomings of the TR, the NRC staff issued a number of RAI questions. In the responses to these RAI questions, AREVA made several changes to the methodology for performing the uncertainty analyses. These items are listed below along with a reference to the sections where the acceptability of each item is assessed in this report.

- Detailed description of each highly ranked phenomenon was provided along with a description of how each will be used in the uncertainty analysis (Section 3.6.1).

- Detailed description of each medium ranked phenomenon pertinent to potentially limiting non-pressurization transients was provided along with a description of how each will be used in the uncertainty analysis (Section 3.6.2).

- A new non-parametric (Monte-Carlo) ordered statistics uncertainty analysis was proposed for use in applications of the EM (Section 3.6.3)

- The uncertainty distribution for each item that is sampled in the non-parametric uncertainty analysis is given (Section 3.6.4)

- A description of how the non-parametric statistics are applied for licensing analysis was provided, to justify the uncertainty analysis approach for the EM (Section 3.6.5).

It is AREVA's assertion that the combination of the constraints on the EM structure, the conservative application of plant parameters and operating conditions, and the penalization of FoMs through non-parametric statistical analyses to account for uncertainty in predicting medium and highly ranked PIRT phenomena will ensure the reported FoMs are conservative.

Following review of the RAI responses, the NRC staff was still not in agreement with the different methodologies used to calculate uncertainty ranges for those items that are sampled in the non-parametric uncertainty analysis. Therefore, the NRC staff decided to determine if the uncertainty range and distribution for each parameter is acceptable based on the data comparisons provided by AREVA. Following a number of meetings between the NRC and AREVA staffs, acceptable uncertainty ranges and distributions were determined. This is discussed in Section 3.6.4.

3.6.1 Treatment of Uncertainties for Highly Ranked Phenomena

Table 3.2 lists the phenomena selected in the PIRT review that are considered to have a high impact on the FoMs for AURORA-B. This table lists how AREVA will treat the uncertainty associated with each phenomenon in the uncertainty analysis and indicates the assessment data that has been used to validate the modeling of each phenomenon. These data are described in greater detail in Section 3.5. The final column of Table 3.2 provides the NRC staff's assessment of AREVA's treatment of the uncertainty for each highly ranked phenomenon.

As noted in the Table 3.2, many of the highly ranked phenomena **[**

] resulting in conservative predictions. The NRC staff expressed a concern with this approach, in that when a Monte-Carlo analysis is performed on a complex system that results in multiple FoMs, it is difficult to state *a priori* that biasing a particular parameter to a perceived bounding value will always result in a conservative result for multiple FoMs. For example, biasing an input parameter to an upper bound may result in a conservative result for one FoM, whereas a lower bound value for that same input parameter may result in a conservative result for another FoM. In other cases, due to the existence of competing effects, a single, generically bounding value for some input parameters, across all plant types and event sequences within the scope of the TR, may be difficult to determine *a priori*. AREVA maintains that the biased models and parameters **[** **]** will always result in conservative predictions of the selected FoMs. The following table will also provide the NRC staff's assessment of the validity of this claim for each highly ranked phenomenon **[**

].

For this reason, among others, this methodology should not be applied to applications beyond the target scenarios stated in this report (Section 3.1) without further review.

Table 3.2 Treatment of Uncertainties for Highly Ranked Phenomena

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	MICROBURN-B2 Qualification (Section 3.5.2.5) Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	The NRC staff finds the Peach Bottom Turbine Trip tests demonstrate **[ ]** is adequately modeled for pressurization transients; the **[ ]** is addressed in the over-prediction of peak and integrated power. While explicit consideration of the uncertainty in this phenomenon is preferable, several points concerning AREVA's approach are worth noting. First, allowing credit for a demonstrated over-prediction of peak and integrated power is consistent with past review practices. Second, AREVA takes conservative positions in setting initial conditions. Third, the uncertainty of other phenomena **[ ]** is sampled. Lastly, the **[ ]** uncertainty tends to be significant for rapid transient events that are largely pressurization-related, which are conservatively predicted. Therefore, the NRC staff finds **[ ]** is reasonably addressed for both pressurization and non-pressurization transients.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	Rod Bundle Pressure Drop (Section 3.5.4.1) Full Integral Simulation Test (Section 3.5.5.1)	Based on the small effect that ranging channel flow has **[ ]** and the fact that this **[ ]** (see Section 3.3.1.5 of this SE), the NRC staff finds the treatment of **[ ]** to be acceptable.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	MICROBURN-B2 Qualification (Section 3.5.2.5)	For pressurization transients, the [ ] is typically dominated by the peak and integral power during the transient, and this phenomenon is addressed by the over-prediction of peak and integrated power. However, [ ] described in Section 3.6.3 for this event. The acceptability [ ] is discussed in Section 3.6.4

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	Critical Power Tests (Section 3.5.4.4)	Determination of CHF is a function of the specific CPR correlation(s) associated with the specific fuel assembly design(s) in a given core. A CPR correlation is evaluated and licensed separately from its application in the AURORA-B EM, and AREVA must show that it is appropriately incorporated in the EM on a case-by-case basis. Additionally, the NRC staff acknowledges that for transients, steady-state CPR correlations have a demonstrated history of being conservative. Therefore, the NRC staff finds the uncertainty **[ ]** acceptable.
[ ]	**[ ]**	Rod Bundle Void Tests (Section 3.5.2.1) Christensen Void Tests (Section 3.5.2.2) Full Integral Simulation Tests (Section 3.5.5.1)	This parameter is included in **[ ]** is discussed in Section 3.6.4.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Rod Bundle Void Tests (Section 3.5.2.1) Christensen Void Tests (Section 3.5.2.2) Critical Power Tests (Section 3.5.4.4) Full Integral Simulation Tests (Section 3.5.5.1)	Via teleconference on February 16, 2017, and within Reference 5, AREVA indicated that, for pressurization transients, [ ], as corroborated by comparisons to the Peach Bottom Turbine Trip Tests, the NRC staff finds the uncertainty treatment [ ] acceptable for pressurization transients. Regarding non-pressurization transients, the nominal value is used. The NRC staff finds this acceptable in general for slow transients, but observed during the review that some scenarios [ ]. To confirm adequacy of the modeling for this event, AREVA indicated in RAI-49 that the change in hot channel MCPR results from [ ]. Therefore, the NRC staff finds the uncertainty [ ] acceptable.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Critical Power Tests (Section 3.5.4.4)	The standard S-RELAP5 [ ]. This modification tends to conservatively bias predictions of the maximum cladding temperature [ ], as discussed in Section 3.5.4.4. However, AREVA benchmarked its [ ] As a result, the NRC staff imposed Limitation and Condition No. 17 to request further justification from licensees on a plant-specific basis [ ] during a transient or accident.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	MICROBURN-B2 Qualification (Section 3.5.2.5) Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	This phenomenon is not impacted by the addition of neutron kinetics modeling and is unchanged from the previously approved MICROBURN-B2 steady-state neutronics code for BWR application (Reference 10). The key parameters [ ]. These parameters are plant-specific parameters. Therefore, the NRC staff concludes that the treatment of this parameter must be reviewed on a plant-specific basis, as described in Section 5.1 of this SE.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[**]**	**[** **]**	Rod Bundle Void Tests (Section 3.5.2.1) Christensen Void Tests (Section 3.5.2.2)	Although the NRC staff does not fully agree with AREVA's argument that the uncertainty associated with **[** **]**, there appears to be sufficient conservatism in the methodology when applied to pressurization transients. Therefore, the NRC staff finds the uncertainty **[** **]** acceptable for pressurization transients. For non-pressurization transients, this phenomenon is **[** **]** described in Section 3.6.3. The acceptability **[** **]** is discussed in Section 3.6.4

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	MICROBURN-B2 Qualification (Section 3.5.2.5) Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	These parameters are included [ ] analysis described in Section 3.6.3. The acceptability [ ] is discussed in Section 3.6.4.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Full Integral Simulation Tests (Section 3.5.5.1)	Sensitivity analyses discussed in the responses to ARAIs 22, 44, and 49 demonstrate that the AOOs discussed in Section 3.1 of this SE are insensitive to [ ] For licensing analyses, the MICROBURN-B2 explicit water rod/channel model will be used. [ ] Because the void fraction applied to the EM model will be no less than the core simulator, as well as the relative insensitivity to [ ], the NRC staff finds the uncertainty treatment of [ ] acceptable.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[**]**	**[** **]**	GE Level Swell (Section 3.5.2.4)	Sensitivity of the EM to bias or uncertainty in **[** **]**. The NRC staff finds this acceptable because **[** **]** is not a phenomenon anticipated to be of great concern during a pressurization event. For depressurization events, as well as events that have characteristics of both depressurization and pressurization (e.g., PRFO), **[** **]** described in Section 3.6.3. The acceptability **[** **]** is discussed in Section 3.6.4.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	No data provided in TR	As discussed in Section 3.5.2.7 of this SE, AREVA's nodalization [ ], no basis or methodology is provided for how the associated uncertainty will be determined. Therefore, as described in Section 5.1 of this SE, changes to the evaluation model [ ] a plant-specific basis without prior NRC review are not approved by this SE.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	RODEX4 Qualification (Section 3.5.2.6) Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	These parameters [ ] described in Section 3.6.3. The acceptability [ ] is discussed in Section 3.6.4.
[ ]	Sampled as "*dopmul*" in non-parametric analysis.	MICROBURN-B2 Qualification (Section 3.5.2.5) Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	This parameter [ ] described in Section 3.6.3. The acceptability [ ] is discussed in Section 3.6.4.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	**Supporting Assessment Data**	**NRC Staff Assessment**
[ ]	**[ ]**	MICROBURN-B2 Qualification (Section 3.5.2.5) Critical Power Tests (Section 3.5.4.4)	Based on the tendency for **[ ]**, the NRC staff finds the uncertainty treatment **[ ]** acceptable.
[ ]	**[ ]**	Jet Pump Performance Tests (Section 3.5.4.2) Full Integral Simulation Tests (Section 3.5.5.1)	This parameter is included **[ ]** described in Section 3.6.3. The acceptability **[ ]** is discussed in Section 3.6.4.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	No data provided in TR	As discussed in Section 3.5.2.7 of this SE, AREVA's nodalization **[ ]**, no basis or methodology is provided for how the associated uncertainty will be determined. Therefore, as described in Section 5.1 of this SE, changes to the evaluation model **[ ]** on a plant-specific basis without prior NRC review are not approved by this SE.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	Full Integral Simulation Tests (Section 3.5.5.1)	Based on the use of plant-specific data for **[ ]** and uncertainty, the NRC staff concludes that the treatment of this parameter will have to be reviewed on a plant-specific basis.
[ ]	**[ ]**	Full Integral Simulation Tests (Section 3.5.5.1)	Based on using plant-specific data for **[ ]** and uncertainty, the NRC staff concludes that the treatment of this parameter will have to be reviewed on a plant-specific basis.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	Steam Separator Tests (Section 3.5.4.3)	This parameter is included in the **[ ]** analysis described in Section 3.6.3. The acceptability **[ ]** is discussed in Section 3.6.4.
[ ]	**]**	Steam Separator Tests (Section 3.5.4.3) Full Integral Simulation Tests (Section 3.5.5.1)	This parameter is included in the **[ ]** analysis described in Section 3.6.3. The acceptability **[ ]** is discussed in Section 3.6.4.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Allis-Chalmers Large Diameter Void Tests (Section 3.5.2.3)	Imposing homogeneous flow [ ] causes an artificially high interfacial drag effect that impacts [ ]. Because the [ ] was assessed against data (Section 3.5.4.3) when using homogeneous flow at the model inlet, the uncertainty [ ] is due in part to the high interfacial drag. Therefore, the explicit sampling of [ ] over an appropriate range (as discussed in Section 3.6.4.6 of this SE) accounts for the interfacial drag uncertainty. With regard to [ ], normal BWR operation results in a mass flux from ~$1.8\text{-}2.2\times10^{6}$ lbm/hr-ft^{2}. For this mass flux range, Reference 45 indicates that the flow is sufficiently homogeneous for large pipes to allow for a homogeneous approximation with a slight over-prediction of pressure drop. Based on the information from Reference 45, [ ], the Peach Bottom Turbine Trip Test assessments, and the ranges of mass flux and void fraction evaluated, the NRC staff concludes the uncertainty treatment [ ] is acceptable.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	AREVA's response to RAI-49a indicates these parameters are part of category known as "plant parameters and initial conditions." The parameters in this category are plant-specific in nature and their range of applicability for licensing analyses is documented by the utility. The uncertainties of these parameters will be used to bias the parameters in a manner that ensures a conservative prediction of the FoMs. Because of this, the NRC staff concludes that the treatment of this parameter will have to be reviewed on a plant-specific basis.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	**[ ]**	Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	Demonstrations of the nodalization and time-step size variations were provided by AREVA in response to ARAI-24. These demonstrations were used to define the range of node lengths and time step sizes that resulted in conservative results for Peach Bottom **[ ]**. The NRC staff also concluded that a limitation and condition should be imposed in Section 5.1 of this SE stipulating that the steam dryers should be modeled **[ ]** Since the identified modeling practices will be consistently used, the NRC staff finds the uncertainty treatment **[ ]** acceptable.

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	Supporting Assessment Data	NRC Staff Assessment
[ ]	[ ]	Allis-Chalmers Large Diameter Void Tests (Section 3.5.2.3)	The response AREVA provided in Tables 20-1 through 20-6 of ARAI-20 and the response to ARAI-25 demonstrate no impact on the majority of transients when [ ] The sensitivity study showed insignificant changes in the FoM for limiting MCPR events (e.g., turbine trip no bypass, feedwater controller failure, and pressure regulator failure open). For those transients that show some impact, there is a small reduction in peak pressure and integral power, which may slightly reduce margin in some cases. Overall, the assumption [ ] is largely conservative. Therefore, based on the conservatism of the [ ] assumption, the Peach Bottom Turbine trip assessments, and the sensitivity results, the NRC staff finds the uncertainty treatment [ ] reasonable.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	**Supporting Assessment Data**	**NRC Staff Assessment**
[**]**	**[** **]**	Peach Bottom Turbine Trip Tests (Section 3.5.5.2)	Based on the minimal impact **[** **]**, the NRC staff finds the uncertainty treatment of **[** **]** acceptable.

3.6.2 Treatment of Uncertainties for Medium Ranked Phenomena

The original uncertainty analysis presented in TR ANP-10300P did not sample medium-ranked PIRT phenomena. In response to NRC staff concerns (as discussed in Section 3.6), AREVA updated the uncertainty analysis and supplied detailed descriptions of each medium ranked phenomenon pertinent to potentially limiting non-pressurization transients and how it will be used in the uncertainty analysis. By definition, medium-ranked PIRT phenomena have a moderate impact on FoMs (i.e., non-negligible, particularly in totality). It has further been the NRC staff's general experience that, when analyzed in a quantitative manner, one or more medium-ranked phenomena may often rank among the most significant influences on a given event in plant-specific statistical calculations (e.g., due to plant- or code-specific variation relative to demonstration cases or past experience, error in subjective judgements during PIRT development, etc.). As a result, the NRC staff reviewed the adequacy of AREVA's proposed treatment of medium-ranked phenomena.

During the NRC staff's review, it was concluded that the treatment of medium-ranked phenomena is adequately addressed for pressurization transients in light of **[ ]** (Section 3.5.5.2 of this SE) and the conservative adjustments to the calculation of **[ ]** in Table 3.2, above). However, for non-pressurization events in general, these compensating factors either do not apply or do not appear to have a significant conservative impact. As such, the NRC staff's review did not find an adequate basis for AREVA's treatment of medium-ranked phenomena for these events. The NRC staff expressed this observation to AREVA via teleconference on January 24, 2017. During a follow-up teleconference on February 16, 2017, AREVA presented the non-pressurization events within the scope of TR ANP-10300P, along with a generic assessment of their expected impact on setting plant operating limits. These events and assessments are shown below in Table 3.3:

Table 3.3 Non-Pressurization Events and Generic Assessment of Impact

SRP Section	Representative Event	AREVA's Assessment
15.1.1-15.1.4	Loss of Feedwater Heater	**[ ]**
15.1.1-15.1.4	Inadvertent RHR Shutdown Cooling	Benign event
15.2.7	Loss of Normal Feedwater Flow	**[ ]**
15.2.7	Loss of RHR Shutdown Cooling	Benign event
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow	**[ ]**
15.3.1-15.3.2	Recirculation Flow Controller Malfunction (Decreasing Flow)	Benign event
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure	**[ ]**

SRP Section	Representative Event	AREVA's Assessment
15.3.3-15.3.4	Reactor Coolant Pump Shaft Break	[ ]
15.4.4-15.4.5	Startup of Idle Recirculation Pump	[ ]
15.4.4-15.4.5	Recirculation Flow Controller Malfunction (Increasing Flow)	[ ]
15.5.1-15.5.2	Inadvertent Operation of ECCS	[ ]
15.6.1	Inadvertent Operation of Pressure Relief Valve	Benign event

For the non-pressurization events listed above in Table 3.3 that may be limiting, AREVA identified that a combination of highly-ranked and medium-ranked phenomena play a role in the event evolution. The list of non-pressurization events, AREVA's generic assessment of event impact on setting operating limits, the identified medium-ranked phenomena, and AREVA's treatment of the associated uncertainty were provided to the NRC staff in the revised RAI-49 response (Reference 52).

Table 3.4 lists the phenomena that are considered to have a medium impact on the FoMs for limiting non-pressurization transients. This table lists how AREVA will treat the uncertainty associated with each of the identified medium-ranked phenomenon and provides the NRC staff's assessment of AREVA's treatment.

The NRC staff examined the list of non-pressurization events and their impacts on plant operating limits and agreed with AREVA's assessments. The NRC staff also examined the non-pressurization events in light of AREVA's uncertainty treatment of both the identified medium-ranked phenomena and highly-ranked phenomena. The NRC staff concluded that a combination of the highly-ranked and medium-ranked PIRT phenomena selected for inclusion in the uncertainty analysis (as listed in Table 3.2 and Table 3.4 of this SE) adequately cover the identified non-pressurization events with three exceptions:

- **[ ]**
- **[ ]**, and
- **[ ]**

The NRC staff's review concluded that the AURORA-B EM contains code models and correlations sufficient for simulating all these events. However, **[**

], and the NRC staff therefore did not assess the adequacy of the uncertainty methodology for these events. Additionally, as discussed below in Table 3.4 [ ], the NRC staff found that, while [ ], AREVA did not supply sufficient justification for the application of the uncertainty analysis to this event. Specifically, AREVA did not propose a method for modeling the uncertainty for [ ]. Therefore, the NRC staff imposed Limitation and Condition No. 13 in Section 5.1 of this SE, wherein licensing applications may rely on nominal calculations with the AURORA-B EM for these events in the course of demonstrating that all regulatory limits are satisfied with significant margin, but the existing uncertainty methodology may not be applied directly to these specific events.

Of the non-pressurization events listed in Table 3.3, of specific note is [ ]. This event is addressed by the restriction on [ ] (Section 3.5.2.7 of this SE). Inadvertent HPCI actuation may set MCPR and/or LHGR limits, depending on the HPCI pump flow capacity, assumed injection temperature, and the initial core power level. However, [ ]. Therefore, the NRC staff concluded that AREVA's proposed method [ ] is a conservative approach for the [ ] event.

Table 3.4 Treatment of Uncertainties for Pertinent Medium Ranked Phenomena

Phenomenon	Treatment (Section numbers and references refer to the TR and the RAI responses)	NRC Staff Assessment
[ ]	**[ ]**	**[ ]** Therefore, the NRC staff finds the approach of sampling the uncertainty **[ ]** through these parameters to be acceptable. The acceptability of the **[ ]** analysis are discussed in Section 3.6.4
[ ] **[ ]**	**[ ]**	The uncertainty in these phenomena is addressed by incorporating an uncertainty in the modeling **[ ]** described in Section 3.6.3. The acceptability of the **[ ]** analysis is discussed in Section 3.6.4.

Phenomenon	**Treatment** (Section numbers and references refer to the TR and the RAI responses)	NRC Staff Assessment
[ ]	**[ ]**	AREVA indicated that the **[ ]** event has **]**. While NRC staff expects that this may be the case, the NRC staff has concerns that the uncertainty in one of the dominant physical processes for this event is not captured in the AURORA-B methodology. Additionally, AREVA did not supply any analyses to justify the position that the **]** event has **[ ]** impact on operating limits. Therefore, the NRC staff imposed Limitation and Condition No. 13 in Section 5.1 of this SE, wherein licensing applications may rely on nominal calculations with the AURORA-B EM for the **[ ]** event in the course of demonstrating that all regulatory limits are satisfied with significant margin, but the existing uncertainty methodology may not be applied directly.

3.6.3 Non-Parametric Uncertainty Analysis Process

AREVA originally proposed performing a **[** **]** sensitivity study to determine the impact of uncertainties on the FoMs for the AURORA-B EM. As a result of a number of RAI questions requesting expanded discussion on the interrelation of parameters, effects of biases, and the rationale behind the parameter selections for sensitivity analyses, AREVA made the decision to re-evaluate the impact of code uncertainties on FoMs using a Monte-Carlo non-parametric order statistics method that has wide acceptance in the nuclear industry. This process is described in the response to RAI-49 and is summarized in this section.

In this process, a baseline case is developed and then various code and modeling uncertainty parameters are randomly varied **[** **]**. The code and modeling uncertainty parameters are independently and randomly varied within their uncertainty ranges for each of the realizations. The applicability of the uncertainty ranges for each of the code and modeling uncertainty parameters used in the non-parametric approach is discussed in Section 3.6.4. Each realization involves **[** **]** randomly sampled values of the uncertainty parameters as part of the **[** **]**.

The limiting FoM results from each realization are sorted in descending order and, using the theory of non-parametric order statistics, a one-sided upper tolerance limit at the 95 percent probability and 95 percent confidence level (95/95) is determined from one of the sorted realizations. The specific sorted realization used (the "acceptance number") from the ordered set is based on the number of realizations that were run (the "sample size"), as shown in Table 3.5. The limiting FoM result from this sorted realization represents an upper tolerance limit, **[** **]**. As noted above, the tolerance limits proposed by AREVA are one-sided. The NRC staff agrees that one-sided upper tolerance limits are appropriate for demonstrating that a FoM remains below a specified regulatory limit.

Table 3.5 Acceptance Number versus Sample Size for Non-Parametric One-Sided Tolerance Limits at the 95 Percent Probability and 95 Percent Confidence Levels.

Acceptance Number	Sample Size

The NRC staff requested that AREVA provide justification for an assumption that is implicit in this method: that the code and modeling uncertainty parameters are independent from each other. Unless this is true, the non-parametric order statistics method cannot be used. In the responses to RAI-49, AREVA stated that the only obvious dependence between uncertainty parameters is the **[ ]**, and that the revised methodology **[**

]. AREVA's response also states that it is their engineering judgement that the remaining parameters do not display any evidence of direct dependence on common physical phenomena, and therefore, the assumption of independence is justified. The NRC staff finds that AREVA has taken reasonable precautions in choosing parameters for statistical sampling in a manner that minimizes adverse impacts of interdependencies.

The original proposal of the non-parametric uncertainty analysis process included a method of sampling the uncertainty distribution of a parameter such that any value sampled outside a **[**

] The NRC staff expressed concern that this truncation would impact the appropriate calculation of a 95/95 one-sided upper tolerance limit; the NRC staff analyses demonstrated that the output of the method is affected by the input sampling range. Based on these concerns, AREVA proposed a change to **[**

] for reasons described in Section 3.6.4.14 of this SE. In addition, **[ ]** (Section 3.6.4.15) and **[ ]** (Section 3.6.4.16) **[ ]**. The NRC staff finds that the truncation at **[ ]** for the input distributions is acceptable, as it falls within the sampling range that was determined by NRC staff analyses to no longer significantly impact the method's output. The NRC staff's evaluations of AREVA's proposal to **[**

] are given in Section 3.6.4.

Overall, the NRC staff finds that the non-parametric uncertainty analysis process as described in the final RAI responses is acceptable.

3.6.4 Uncertainty Parameters for Non-Parametric Uncertainty Analysis

Table 3.6 below lists the uncertainty parameters that will be used for the non-parametric uncertainty analyses performed using AURORA-B for the transients and accident scenarios discussed in this SE. Except where noted, these uncertainty values will be used for all analyses using the approved methodology. Where exceptions are given, the methodology for how an acceptable uncertainty will be derived is discussed in the sections to follow. [

] is discussed in Section 3.6.3.

Table 3.6 Uncertainty Parameters for Non-Parametric Uncertainty Analysis with AURORA-B for Transient and Accident Scenarios

The following sections discuss the applicability of the ranges and distributions proposed by AREVA for each of the uncertainty parameters used in the non-parametric uncertainty analysis shown in Table 3.6.

Overall, the NRC staff finds the ranges and distributions proposed by AREVA for each of the uncertainty parameters used in the non-parametric uncertainty analysis to be acceptable.

3.6.4.1 [ ]

In the response to RAI-49b, AREVA presented results from a **[** **]** sensitivity study. As a starting point, AREVA states that **[** **]**. AREVA's justification for this is that small changes in **[** **]** While this is a simplification of the complex phenomena that interact to create the **[** **]** effect, the NRC staff finds it is a reasonable approximation of the effect for small changes **[** **]**.

To **[** **]** and observe the resulting change in power distribution, AREVA adjusted the void fraction determined by the **[** **]** correlation in MICROBURN-B2 by adding or subtracting the difference in void fraction from two **[** **]**. The two separate values of **[** **]** were needed to produce a **[** **]** in predicted void fraction, resulting in **[** **]**. The ±0.05 change was somewhat arbitrarily defined; the basis for it is that it bounds the ATRIUM-10 void fraction test data. AREVA indicates that when performing the sensitivity analyses, these **[** **]** were imposed over all fuel in the core from BOL. Therefore, the impact on the fuel as it depleted was captured, and the difference in depletion changes the sensitivity of void fraction modifications considerably. This will exaggerate the results of the sensitivity studies, making them more conservative.

To observe the effects the change in void fraction **[** **]** had on power distribution, AREVA compared the calculated MICROBURN-B2 **[** **]**

As a final point, AREVA presents results of the EPICURE critical experiments wherein the moderator density reactivity was evaluated. According to AREVA, the results show reactivity worth changes consistent with a **[** **]** reactivity uncertainty.

The NRC staff observed that **[** **]** was not entirely consistent with the EPICURE results. **[**

]. The NRC staff concluded that [

] would not be conservative for depressurization events. The NRC staff expressed this observation to AREVA via teleconference on January 24, 2017, and AREVA agreed to examine the data and present an alternative [ ]. AREVA submitted this information as part of the revised RAI responses (Reference 52). For non-pressurization events, a [
]. The NRC staff examined the EPICURE data with this uncertainty range and found it was sufficiently conservative for depressurization events.

Based on the discussion above, the NRC staff finds the sampling [

] to be acceptable.

3.6.4.2 [ ]

According to AREVA's response to [

]

[
]. While there are different inherent uncertainties between these codes when calculating the [
], the NRC staff does not expect a difference in predicted [

] Therefore, the NRC staff finds the sampling ranges and [
] acceptable.

3.6.4.3 [ ]

In response to RAI-49, AREVA proposed to introduce an uncertainty [
]
based on the literature (Reference 47), and that it will be applied at a core-level. However, AREVA also identified in RAI-49 of the revised RAI responses that variation [
] exists, and that this variation would be applied [
].

The justification [ ] begins with AREVA discussing that variation [ ] is due to the contribution [
]. The contributions from [
]. However, the contributions [
], is

very well-predicted. Therefore, [ ]. The NRC staff finds that, while the justification for not needing to examine the uncertainty [ ] is based largely on qualitative reasoning, the well-predicted isotopic concentration does suggest that the uncertainty [ ] neutron fraction is largely dominated by the uncertainty [ ]. Therefore, the NRC staff finds AREVA's decision [ ] reasonable.

AREVA presented benchmarking results in RAI-49 that compared [

] The NRC staff examined the [ ] data and noted these analyses were performed with [ ], which does not cover a large part of the realistic range of possible histories. Additionally, the NRC staff examined the [ ] data and noted that this [ ] is not representative of [ ]. Therefore, the NRC staff agrees with the degree of conservatism that AREVA has elected to use [ ].

Having assessed the magnitude of [

]:

[

[ ]

where

[

]

]

[

]

Since the relative contribution of **[**

] AREVA presented a table of the results in Table 49-9 of its response to RAI-49 and indicated that **[**

]

AREVA's use of **[**

] The NRC staff finds this provides a resolution sufficient for acceptably accurate **[** **]**. Based on this and the discussion above, the NRC staff finds the determination **[**

], to be acceptable. Based on a review of the literature, the NRC staff finds the use of a **[** **]** to also be acceptable.

3.6.4.4 [ ]

In response to RAI-49, AREVA indicated that **[** **]** will be sampled **[**
]. AREVA also discussed that a **[**
] determined from published uncertainties and **[**
].

[**]**, the analysis can continue over an extended period after the reactor has tripped, in which case **[** **]**. AREVA's choice **[**

] uncertainties. For additional conservatism, AREVA confirmed via teleconference with the NRC that an **[**

], and the NRC staff therefore agrees this adds additional conservatism. Based on this discussion, the NRC staff finds the use of a **[**
] to be acceptable **[** **]**.

3.6.4.5 [ ]

In response to RAI-49b, AREVA indicated that the uncertainty **[** **]** is sampled for non-pressurization transients, as well as other events, such as **[**
]. To facilitate sampling **[**

]

[

]

The NRC staff agrees that varying [

], the NRC staff finds it does create a relationship that allows for approximately [

] Therefore, the NRC staff finds the upper and lower limits [ ] to be acceptable. The NRC staff further agrees that, as long as the [ ].

The NRC staff notes that the [ ]. AREVA has indicated in Table 49-5 of RAI-49 that the uncertainties associated with this event, as well as other events [ ]. The NRC staff finds this acceptable.

3.6.4.6 [ ]

Based on the NRC staff analyses of the [ ] (discussed in Section 3.5.4.3), the NRC staff finds [ ] acceptable.

3.6.4.7 [ ]

In the response to RAI-59, AREVA provided results for [ ] pressure drop predictions. These results demonstrated that the measurement data could be bounded **[ ]**. This is not dissimilar from applying prediction error bounds, as discussed for the steam separator carryunder results (Section 3.5.4.3). Based on these results, the NRC staff finds the sampling ranges and **[ ]** acceptable.

3.6.4.8 [ ]

AREVA's response to RAI-57 indicated that **[ ]** similar to that which NRC staff performed in assessing the steam separator carryunder results (discussed in Section 3.5.4.3). These results demonstrated a baseline mean, ±2σ bounds, and distribution for the prediction error. The uncertainty **[ ]** was then determined by multiplying the nominal coefficient **[ ]**, the NRC staff finds this approach **[ ]** acceptable.

3.6.4.9 [ ]

[ ] uncertainty (see the above section), AREVA's response to RAI-57 also included a prediction error analysis for **[ ]**, there was no need to develop a scaling factor. Therefore, **[ ]**. Based on the results **[ ]**, the NRC staff finds the sampling range and **[ ]** acceptable.

3.6.4.10 [ ]

AREVA indicated in the response to RAI-49b that uncertainty **[ ]**. Therefore, AREVA proposed a methodology by which the uncertainty **[ ]** is determined. The methodology is **[ ]** AREVA provided an example of the methodology **[ ]**.

The first set **[** **]** sampled the full range **[** **]** normally used for these analyses **[** **]**. This set of sampling parameters includes an added **[** **]**

The second set **[** **]** sampled only **[** **]**:

[**]**

[**]**:

[**]**

[**]**

As this approach is consistent with general practice, the NRC staff finds this approach **[** **]** acceptable for **[** **]**. However, it is the NRC staff's assessment that **[** **]**. As a result, the NRC staff imposed Limitation and Condition No. 6, from which it follows that **[** **]** Additionally, manufacturing uncertainties and fuel data for non-AREVA fuel may not always be available. Therefore, when non-AREVA fuel is present in a core design, **[** **]**.

As mentioned above, the uncertainty **[** **]** and therefore must be **[** **]** and conservatively determined **[** **]**. This has been captured in Limitation and Condition No. 3 in Section 5.1 of this SE.

3.6.4.11 [ ]

Following a similar approach to that used for **[ ]** (Section 3.6.4.10) **[ ]**. The NRC staff expressed concern that this uncertainty could be too small to address possible biases between **[ ]**. In response, AREVA indicated in RAI-49b that a larger uncertainty value was **[ ]** (as given in Table 5.5 of Reference 12) **[ ]** of the RODEX4 models to an international database. When adapting it to the average rod methodology used by AURORA-B, AREVA noted the following:

- **[ ]**
- **[ ]**
- **[ ]**

AREVA's implication is that these considerations will **[ ]** While the NRC staff agrees that certain aspects of AREVA's argument will]. In spite of this, the NRC staff ultimately finds this value and **[ ]** to be acceptable for calculating **[ ]** for possible biases between AREVA fuel and **[**

]

3.6.4.12 [ ]

AREVA clarified in RAI-49b that **[**

] Therefore, this value will be used **[**
]. As this value is associated with the **[**
], the NRC staff finds the value and the **[** **]** acceptable for calculating **[** **]**. The NRC staff also notes that, **[**
]. It is therefore conservative to use in applications calculating **[** **]** and acceptable for use in applications where the uncertainty **[**
].

3.6.4.13 [ ]

AREVA clarified in RAI-49b that the **[**
] as discussed above in Section 3.6.4.10. **[**

]:

[ ]

[
]. The NRC staff finds this value and **[** **]** acceptable **[**
]. However, it is the NRC staff's assessment that **[**
] and should therefore not be determined and **[**
]. This is captured in Limitation and Condition No. 6. Therefore, the uncertainty as presented **[**

]

3.6.4.14 [ ]

AREVA clarified in RAI-49b that the uncertainty **[**

]

Based on the above discussion and review [ ], the NRC staff finds the [ ] range of [ ] to be acceptable.

As described and evaluated above in Section 3.3.3.2, AREVA's response to RAI-16 indicates [ ] The NRC staff finds this approach acceptable.

3.6.4.15 [ ]

In response to RAI-49b, AREVA indicated that [ ]

AREVA provided a description in RAI-49b [ ] (Section 3.3.1.2).

[ ]; NRC staff performed a linear regression on the data with an r^2 value [ ] Therefore, the NRC staff finds these values and [ ] to be acceptable. However, the NRC staff notes that the choice in [ ]. As such, AREVA should ensure these values bound the uncertainty associated with [ ].

AREVA also indicated in response to RAI-49 that the scope of application [

] The NRC staff finds this response acceptable.

3.6.4.16 [ ]

In response to RAI-49b, AREVA indicated that **[**

]

[

]

[

] The NRC staff finds the choice to bound data **[**
] to be reasonable, **[**
]. Therefore, the NRC staff finds these upper and **[**
] to be acceptable.

AREVA also indicated in response to RAI-49 that the scope **[**

] The NRC staff finds this response acceptable.

3.6.4.17 [ ]

In response to RAI-49, AREVA indicated that uncertainty in **[**

]

AREVA stated in response to RAI-49 that the uncertainties in **[**

]

While the NRC staff finds the use of **[**

]

Based on the above discussion, the NRC staff finds the **[ ]** to be adequate for determining the uncertainty associated with **[**

] This has been captured in Limitation and Condition No. 3 in Section 5.1 of this SE. Additionally, manufacturing uncertainties and fuel data for non-AREVA fuel may not always be available. Therefore, **[**

].

AREVA also indicated in response to RAI-49 that the scope of application **[**

] The NRC staff finds this response acceptable.

3.6.5 Application of Non-Parametric Uncertainty Analysis for Licensing Analyses

The non-parametric uncertainty analysis process discussed in Section 3.6.3 will be used to determine conservative values for the relevant FoMs in plant-specific transient simulations. First-time applications will need to establish that, for the potentially limiting events, the variations in uncertainties are quantified and the appropriate level of conservatism is applied to ensure conservative operating limits. The NRC staff expects that a LAR will be necessary to permit a licensee's initial application of this methodology, and intends to review on a plant-specific basis whether the level of conservatism used is appropriate.

AREVA's overall framework for applying AURORA-B in plant-specific licensing applications involves several steps. **[**

] AREVA has opted for this approach in order to assure the propagation of biases and uncertainties in PIRT phenomena are suitably accounted for in each transient simulation. **[**

], the NRC staff imposed Limitation and Condition No. 18, which stipulates that a description and justification for the conservative measures used will be supplied on a plant-specific basis.

[

]

Once conservative measures to account for calculation biases and uncertainties in the application of the AURORA-B EM and ensure conservative operating limits are established for a reactor, **[**

]:

- **[ ]**
- **[ ]**
- **[ ]**
- **]**
- **[ ]**
- **[ ]**

[ ]. Furthermore, if the 95/95 FoM for a given parameter calculated according to the defined conservative measures shows substantial variation from the corresponding value calculated in the most recent full statistical analysis, it will be necessary for AREVA to re-perform the full statistical analysis for the affected scenario and determine new conservative measures. In this regard, the NRC staff considers a substantial variation to be a difference in magnitude exceeding the 1σ value associated with the FoM from the most recent statistical analysis. This is reflected in Limitation and Condition No. 18. In implementing such updates to previously approved safety analysis methods, the NRC staff notes that licensees should further consider the discussion in Section 4.0 of this SE and continue to ensure compliance with applicable regulations, such as 10 CFR 50.59.

In reviewing the non-parametric statistical uncertainty methodology proposed by AREVA, the NRC staff observed that the approach would be sufficient to determine 95/95 one-sided upper tolerance limits under the stipulation that a single FoM were to be drawn from a given set of simulations. If, on the other hand, multiple FoMs were to be drawn from a single set of simulations, as is the case in many LOCA methodologies that rely on trivariate non-parametric order statistics to obtain values for peak cladding temperature, local cladding oxidation, and core-wide oxidation, then the sample size for the set of simulations would need to be increased **[ ]** to achieve a 95/95 joint one-sided upper tolerance limit. In light of the five FoMs considered in ANP-10300P (as discussed in Section 3.4.3 of this SE), the NRC staff discussed this point with AREVA during the review **[ ]**.

While AREVA's stated intent is appropriate and consistent **[ ]**, the NRC staff noted that the necessity of performing additional simulations to provide a 95/95 joint one-sided upper tolerance limit for multiple FoMs depends on whether or not the same set of conservative analysis conditions (i.e., initial conditions, boundary conditions, modeling techniques, etc.) is used to obtain the FOMs. For example, if it is posited that selection of different analysis conditions is necessary to determine conservative values for the PCT and peak pressure FoMs for a particular ATWS scenario, then two separate sets of non-parametric simulations should be performed, each with its own appropriate set of conservative analysis conditions, according to **[ ]** sample sizes shown above in Table 3.5. If, on the other hand, it is posited that the turbine trip without bypass event could set limiting values for the MCPR and (AOO) peak pressure FoMs, and that an identical set of conservative analysis conditions is appropriate for determining both FoMs, then a single set of simulations should be performed using an increased sample size appropriate for the bivariate case.

Although methods involving multivariate non-parametric order statistics are well known and have been implemented by all vendors for LOCA analysis (e.g., AREVA's Realistic Large Break LOCA Methodology for PWRs, TR EMF-2103P-A, Revision 3, Reference 48), **[ ]**. Therefore, in light of the foregoing discussion, the NRC staff concludes that the sample sizes proposed in TR ANP-10300P (as reflected above in Table 3.5) are acceptable **[ ]**; however, should plant-specific implementations require **[ ]**, then additional plant-specific justification will be necessary to demonstrate that a **[ ]** one-sided upper tolerance limit for all potentially limiting FoMs. This is Limitation and Condition No 18 in Section 5.1 of this SE.

The NRC staff further noted that ANP-10300P does not explicitly include the following additional restrictions on the calculational process that past reviews (e.g., EMF-2103P-A, Revision 3) have found necessary to ensure that the predicted results are statistically representative:

- While AREVA may choose the sample size (or acceptance number), as per Table 3.5 **[ ]**, this choice must be made before initiating the statistical simulations that are intended to support a given plant licensing application. As discussed in Chapter 24 of NUREG-1475, Revision 1 (Reference 49), applying *a posteriori* knowledge of the statistical results to modify the initial choice of sample size would reduce the confidence level associated with the calculated tolerance limits; hence, the NRC staff does not find such practice acceptable.

- **[ ]**

- **[ ]**

In light of the discussion above, the NRC staff determined that additional restrictions are required to ensure that the FoMs calculated by AREVA in accordance with ANP-10300P would satisfy the 95/95 criterion. Therefore, the NRC staff added Limitation and Condition No. 19 in Section 5.1 of this SE to ensure that (1) AREVA will use **[ ]** statistical calculations, (2) AREVA will choose the sample size prior to initiating statistical calculations, (3) AREVA will not arbitrarily discard undesirable statistical results, and (4) AREVA will maintain an auditable record to demonstrate that its process for performing licensing calculations has been performed in a statistically representative manner.

3.7 Quality Assurance

Following the review guidance provided in Chapter 15.0.2 of the SRP, the final area of review for transient and accident analysis methods focuses on a quality assurance program. The associated acceptance criteria indicate the code must be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

AREVA has an established NRC-approved software quality assurance program, and the AURORA-B TR discusses both the development and maintenance of the EM under this program. Therefore, the NRC staff finds the AURORA-B EM is adequately maintained under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B.

4.0 UPDATES AND CHANGE PROCESS

In Section 9.0 of the TR, AREVA presents a change process by which the AURORA-B EM is intended to be modified. It is stated that the changes described in the process are important to maintaining modern and robust computer codes, and that without the flexibility offered by the change process, the EM will not keep pace with subsequent updates and improvements that may arise as new data or expanded assessments become available. Essentially, the purpose of the process is to maintain the AURORA-B EM as state-of-the-art. However, many of the desired changes and modifications discussed in the change process are intended to be implemented without prior NRC review and approval. While the NRC staff understands the desire to maintain a code as state-of-the-art (and openly acknowledges the benefits this can reap), the NRC staff must balance this with maintaining enough regulatory oversight to ensure that transient and accident analysis codes provide a realistic or conservative result and that they adhere to all applicable regulatory requirements. As such, the NRC staff reviewed AREVA's proposed change process for the AURORA-B EM and assessed the impact such changes might have on the EF's regulatory compliance.

The change process encompasses three areas:

(1) Use of new (approved and unapproved) models, correlations, or CCDs within the EM,

(2) Changes in EM to support changes in plant equipment and operating strategies, and

(3) Code modifications to revise existing models, correlations, or CCDs within the EM.

The first of these areas discusses changes with respect to the application and use of new CCDs, CPR correlations, new or revised fuel designs, and use of unapproved methodologies within the AURORA-B EM. As discussed throughout this SE, the AURORA-B EM is comprised of several CCDs that each performs the role of modeling a major phenomenological area, such as thermal-hydraulics, fuel rod thermal-mechanical performance, etc. AREVA states that each of these phenomenological areas will be performed with an approved methodology, and that the applicability of a new methodology to perform such a role within the AURORA-B EM will be demonstrated in an approved document. The NRC staff interprets this to mean that any new methodology that will be incorporated into the AURORA-B EM to either expand its capability or to replace an existing CCD will occur as a result of NRC review and approval of the AURORA-B

EM with the new methodology incorporated into it. An existing NRC approved methodology will not be implemented within the AURORA-B EM without NRC review of the updated EM.

The first change process area also discusses the addition of new CPR correlations within the AURORA-B EM. As with the CCDs, AREVA states that the applicability of the correlations to transient simulations will be demonstrated in an approved document. When CPR correlations are reviewed and approved by the NRC, they are tied to the methodologies employed to demonstrate their qualification. As such, the NRC interprets AREVA's intent of "demonstrating in an approved document" to mean the NRC review and approval of new CPR correlations whose qualifications are presented via analyses performed using the approved version of the AURORA-B EM. A CPR correlation presented to NRC for review and approval whose qualification is provided via a different methodology would not be directly applicable for incorporation into the AURORA-B EM without additional review of supporting qualifications. Specifically, as discussed in Section 3.3.1.5 and Section 3.5.4.4, existing and new CPR correlations that would be used with the AURORA-B EM require confirmation that they too will provide conservative results commensurate with the CPR analyses presented in the TR. This requirement is expressed in Limitation and Condition No. 22 in Section 5.1 of this SE.

With regard to new or revised fuel designs, the first change process area states that the licensing of these designs will address the impact of the new or revised design on transient behavior. Specifically, licensing evaluations performed for the fuel shall ensure that the AURORA-B EM and its constituent CCDs will continue to be used within the limitations and conditions of their approval, and within their verification, validation, and assessment bases. Because this approach ensures the EM and the CCDs will be utilized within their approved scopes, the NRC staff finds this approach acceptable. However, this acceptance is conditional; new or revised fuel designs must exhibit substantially similar thermal and thermal-hydraulic phenomenological behaviors as those fuel designs already approved for use in the AURORA-B EM. New fuel designs exhibiting a large deviation from the established behaviors will require NRC review and approval prior to their implementation in AURORA-B, as per Limitation and Condition No. 23 in Section 5.1 of this SE.

Clauses discussing the use of unapproved methodologies for lead assemblies are also provided in the first change process area. With respect to these, AREVA indicates the use of unapproved CCDs or correlations supporting lead assemblies within the AURORA-B EM will not occur without satisfying the requirements of 10 CFR 50.59, informing the NRC of their use through a LAR (10 CFR 50.92), or other appropriate means. The NRC staff finds this approach acceptable.

The second of the areas discussed in the change process concerns proposed plant modifications that introduce new features, hardware, and/or enhancements. In ARAI-34, the NRC staff identified that this area (Section 9.2 of the TR) was not sufficiently detailed for the NRC staff to review. AREVA responded that the impact of introducing new features, components, operating strategies, or enhancements for supported plants will be reviewed to assess if the proposed change(s) can be adequately simulated by the approved version of AURORA-B (e.g., existing phenomenological models are sufficient, new plant configuration can be captured through updated analysis inputs and initial conditions, etc.). Additionally, review of the new features will ensure their characteristics (1) do not adversely impact the bias, uncertainties, and sensitivities of highly ranked or pertinent medium-ranked phenomena; (2) are bounded by analysis parameters, and (3) are within the limitations and conditions identified within the AURORA-B SE and the SEs of its CCDs. As this would ensure that all existing models are used within their reviewed scope, the NRC staff finds this acceptable. However, if the review determines new modeling is needed, then the results of a proposed plant modification review will ultimately be provided to affected plant licensees as part of the process for making code and methodology modifications, and code modification is an area of the change process that NRC staff has concerns with, as discussed below.

The third and final area of the change process concerns code modifications to revise existing analysis models, correlations, and CCDs used within the AURORA-B EM during the course of its lifecycle. **[**

]:

- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

[

]

In ARAI-36, the NRC staff noted that the changes these potential modification areas cover are referred to as "discretionary modifications." This language appears inconsistent with the provisions of 10 CFR 50.59, and the changes covered by the modification areas appear to be in disagreement with the RG 1.203 "frozen code" concept. AREVA responded that evaluating code and methodology modifications within the framework of 10 CFR 50.59 is the responsibility of the licensee. Information is provided to the licensee by AREVA when code and methodology modifications are made to assist in the evaluation. As such, Section 9.4 of the TR, which discusses code modifications, provides guidance when making the evaluation. AREVA noted that Section 9.4 of the TR does not address how to determine if a particular modification would fall within the provisions of 10 CFR 50.59; for example, determining if the implementation of a different cladding material type within the plant would constitute such a change. Rather, Section 9.4 of the TR only addresses how to evaluate or apply changes to the AURORA-B EM itself. The NRC staff finds this inadequate.

Concerning making changes without further NRC approval and **[**
] RAI-19 requested additional information for the following:

- The potential for compensating errors or model/methodology changes resulting in “no significant change” in results.

- Individual changes may result in “no significant change” but the cumulative effect of several changes over a period of months/years could be significant. There have been cases in past NRC reviews where each individual model or methodology change was not significant (within the criteria proposed in Section 9.3 of this TR) but the combined changes were significant (as determined by NRC review).

- The potential for adverse interactions or incompatibilities with other models in the code.

- Completeness of verification and validation (V&V) of the new model, and of the EM in general, relative to the results presented in Section 6 of the TR.

- The applicability of new validation data and its impact on the overall uncertainty analysis method reviewed by the NRC staff.

AREVA responded that their Software Quality Assurance Procedures for NRC-approved methods require both the V&V of code modifications and the assessment of the changes on the approved methodology results. This involves running a V&V test suite as well as a series of continuity of assessment (CoA) tests designed to evaluate the impact of the code modifications on the EM performance. While the NRC staff found this response reassuring, further investigation of each of the potential modification areas was warranted.

Of the listed potential modification areas, the NRC staff has no objection to (and therefore finds acceptable) the areas of **[**

]

However, of the potential modification areas listed, the NRC staff objected to the areas of **[**

] This has the potential to introduce instabilities, non-conservativisms, and inaccuracies into the calculation of safety parameters, which is inconsistent with the NRC staff's responsibilities of ensuring that transient and accident codes provide a realistic or conservative result and that they adhere to regulatory requirements. The NRC staff considers AREVA's proposal to make changes in these areas without prior NRC review to be outside the current scope of approval. Therefore, AREVA should continue to use existing regulatory processes when making code modifications in these areas (e.g., 10 CFR 50.59, submittal of TR supplements, etc.).

The exception to this non-acceptability is the parallelization of the individual CCDs **[** **]**. The NRC staff acknowledges there is a difference between changing how an equation is discretized versus parallelization, which is how an existing discretization of an equation is split among processors for the purposes of increasing the speed of its computation. Altering how an equation is discretized changes the numerical methodology of its solution, which could impact accuracy and/or introduce instabilities and non-conservative regions within an equation's application space. In parallelization, the existing elements of a discretized equation are split among existing processors and solved simultaneously. This does not alter the numerical methodology or impact the accuracy of the solution; whether the discretization is solved on a single processor or multiple processors, the results will be the same. This applies to the level of CCDs as well; whether multiple CCDs are sequentially executed on a single processor or each CCD is executed on its own processor, the resulting calculation is the same. The NRC staff therefore finds parallelization of individual CCDs to be acceptable.

On a final note, AREVA's change process makes mention that any changes to the individual CCDs that receive NRC review and approval (e.g., an approved supplement to RODEX4 that describes a hydrogen uptake model) may be incorporated into the AURORA-B EM without prior NRC approval. Essentially, because a foundational component of the AURORA-B EM has been updated and approved, the AURORA-B EM implicitly inherits the update. While the NRC staff does not object to this concept in principle, no discussion of the process by which this would be executed was supplied by AREVA. Therefore, the NRC staff does not find this approach acceptable.

5.0 LIMITATIONS AND CONDITIONS

The NRC staff is limiting its approval to the AURORA-B EM as presented in the TR (Reference 2) and the revised responses provided to ARAI and RAI questions (Reference 52). Approval for use of the AURORA-B EM and update process that has been outlined and reviewed by NRC staff in this SE is contingent upon the satisfaction of the following limitations and conditions:

5.1 Range of Applicability

1. AURORA-B may not be used to perform analyses that result in one or more of its CCDs (S-RELAP5, MB2-K, MICROBURN-B2, RODEX4) operating outside the limits of approval specified in their respective TRs, SEs, and plant-specific LARs. In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2.

2. The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. Because AURORA-B makes use of the Cathcart-Pawel oxidation correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness (See Section 3.4.3.1).

   Should the NRC staff position regarding the appropriate acceptance criterion for the Cathcart-Pawel correlation change, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.

3. Parameter uncertainty distributions and their characterizing upper and lower 2σ levels are presented in Table 3.6 and discussed in Section 3.6 of this SE. The distribution types will not be changed and the characterizing upper and lower 2σ uncertainties will not be reduced without prior NRC approval. In the cases of the parameters **[ ]**, the respective methodologies discussed in Section 3.6.4.10 and Section 3.6.4.17 shall be used when determining the associated upper and lower 2σ levels. The **[ ]** is subject to Limitation and Condition No. 4, below.

4. As discussed in Section 3.3.1.2, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are modeled in licensing analyses using AURORA-B, AREVA must justify that the AURORA-B EM can acceptably predict void fraction results for the new fuel designs within the **[ ]** prediction uncertainty bands. Otherwise, the prediction uncertainty bands should be appropriately expanded, and the **[ ]** should be appropriately updated utilizing the methodology discussed in Section 3.6.4.15 of this SE.

5. As discussed in Section 3.3.2.4.4, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are included in licensing analyses performed using the AURORA-B EF, AREVA must justify that the **[ ]** void-quality correlation within MICROBURN-B2 is valid for the new fuel designs at EPU and EFW conditions.

6. The 2σ ranges **[ ]** until AREVA supplies additional justification (e.g., as part of a first-time application analysis) demonstrating an acceptable alternative for NRC review and approval. For **[ ]** will be utilized when performing licensing analyses to determine peak cladding temperature and maximum local oxidation.

   Should the NRC staff position regarding these uncertainties change as a result of additional justification, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.

7. As discussed in Section 3.6 of this SE, licensees should provide justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. Licensees should further justify that the input values ultimately chosen for these key plant parameters and initial conditions will result in a conservative prediction of FoMs when performing calculations according to the AURORA-B EM described in ANP-10300P.

8. The sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses will be truncated at no less than ±6σ **[ ]**

9. For any highly ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.2 of this SE. For any pertinent medium ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.4 of this SE.

10. The assumptions of **[ ]** will be used in the AURORA-B EM to ensure the uncertainty in

SL03: **[** **]** is conservatively accounted for.

11. AREVA will provide justification for the uncertainties used for the highly ranked plant-specific PIRT parameters C12, R01, R02, and SL02 on a plant-specific basis, as described in Table 3.2 of this SE.

12. When applying the AURORA-B EM to the **[** **]**, any changes to AURORA-B to enhance **[** **]** on a plant-specific basis without prior NRC review and approval are not approved as part of this SE, as described in Table 3.2 of this SE.

13. The AURORA-B uncertainty methodology discussed in Section 3.6 of this SE may be used in licensing applications for the events listed in Section 3.1 of this SE, with the exception of three specific events identified in Section 3.6.2 of this SE: **[** **]**. These events are generally expected to be benign and hence non-limiting. While the NRC staff's review concluded that the AURORA-B EM contains code models and correlations sufficient for simulating these events, the uncertainty methodology developed in the TR did not address certain important phenomena or conditions associated therewith. Therefore, while licensing applications may rely on nominal calculations with the AURORA-B EM for these events in the course of demonstrating that all regulatory limits are satisfied with significant margin, the existing uncertainty methodology may not be applied directly to these specific events.

14. The scope of the NRC staff's approval for AURORA-B does not include the ABWR design.

15. For application to BWR/2s at EPU or EFW conditions, plant-specific justification should be provided for the applicability of AURORA-B, as discussed in Section 3.1 of this SE.

16. **[** **]** is not sampled as part of the methodology, justification should be provided on a plant-specific basis that a conservative flow rate has been assumed **[** **]**.

17. If the AURORA-B EM calculates that the film boiling regime is entered during a transient or accident, AREVA must justify that the uncertainty associated with heat transfer predictions in the film boiling regime is adequately addressed.

18. As discussed in Section 3.6.5 of this SE regarding conservative measures:

    a. Plant-specific licensing applications shall describe and provide justification for the method for determining and applying conservative measures in future deterministic analyses for each FoM (e.g., biasing calculational inputs, post-processing adjustments to calculated nominal results), and

    b. If the 95/95 FoM for a given parameter calculated according to the defined conservative measures during a deterministic analysis shows a difference in magnitude exceeding 1σ from the corresponding value calculated in the most recent baseline full statistical analysis, AREVA must re-perform the full statistical analysis for the affected scenario and determine new conservative measures.

19. As discussed in Section 3.6.5 of this SE, the following stipulations are necessary to ensure that the FoMs calculated by AREVA in accordance with ANP-10300P would satisfy the 95/95 criterion:

    a. AREVA will use multivariate order statistics when multiple FoMs are drawn from a single set of statistical calculations,

    b. AREVA will choose the sample size prior to initiating statistical calculations,

    c. AREVA will not arbitrarily discard undesirable statistical results, and

    d. AREVA will maintain an auditable record to demonstrate that its process for performing statistical licensing calculations has been executed in an unbiased manner.

5.2 Code Updates and Changes

The NRC staff is limiting its approval to the AURORA-B EM as presented in the TR (Reference 2) and the revised responses provided to ARAI and RAI questions (Reference 52). Any changes to the calculational methodology, numerical methods, underlying principles, bases, assumptions, range of applicability, etc., are subject to the following limitations and conditions:

20. The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM.

21. NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval.

22. As discussed in Section 3.3.1.5 and Section 4.0 of this SE, the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EF. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 50). Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology.

23. Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the

AURORA-B EM (i.e., thermal energy is conducted through a cylindrical ceramic fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B.

24. Changes may be made to the AURORA-B EM in the **[**

    ] areas discussed in Section 4.0 of this SE without prior NRC approval.

25. The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of this SE.

26. AREVA must continue to use existing regulatory processes for any code modifications made in the **[**

    ] areas discussed in Section 4.0 of this SE.

6.0 CONCLUSIONS

The NRC staff reviewed the AURORA-B EM described in TR ANP-10300P against the six review areas and associated acceptance criteria specified in Section II of SRP 15.0.2 for transient and accident analysis methods, as discussed in Section 2.0 of this SE:

- Documentation
- Evaluation Model
- Accident Scenario Identification Process
- Code Assessment
- Uncertainty Analysis
- Quality Assurance Plan

The NRC staff's review of the documentation provided in the TR and in numerous documents submitted as part of the review revealed that the AURORA-B EM assessment, the field equations and closure relationships within each of the CCDs, and code limitations are adequately described and discussed. The identification of applicable target scenarios and plant configurations for the AURORA-B EM is also presented. These areas are further discussed in Section 3.2 of this SE. Therefore, the NRC staff finds that the AURORA-B EM meets the criteria for the first review area, Documentation.

The NRC staff reviewed the EM and assessed whether models were present for all phenomena and components determined to be important or necessary to simulate the target scenarios under consideration. Many of the CCDs included in the EM have already received prior NRC review and approval. In such cases, the NRC staff examined their applicability to the target scenarios and assessed their coupling to each other within the EM. In those instances where new models were supplied or older ones improved upon to predict BWR phenomena, the NRC staff reviewed the bases for these models. The NRC staff also reviewed comparisons of predicted results to experimental data, benchmarks, and code-to-code comparisons when assessing the adequacy of the CCDs and new/updated models to predict individual phenomena pertinent to the target scenarios. The results demonstrated acceptable predictive capabilities (see Section 3.3). The NRC staff concludes that acceptable models are included in the EM to predict the relevant physical and phenomenological processes important for the transient and accident scenarios discussed in the documentation. Therefore, the NRC staff finds that the AURORA-B EM meets the criteria for the second review area, Evaluation Model.

The NRC staff reviewed the accident scenario identification process and assessed the identification and ranking of the reactor component and physical phenomena modeling requirements based on their importance to the modeling of the target scenarios and their impact on the FoMs. AREVA generated a PIRT, which defines the application of AURORA-B for the analysis of AOOs and certain infrequent event and accident scenarios. Section III of Chapter 15.0.2 of the SRP makes note that a PIRT is one example of an acceptable structured process for identifying and ranking target scenario phenomena. The NRC staff examined the identified phenomena and their ranking as manifest in the PIRT and concluded the process is applicable to the target scenarios identified in the TR. The NRC staff also reviewed the importance of the chosen FoMs with respect to the target scenarios and which of the identified high- and medium-ranked phenomena influence them the most. The NRC staff concluded that each FoM adequately serves to evaluate compliance with one or more reactor safety criteria (Section 3.4.3) that each of the target scenarios challenges. Therefore, the NRC staff concludes that the AURORA-B EM meets the acceptance criteria for the third review area, Accident Scenario and Identification Process.

The NRC staff reviewed the acceptability of the EM's predictive capability over the range of conditions encountered in the target scenarios via assessment of results against both separate effects tests and integral effects tests. As discussed in Section 3.5, these tests were used to demonstrate the ability of the EM to predict one or more highly ranked PIRT phenomena. In general, the AURORA-B EM was shown to reasonably or conservatively predict each of the highly ranked phenomena, even in scenarios with complex phenomenological interactions like the Peach Bottom Turbine Trip Tests. The NRC staff reached a similar conclusion for pertinent medium ranked phenomena. The NRC staff found that the tests provided reasonable assurance that all of the important phenomena for the target scenarios of interest were acceptably treated by the EM, and that the AURORA-B EM demonstrated reasonable predictive capability over the intended range of applications. Therefore, the NRC staff concludes that the AURORA-B EM meets the acceptance criteria for the fourth area of review, Code Assessment.

The NRC staff reviewed the modified uncertainty analysis that AREVA presented in response to staff RAI questions and assessed whether the major sources of uncertainty are addressed consistently with the results of the accident scenario identification process. The modified uncertainty analysis method is predominately discussed in RAI-49. In this approach, the uncertainty associated with each highly ranked (and pertinent medium ranked) PIRT phenomenon is either **[ ]** or considered to be predicted conservatively enough by the EM that it need not be sampled. The NRC staff reviewed how each of the phenomena was dispositioned. For those instances where phenomena were not sampled, the NRC staff concluded the uncertainties were sufficiently covered by conservative predictions, restrictions on user modeling and input (e.g., nodalization), and the imposition of modeling conventions (e.g., **[ ]**) to justify their exclusion from being sampled. For those instances where phenomena were **[ ]**, the NRC staff concluded the **[ ]** the uncertainties are acceptable. Based on the NRC staff evaluations, **[**

] were also found to be acceptable based on their prevention of non-physical sampling results or their correspondence to extremes of physical phenomena. The NRC staff also examined the non-parametric order statistics approach and found it acceptable. However, because it is necessary for the NRC staff to assess the acceptability of [ ] for each sampled parameter, the uncertainty method presented for AURORA-B may not [ ] for which the NRC staff found the uncertainty range methodologies acceptable, as respectively discussed in Section 3.6.4.10, Section 3.6.4.15, and Section 3.6.4.17 of this SE). The parameter uncertainties that have been listed and documented in this report must be used for all safety analyses performed using AURORA-B. Therefore, the NRC staff concludes that the AURORA-B EM meets the acceptance criteria for the fifth area of review, Uncertainty Analysis.

Lastly, the NRC staff reviewed AREVA's quality assurance plan for the EM. Because AREVA has an established NRC-approved software quality assurance program, and the AURORA-B TR discusses both the development and maintenance of the EM under this program, the NRC staff concluded the AURORA-B EM is adequately maintained under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff concludes that the AURORA-B EM meets the acceptance criteria for the final area of review.

In summary, the NRC staff finds that the total assessment of the code, as documented in the TR and various responses to RAI questions, adequately demonstrates that AURORA-B is suitable to model all operating domestic BWR reactor designs, up to and including operation at EPU conditions with expanded power and flow windows, for the target scenarios of interest (excepting those listed in Limitation and Condition No. 15) by demonstrating acceptable performance in each of the highly ranked and pertinent medium-ranked phenomena. The range of applicability includes all forced circulation BWR plant types with external recirculation pumps, both those with jet pumps (BWR/3 through BWR/6 plants) and without (BWR/2). The target scenarios of interest are those AOOs and accidents identified in Section 3.1. Additionally, NRC approval of the AURORA-B EM is contingent upon adherence to the limitations and conditions set forth in Section 5.0.

The NRC evaluation of the CASMO-4/MICROBURN-B2 code system within this SE constitutes qualification of the code system for EPU and EFW conditions. This qualification is contingent upon use of fuel designs for which the [ ] void-quality correlation has been validated, or for which AREVA has justified that the [ ] void-quality correlation is valid at EPU and EFW conditions, as discussed in Section 3.3.2.4.4.

Finally, NRC approval of the AURORA-B EM should not be misconstrued as generic approval of S-RELAP5 for BWR analyses. S-RELAP5's approval for use in BWR analyses extends only insofar as it is incorporated into the AURORA-B EM because the review of the code's applicability to BWRs was performed in part by evaluating the AURORA-B EM as a whole.

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