FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR TOPICAL REPORT ANP-10333P, REVISION 0,

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

APPLICATION TO CONTROL ROD DROP ACCIDENT (CRDA)"

AREVA INC.

PROJECT NO. 728/DOCKET NO. 99902041

1.0 INTRODUCTION

By letter dated March 31, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14098A331), AREVA, Inc. (AREVA), submitted to the U.S. Nuclear Regulatory Commission (NRC) staff for review Topical Report (TR) ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)" (Reference 1, herein described as the "CRDA TR"). The CRDA TR is an extension of the AURORA-B methodologies described in TR ANP-10300P (Reference 2, herein described as the "base AURORA-B TR"); at the time, the base AURORA-B TR had not yet been approved by the NRC. As a result, the CRDA TR was accepted for review but the acceptance letter indicated that NRC staff review would not commence until the base AURORA-B TR review was substantially complete. The base AURORA-B TR has been approved and the CRDA TR review documented in this safety evaluation (SE) considered all findings from the aforementioned review. Consistent with the NRC approval of the methodologies documented in the base AURORA-B TR, this SE addresses the applicability of the CRDA TR to boiling water reactor (BWR) product lines 2-6 (BWRs/2-6) only. The CRDA TR indicates that it can be applied to advanced boiling water reactors (ABWRs), but the NRC has not reviewed the underlying AURORA-B methodologies in the base AURORA-B TR for applicability to ABWRs. Therefore, the expansion of the AURORA-B methodologies for analysis of the CRDA in ABWRs was not evaluated by the NRC staff.

AURORA-B, as described in the base AURORA-B TR, is a multi-physics, multi-code package developed for predicting the dynamic response of BWRs during transient and accident scenarios (with the exception of selected scenarios). AREVA refers to the collection of codes within AURORA-B and the manner of their application as an "evaluation model (EM)." The NRC staff has adopted the same terminology in this SE. One of the scenarios that is not covered by the base AURORA-B TR is the CRDA scenario, so the additional methods and models presented in the CRDA TR are intended to expand the scope of the AURORA-B analysis methodologies to include CRDA analysis. This included methodology enhancements to address CRDA specific applications, characterization of an analysis procedure to identify and assess the limiting CRDA scenarios, assessment of the AURORA-B EM to accurately model CRDA specific phenomena, and evaluation of the uncertainties related to CRDA specific phenomena that are not already assessed as part of the approval of the base AURORA-B TR.

2.0 BACKGROUND

The NRC approved the base AURORA-B TR in Reference 3. This TR describes the base AURORA-B code system and its assessment for a large range of accident and transient scenarios. Much of this material is relevant to the CRDA analysis methodology, so the NRC staff review of the CRDA TR will focus on the specific issues that are unique to the CRDA event compared to the events discussed in the base AURORA-B TR. The limitations and conditions from the SE for the base AURORA-B TR are listed below. The limitations and conditions that specifically affect, or that need to be reconsidered for applicability to, analysis of the CRDA event are marked with an asterisk. All other limitations and conditions remain valid for use of the AURORA-B code system to evaluate the CRDA event. Note that all references to sections are to sections in the SE for the base AURORA-B TR.

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

2. **[**

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   ], 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 2 sigma (σ) levels are presented in Table 3.6 of Reference 3 and discussed in Section 3.6 of Reference 3. 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 of Reference 3 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 of Reference 3, 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 Reference 3.*

5. As discussed in Section 3.3.2.4.4 of Reference 3, 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 extended power uprate (EPU) and extended flow window (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 Reference 3, 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 figures of merit (FoMs) when performing calculations according to the AURORA-B EM described in ANP-10300P.*

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

9. For any highly ranked Phenomena Identification and Ranking Table (PIRT) phenomena whose uncertainties are not addressed in a given **[ ]** analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.2 of Reference 3. For any pertinent medium ranked PIRT phenomena whose uncertainties are not addressed in a given **[ ]** analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.4 of Reference 3.*

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 Reference 3.*

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 Reference 3.*

13. The AURORA-B uncertainty methodology discussed in Section 3.6 of Reference 3 may be used in licensing applications for the events listed in Section 3.1 of Reference 3, with the exception of three specific events identified in Section 3.6.2 of Reference 3: **[** **]**. 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 Reference 3.*

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 Reference 3 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 Reference 3, 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.*

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 Reference 3, the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EM. 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”. 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 Reference 3 without prior NRC approval.

25. The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of Reference 3.

26. AREVA must continue to use existing regulatory processes for any **[**

] areas discussed in Section 4.0 of Reference 3.

The items marked with asterisks affect the acceptability of the use of the AURORA-B code system for CRDA analyses or are not applicable to CRDA analyses. A brief summary of the NRC staff considerations related to these limitations and conditions as applied to the analysis of the CRDA event is provided below.

- Items 2, 5, 12, 13, 16, and 17 concern phenomena that are not of any significance for the CRDA event, so they are not necessary for the specific purpose of using the AURORA-B EM to analyze the CRDA event.

- Items 3, 4, 6-11, 18, and 19 concern the uncertainties associated with the anticipated operational occurrences discussed in the base AURORA-B TR. The CRDA TR handles the uncertainties specific to the CRDA event's PIRT in a different manner, as discussed in Section 4.4, and these conditions are superseded by conditions specific to use of AURORA-B for CRDA analysis, as discussed in Section 5.0.

- Item 15 concerns the applicability of the AURORA-B EM to BWR/2s at EPU or EFW conditions. The model used to evaluate the CRDA TR is limited to the core region, and consequently, does not include any of the design features unique to BWR/2s. Therefore, limiting the applicability of the AURORA-B EM to analyze CRDA events for BWR/2s is not necessary.

- Items 20 and 21 are addressed for the CRDA analysis methodology and the associated code enhancements via NRC approval of this TR. However, consistent with these items, NRC approval does not generically extend to use of the code enhancements as part of analyses for events other than the CRDA.

As stated previously, all other limitations and conditions associated with the base AURORA-B TR remain applicable for CRDA analyses.

3.0 REGULATORY EVALUATION

The regulation at 10 CFR 50.34, "Contents of Applications; Technical Information," requires that the licensee/applicant provide safety analysis reports to the NRC detailing the performance of systems, structures, and components provided for the prevention or mitigation of potential accidents.

General Design Criterion (GDC) 13, "Instrumentation and control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," addresses the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges. This regulatory requirement primarily applies to ensuring that the limiting system

operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded. This is satisfied by ensuring that the initial conditions represented in the CRDA analyses are sufficiently representative of the most conservative condition allowed by the aforementioned controls.

The regulations at GDC 28 of 10 CFR Part 50, Appendix A, require that the effects of postulated reactivity accidents result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor result in sufficient damage to significantly impair core cooling capacity.

The regulations at 10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

The acceptance criteria for CRDA events to satisfy GDC 28, 10 CFR 100.11, and 10 CFR 50.67 are defined in Chapter 15 of the Standard Review Plan (SRP), otherwise known as NUREG-800 (Reference 4). Satisfying these acceptance criteria is necessary for CRDA events to meet the aforementioned regulatory requirements. Specifically, SRP Section 15.4.9.II states the acceptance criteria are:

1. Acceptance criteria from SRP Chapter 4.2. Appendix B, in particular, provides interim acceptance criteria for reactivity initiated accidents (RIAs).

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that causes stress to exceed the “Service Limit C” as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

SRP Section 4.2 provides an extensive discussion of acceptance criteria related to high temperature cladding failure, pellet clad mechanical interaction (PCMI) induced cladding failure, core coolability, and fission product inventory determination for dose assessment purposes. Regulatory Guides 1.183 and 1.195 are also referenced for further guidance related to fission product inventories.

The NRC staff is currently developing new guidance for RIA acceptance criteria that will supersede SRP Section 4.2. As part of this review, the NRC staff considered the applicability of the TR methodology to the current draft guidance, DG-1327 (Reference 5). Where appropriate, the draft criteria, along with any potential implications to acceptability of the TR methodology, are discussed in this SE. Prior to use of this TR methodology with the final approved RIA acceptance criteria, any changes relative to this version of DG-1327, as released for public comment on November 21, 2016, must be evaluated to verify that there have been no changes beyond clarifications or editorial changes consistent with the discussion of the criteria in this SE or adjustments to the numeric thresholds for specified limits that do not go outside the bounds of the values used to validate the methodology and uncertainties discussed in the TR.

The CRDA TR is an application of an EM to perform licensing analyses for an accident that the EM has not previously been approved. As such, additional guidance for the evaluation may be found in SRP Chapter 15.0.2, “Review of Transient and Accident Analysis Methods” (Reference 4). This chapter includes provisions for the review of submittals related to evaluation models, which can also be applied to EMs.

In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 along with the applicable acceptance criteria in SRP Chapters 4.2 and 15.4.9 in conducting its review of the CRDA TR. The new acceptance criteria applicable to the CRDA event contained in DG-1327 was also considered, with the understanding that the guidance has not yet been finalized. In accordance with SRP Chapter 15.0.2, the review covered the areas of: (1) documentation, (2) EM, (3) accident scenario identification process, (4) code assessment, (5) uncertainty analysis, and (6) quality assurance plan. To the extent possible, the NRC staff leveraged the prior review and approval of the AURORA-B code system for anticipated operational occurrence (AOO) analyses in the base AURORA-B TR.

4.0 TECHNICAL EVALUATION

The CRDA TR describes a methodology by which the AURORA-B code system approved in the base AURORA-B TR can be extended for use in analyzing the CRDA event. The NRC staff review of the CRDA TR focused on four specific areas:

1. Accident scenario description and phenomena identification and ranking – the licensee's break-down of the CRDA event, characterization and ranking of the pertinent phenomena, and characterization of the consequences (i.e., FoMs).

2. Evaluation methodology – the proposed CRDA analysis methodology, including initial conditions, assumptions, and approach to ensuring that the SRP Chapters 4.2 and 15.4.9 acceptance criteria are met. Since this methodology includes use of the EM, by extension, this area includes the application of the EM to analyze the CRDA event.

3. Code assessment – the assessments performed by the licensee to validate the AURORA-B code system performance for CRDA specific phenomena.

4. Uncertainty analysis – the licensee's evaluation and propagation of uncertainties in the analysis.

In addition, the NRC staff considered whether the licensee provided adequate quality assurance (QA) and documentation support for the CRDA methodology. This aspect is not explicitly discussed in detail for this SE because the bulk of the QA and documentation support is captured by the various QA program documents, code documentation, and methodology discussion in the base AURORA-B TR. The additional documentation required to address the CRDA methodology is largely captured by the CRDA TR. As such, the NRC staff acceptance of the adequacy of the licensee's discussion of each area implicitly includes acceptance of the licensee documentation associated with that area. Where applicable, any documentation inadequacies are discussed as part of the NRC staff's considerations.

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

4.1 Accident Scenario Description and Phenomena Identification and Ranking

As per the review guidance in Chapter 15.0.2 of the SRP, the accident scenario description and phenomena identification and ranking process is intended to ensure that the dominant physical phenomena influencing the outcome of the given accident scenario are correctly identified and ranked. Once an accident scenario has been described, then FoMs can be determined for use in evaluating whether acceptance criteria are met. The subsequent phenomena identification

and ranking process will determine the physical phenomena affecting the FoMs and rank them by their importance. By doing so, AREVA can demonstrate that reasonable assurance exists that they are accurately capturing and modeling the dominant physical phenomena necessary for evaluation of the accident scenario in question.

Section 4 of the CRDA TR describes the accident scenario. The description of the CRDA event is consistent with other readily available documents, such as updated final safety analysis reports and other TRs related to BWR CRDA events. The scenario is relatively simple in that it consists of a rapid reactivity addition due to a single control rod falling out of the core. The resulting local power excursion is terminated primarily by Doppler reactivity feedback as the fuel temperature increases. Long term shutdown is assured by negative thermal hydraulic reactivity feedback and/or a reactor scram. The CRDA event may occur during startup or when the reactor is operating at full power. In the former case, constraints imposed on rod movements due to technical specification (TS) restrictions and rod withdrawal sequences serve to limit the potential rod patterns and the resulting rod worths. In the latter case, the initial operating characteristics of the fuel and moderator lend themselves to more effective Doppler reactivity feedback and quicker thermal hydraulic reactivity feedback through increased voiding from direct moderator heating.

Section 5 of the CRDA TR discusses the relevant FoMs, which are directly derived from the applicable acceptance criteria in SRP 15.4.9.II (and, by extension, the interim RIA acceptance criteria in SRP 4.2 Appendix B). They are: (1) fuel enthalpy, (2) minimum critical power ratio (MCPR), (3) peak system pressure, (4) fission product inventory released, and (5) core coolability. The acceptance criteria in DG-1327 are based on the same parameters. Of these parameters, the MCPR and peak system pressure FoMs are addressed in the base AURORA-B TR. While the base AURORA-B TR focuses on AOOs that are driven by global processes rather than the kind of highly localized processes that drive the CRDA event, the phenomena that affect the MCPR and peak system pressure FoMs are similar. The fission product inventory released and core coolability are both evaluated based on secondary parameters derived from the calculated fuel enthalpy data (i.e., fission product inventory released is calculated based on the number of fuel rods predicted to fail based on fuel enthalpy, and core coolability is determined based on enthalpy-driven thermal hydraulic processes). Therefore, the NRC staff finds it acceptable that the CRDA TR only addresses the phenomena identification and ranking for the fuel enthalpy FoM.

The NRC staff reviewed the PIRT provided as Table 5.1 in the CRDA TR. The identified phenomena were consistent with PIRTs for other RIAs, prior precedents for the CRDA event, and the NRC staff's technical understanding of the relevant events in the accident progression. In summary, the PIRT includes: (1) initial conditions that would affect initial enthalpy or reactivity feedback, (2) parameters that would affect the positive reactivity addition from the rod drop, (3) parameters that would affect the timing and/or magnitude of the negative reactivity feedback terminating the power excursion, and (4) parameters affecting the transfer of heat away from the limiting locations. The importance assigned to each parameter is generally consistent with the results from sensitivity studies documented later in the CRDA TR.

For BWRs, past precedents and the sensitivity studies documented in the CRDA TR show that in the absence of specific controls intended to minimize the potential consequences of the CRDA, the conditions which maximize the potential for fuel failures occur at cold zero power (CZP) conditions. This is due to the fact that increased temperatures result in increased mitigation via Doppler and moderator reactivity feedback mechanisms (see Section 4.2.2.2.1 for further discussion). The short time scale for the CZP CRDA scenario means that thermal

hydraulic feedback is of relatively little consequence, since the limiting parameters reach their maximum values before significant heat transfer occurs. Consequently, the primary phenomena affecting the CRDA event are expected to be those that affect the magnitude of the reactivity addition or the Doppler reactivity feedback. The importance assigned to the phenomena listed in the PIRT is consistent with this expectation.

As a result of the above discussion, the NRC staff has determined that AREVA appropriately characterized the CRDA scenario, identified the appropriate FoMs, and evaluated the sensitivity of the FoMs to the relevant phenomena.

4.2 Applicability of Evaluation Model to CRDA Event

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

Each of the CCDs within the AURORA-B EM (S-RELAP5, MICROBURN-B2, MB2-K, and RODEX4) have been evaluated and found to be acceptable for performing safety analyses during the review and approval of their individual TRs as well as the base AURORA-B TR. As a result, this review focused on how the AURORA-B EM is applied to analyze the CRDA event. The scope of this review included: enhancements or modifications made to the CCDs to address the CRDA event, applicability of the modeling schemes discussed in the base AURORA-B TR to the CRDA event, and any potential limitations to the proposed analysis procedure to identify and assess the limiting CRDA scenarios.

4.2.1 Code Modifications to the AURORA-B Evaluation Model

A number of new methods and models were incorporated into the AURORA-B CCDs to address CRDA specific phenomena, including:

1. *Pin Power Reconstruction at Cold Conditions:* The MB2-K pin power reconstruction methodology performed at hot conditions is extended to cold conditions, in order to allow for determination of peak pin powers during the CRDA.

2. *Peak Rod Heat Structure:* **[**

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3. *Hydrogen Pick-up Model:* The pellet-clad mechanical interaction (PCMI) failure criteria are based in part on the hydrogen content of the cladding. Therefore, a hydrogen pick-up model needed to be included to calculate and track this parameter.

4. *Additional Information Input for Codes:* New information inputs were added to MB2-K and S-RELAP5, the neutronic and thermal-hydraulic codes of the AURORA-B code

system, respectively. These inputs provide additional information needed by the AURORA-B code system to produce some of the CRDA specific parameters.

Each modification to the CCDs approved as part of the review of the base AURORA-B TR is discussed in the subsequent subsections.

4.2.1.1 Pin Power Reconstruction at Cold Conditions

The MB2-K CCD was updated to allow use of the pin power reconstruction methodology at cold as well as hot conditions. The use of the pin power reconstruction methodology was assessed as part of the validation of the MICROBURN-B2/CASMO-4 methodology that was approved by the NRC (Reference 6). The supporting TR describes validation that was performed using cold and hot conditions. The main purpose of the pin power reconstruction methodology is to capture the impact of highly localized flux conditions experienced by individual pins, such as the presence of a nearby control rod. The validation suite in Reference 6 did not explicitly include pin power reconstruction at cold conditions. However, the performance of the overall neutronics module and cross section libraries were assessed at cold conditions, including criticals and reaction rate measurements for cold critical experiments and cold criticals from commercial reactors. The pin power reconstruction methodology itself was validated against gamma scans from commercial reactors. While the gamma scans are representative of pin powers at hot conditions rather than cold conditions, the neutronic relationships used in the pin power reconstruction are expected to be applicable for cold conditions as well.

The CRDA TR documents sensitivity studies in Section 8.7.2.6 that were performed to assess the use of nodal average powers in constructing the pin-specific power history effects with the AURORA-B pin power reconstruction for hot operating conditions with respect to the pin-specific enthalpy calculated for the CRDA TR. The studies included in Revision 0 of the CRDA TR did not seem to separate out the burnup and power history effects on the calculated pin enthalpies, so the NRC staff asked AREVA to provide a clearer understanding of how the power history affects the pin enthalpies independently from the burnup of the fuel. In response to RAI-3 (Reference 8), AREVA provided the results from additional studies which varied the depletion power **[ ]**. The studies confirmed the independence of the **[ ]**[1] relative to the depletion power, which is consistent with the fact that this part of the CRDA transient is essentially adiabatic. Therefore, the change in gap thermal conductivity due to the variation in fission gas releases as a result of increased depletion powers would not affect the magnitude of the **[ ]**. At high depletion powers, the reduction in gap thermal conductivity can slow the conduction of heat out of the pellet, which may increase the **[ ]**. The studies provided by AREVA show that the change in **[ ]** due to rod power history effects are not significant for the range of variation associated with the known uncertainties on nodal power distribution and pin peaking factors.

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[1] Three terms will be used frequently in this SE related to enthalpy. Two of them, the prompt enthalpy rise and the total enthalpy, are defined in SRP 4.2 Appendix B. The prompt enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt power pulse. The total enthalpy is defined as the maximum radial average fuel enthalpy achieved during the transient. The third term, the total enthalpy rise, is defined as the radial average fuel enthalpy rise at the time corresponding to the total enthalpy. In the CRDA TR, the total enthalpy is computed by determining the total enthalpy rise, applying the uncertainty multiplier, and then adding the result to the initial radial average fuel enthalpy. This is appropriate because the uncertainty in the initial radial average fuel enthalpy for CZP conditions is very small relative to the uncertainty in the enthalpy rise.

As a result of the above discussion, using rod power history effects constructed based on nodal average powers, the NRC staff concludes that extension of the pin power reconstruction methodology for use at cold conditions is appropriate.

4.2.1.2 Peak Rod Heat Structure

AREVA describes the addition of a peak rod heat structure to the S-RELAP5 CCD's modeling capabilities. **[**

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The heat transfer models used to compute the peak fuel enthalpy based on the maximum peaking factor for each axial node are the same models reviewed and approved by the NRC for the base AURORA-B TR and the S-RELAP5 TR. Since the peak rod heat structure **[**

]. Therefore, the peak rod heat structure does not need additional validation to justify its acceptability for its intended use.

4.2.1.3 Hydrogen Pick-Up Model

The current SRP 4.2 Appendix B criteria and the proposed DG-1327 criteria for correlating the enthalpy rise for fuel rods with probable failure due to PCMI are at least partially dependent on the hydrogen present in the cladding as zirconium hydrides. This hydrogen content is dependent on the past operating conditions experienced by the fuel rods. Therefore, an explicit hydrogen uptake model is necessary to support an assessment of the number of fuel rods expected to experience PCMI failure. This model was submitted to the NRC as part of the most recent RODEX4 supplement, and subsequently approved by the NRC in Reference 7 for cold-worked, stress-relieved, and recrystallized Zircaloy-2 cladding. Hydrogen pickup models for any other cladding types would need to be approved by the NRC prior to use in evaluation of the PCMI failure criteria. The hydrogen content is calculated by the hydrogen pickup model, but does not impact the thermal hydraulic performance of the fuel.

Since the model has been approved by the NRC for its intended use in calculating hydrogen uptake and does not need to be evaluated for its impact on the CRDA event, it is acceptable for its intended purpose in applying the PCMI failure acceptance criteria.

4.2.1.4 Miscellaneous Evaluation Model Enhancements

A number of additional enhancements were made to the AURORA-B CCDs to support analysis of the CRDA event. The primary enhancements described in the CRDA TR are inclusion of the moderator temperature in the cold cross section library generation to support cold voided feedback, and data transfer between MB2-K and S-RELAP5 (specifically the peak pin powers and moderator temperatures). These enhancements support the application of existing CCD

capabilities to the CRDA event, and do not extend the CCD capabilities beyond the range for which they have already been validated. Therefore, no further validation is necessary to justify the acceptability of these enhancements for use in analyzing the CRDA event.

4.2.2 Applicability of AURORA-B Modeling Schemes to CRDA

The CRDA TR describes the coupling between the three AURORA-B CCDs (MB2-K, S-RELAP5, and RODEX4). This coupling is the same as that described in the base AURORA-B TR, so the NRC staff's review of the AURORA-B modeling schemes focused on ensuring that the modeling guidance for CRDA specific applications is appropriate. This review included an assessment of the modeling guidance provided in the CRDA TR and the procedure provided for performance of the CRDA analysis.

4.2.2.1 Modeling Guidance

The modeling guidance provided in the CRDA TR includes guidance on modeling of the plant hydraulic components, nodalization (including channel grouping), time step size specification, and characterization of the effective Doppler temperature. In addition, the NRC staff considered the potential impact for physical core configurations which have not previously been assessed, such as axially heterogeneous control rods or features that may be relevant for other vendors' fuel designs.

4.2.2.1.1 Overall Plant Modeling Guidance

The overall plant modeling recommendations provided in the CRDA TR essentially consist of a "core-only" model. Much of the plant, including the recirculation pumps, downcomer, steam dryers/separators, and systems external to the reactor pressure vessel, are not included in the model. The CRDA event is a fast, localized event that will terminate before any significant feedback can occur from outside the core. As a result, there is not much need to model thermal hydraulic components outside the core. **[**

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The nodalization within the core is consistent with the nodalization of the steady state core simulator (MICROBURN-B2) and includes the fuel channels and bypass. This nodalization has been previously validated as part of the MICROBURN-B2/CASMO4 methodology (Reference 6) as being sufficient to capture the impact of local reactivity characteristics. **[**

]. A fuel channel grouping strategy is used to reduce the number of fuel channel models necessary (see Section 4.2.2.1.2 for further discussion of this strategy). The fuel channels are modeled using thermal hydraulic parameters consistent with the physical geometry and composition of the fuel and related components. **[**

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One of the RIA acceptance criteria involves thermal hydraulic conditions outside of the core the verification that the peak system pressure does not cause stresses to exceed Emergency

Condition (Service Limit C), as defined in Section III of the ASME Boiler and Pressure Vessel Code (Reference 9). The CRDA TR shows a representative evaluation of the pressure response due to a CRDA event, using the lower plenum as the reference thermal hydraulic component. The greatly simplified modeling of the volumes within the reactor pressure vessel (RPV) does not lend itself to a very sophisticated evaluation, but any significant pressure response would be expected to manifest in the lower plenum. The amount of energy generated by a CRDA is expected to be very small relative to the coolant volume within the RPV. Consistent with this expectation, the representative evaluation shows minimal impact on the system pressure. Section 9.0 of the CRDA TR describes how the reference EM is to be implemented for the CRDA analysis, and the peak system pressure evaluation is not included. This is acceptable because plant-specific and cycle-specific variations are not expected to yield a significant enough change in the total energy generated by a CRDA event to challenge the limit on RPV pressure.

Based on the above discussion, the primary influences on the limiting FoMs are captured at a level of detail consistent with the fidelity needed to accurately capture the CRDA event. Therefore, the NRC staff finds the proposed “core-only” modeling scheme for the thermal hydraulic components to be acceptable.

4.2.2.1.2 Channel Grouping Guidance

A key modeling approach that differs significantly from the base AURORA-B TR is the use of fuel channel grouping. The CRDA event primarily impacts the fuel assemblies grouped near the control rod of interest, so computational time savings can be realized by fuel channel grouping for fuel assemblies far from the control rod of interest. This is a strategy in which multiple fuel channels are modeled as a single thermal hydraulic component in the S-RELAP5 thermal hydraulic model, even though they are modeled as separate fuel assemblies in the MB2-K core simulator model. This modeling approach effectively averages the fuel and moderator temperature response from the change in power for all fuel assemblies in a given group, then feeds the averages back to MB2-K for use in the neutronics calculation for each individual fuel assembly in that group.

When this type of approach is adopted, in order to ensure that the results are not non-conservative, the guidance for grouping channels must be established in a manner that ensures that:

1. Individual fuel channels are modeled when necessary, in order to capture highly localized limiting phenomena;

2. Fuel channels that are combined into a single thermal hydraulic component are hydraulically similar, so the averaging of thermal hydraulic properties is performed based on a consistent axial distribution of key hydraulic parameters; and

3. Any other possible variations in input parameters would yield equivalent or conservative results relative to a higher resolution model.

The CRDA TR describes the approach used to determine how to select the fuel channel groups. First, the individual fuel assemblies are explicitly modeled for either **[ ]** around the target rod for the drop evaluation. Secondly, fuel may be associated with **[**

] (referred to as “buffer

rings"). Finally, the remainder of the fuel assemblies is grouped on a core-wide basis. The grouping algorithm is intended to ensure that all fuel assemblies assigned to a specific group will have identical geometric design, orifice design, nuclear design, and were part of the same reload batch. This implies that there will usually be multiple fuel channel thermal hydraulic structures for each ring. The first two grouping parameters ensure that (2) above is met. The last two grouping parameters ensure that (3) is partially met by ensuring that the variation in neutronic parameters for fuel assemblies grouped together is not too severe.

Section 8.7.1 of the CRDA TR documents a series of sensitivity studies performed using the reference analyses to assess the impact of variations in the channel grouping strategy. In summary, two different grouping parameters were perturbed: the number of individual fuel channels explicitly modeled, and the number of buffer rings. Two general conclusions can be gleaned from the study results. First, the enthalpy results calculated **[**

] (i.e., (1) above is met **[**

] leads to a suppressed negative reactivity feedback response due to the lower temperatures of the **[** **]** fuel. A similar logic can be used to infer that other influences such as variations in burnup or power within a channel grouping would yield slightly more conservative results due to the dampening of the Doppler reactivity feedback mechanism for the more reactive fuel elements in the group (i.e., (3) above is met). AREVA does make a specific recommendation that the number of buffer rings should be **[** **]**, but this recommendation is driven by the balance between calculational expense due to additional channel modeling and the potential for unnecessary conservatism due to dampening the negative Doppler feedback response for fuel near the region of interest. As long as the channel grouping strategy follows the guidance outlined in the CRDA TR, any number of buffer rings would be acceptable.

As a result of the above considerations regarding the potential impacts of the channel grouping strategy on the results calculated for the CRDA event, and inferences from the AREVA sensitivity studies, the NRC staff finds the proposed fuel channel grouping strategy to be acceptable.

4.2.2.1.3 Time Step Guidance

The base AURORA-B TR addressed time step sizes for AOO analyses. However, the CRDA event is a much faster transient, which implies that smaller time steps may be necessary. The NRC staff asked for clarification on how the maximum time step size recommendation was established. AREVA replied to RAI-11 by providing more information (Reference 8) related to the sensitivity studies on the maximum time step size than what was provided in the CRDA TR. AREVA explained that their maximum time step size recommendation was based on the fact that this value appeared to be the point at which the calculated enthalpy curves begin to converge for smaller maximum time step sizes.

In general, larger maximum time step sizes may result in a delay in the neutronic response to fuel temperature changes. The sensitivity study results show **[**

]. There is no clear trend in the relationship of the prompt

enthalpy rise with the maximum time step size, but the prompt enthalpy rise is a result of competing reactivity effects between the positive reactivity insertion from the rod drop and the subsequent negative Doppler reactivity feedback that terminates the power pulse. Small changes in the time sensitivity of these effects can lead to changes in the slope of the prompt enthalpy rise as well as a change in the definition of the time at which the prompt enthalpy rise is determined (due to the fact that the prompt enthalpy rise is defined based on the width of the power pulse). The AREVA recommendation is consistent with a value that appears to result in reasonable convergence, but also bounds the prompt enthalpies for the reference analysis.

The NRC staff considered the impact of the time step size recommendations on the stability of the numeric convergence. Implicit numeric schemes, such as those used in the MB2-K CCD, do not generally suffer from accuracy problems as time step sizes become smaller. The primary issue with coupled solutions using implicit numeric schemes is related to the convergence problems that may arise from the efforts by the code to iterate between unstable thermal hydraulic phenomena and the reactivity feedback. For the CRDA event, the limiting scenario occurs during CZP conditions. Direct moderator heating would not result in much change in the moderator conditions due to the significant amount of subcooling that exists. Therefore, the smaller time steps would not be expected to cause instability in the coupled calculation for the CRDA event, unlike the AOO events from the base AURORA-B TR which involve significant variations in fluid conditions.

As a result of the above discussion, the NRC staff finds the time step recommendation in the CRDA TR to be reasonable for its intended application.

4.2.2.1.4 Effective Doppler Temperature Characterization

The CRDA TR describes how relevant information is passed between the S-RELAP5 and MB2-K CCDs for the purpose of calculating the Doppler reactivity feedback. When the power excursion occurs due to a CRDA event, then a complicated interplay of different factors may affect the overall Doppler reactivity response for a given node. Some examples include varying radial temperature profiles within the fuel pellet, self-shielding, and fuel composition variations due to irradiation. The CRDA TR states that the Doppler reactivity feedback is incorporated in the MB2-K calculation models via use of a “Doppler effective” temperature. **[**

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The CRDA event at CZP conditions is a very fast transient for which the Doppler reactivity feedback mechanism is the primary means to mitigate the event consequences. The base AURORA-B TR does consider the Doppler reactivity feedback in AOO events, which provides assurance that the overall reactivity feedback impact for at-power CRDA events is appropriately incorporated. However, the Doppler reactivity effect is difficult to separate from other phenomena such as the moderator reactivity feedback. As a result, the NRC staff requested further justification that the selected weighting was appropriate for use when analyzing CRDA events. In the response to RAI-1 (Reference 8), AREVA provided an extended discussion of the radial temperature response for fuel pellets during the CRDA event and the impact of changes in the pellet due to irradiation. The physical effects of fuel pellet geometry are captured by the RODEX4 CCD in defining the fuel properties, which is an NRC-approved methodology for this purpose. The radial temperature profile response requires further consideration.

For burned fuel, there is more fissile material near the pellet surface due to plutonium (Pu) buildup as a result of neutron capture. Pu production is less pronounced in the pellet interior because of the self-shielding effect, in which neutrons at lower energies are absorbed before they can travel into the pellet interior. When the power excursion associated with the CRDA event occurs, this results in more power production, as well as a more rapid temperature rise, near the pellet surface. After the initial power pulse is arrested, heat is quickly conducted from the pellet surface to the cladding and surrounding coolant, while the temperatures in the pellet interior continue to rise.

In general, use of a lower effective Doppler temperature would result in less Doppler reactivity feedback, so the weighting needs to be demonstrated to be satisfactorily conservative with respect to the temperature response for different parts of the fuel pellet. The CRDA TR and RAI-1 response indicates that the **[** **]**, but the pellet surface temperature is given some weight. In order to determine whether this approach is reasonably conservative, the NRC staff considered two separate phases of the temperature transient for the CRDA event at CZP conditions. The two enthalpy dependent acceptance criteria associated with PCMI failure and high temperature failure are the prompt enthalpy rise and the total enthalpy, respectively, and the limiting values for each parameter are primarily dependent on different phases of the temperature transient.

The first phase is a very rapid temperature increase due to the prompt power excursion caused by the rod drop. This phase is essentially adiabatic because the duration of the temperature increase is much smaller than the time constant for heat conduction. The fuel pellets start with a flat temperature profile, so the shape of the radial temperature profile immediately after the prompt power pulse will be proportional to the radial power distribution. **[** **]**. The net result is that the pellet surface temperature increases more rapidly than the pellet average temperature, so inclusion of the pellet surface temperature in the Doppler effective temperature will have the effect of increasing the Doppler reactivity feedback during the initial prompt power pulse. However, the studies performed by AREVA show that while the strengthening of the Doppler reactivity feedback results in a lower peak power, **[** **]**. The prompt enthalpy rise is defined as the enthalpy rise at the time corresponding to one pulse width after the peak of the prompt power pulse. Therefore, even though the net power generation is smaller, **[** **]** results in a higher calculated prompt enthalpy rise, even though the peak power for the CRDA transient is lower. The NRC staff noted that DG-1327 does not clearly include the concept of the prompt enthalpy rise, however, this is expected to be captured in the final regulatory guidance. If it is not, then this discussion needs to be re-visited to confirm that this analytical approach continues to yield more conservative results for the enthalpy rise value used in evaluating PCMI failures.

The second phase of the temperature transient is a gradual increase of the overall pellet temperature due to continuing power generation (albeit at a much lower level than the prompt power excursion). A limiting temperature value is reached when heat conduction becomes established enough such that the heat transfer from the pellet interior to the coolant is sufficient to compensate for any residual power generation. As discussed in the AREVA response to RAI-1, the pellet surface temperature begins to decrease once the essentially adiabatic phase

of the temperature transient ends, because it is adjacent to the much cooler cladding. Heat transfer is much slower from the interior of the pellet, so the net effect of giving **[ ]** weight to the surface temperature in the Doppler effective temperature is to **[**

] will increase the total enthalpy.

As a result of the above discussion, the NRC staff has determined that both the prompt enthalpy rise and the total enthalpy are calculated to be higher for the CRDA event at CZP conditions for the given weighting. In reality, the pellet surface temperature should be weighted relative to the temperatures in the rest of the pellet because the Doppler reactivity feedback response is non-linear with respect to fuel temperature. **[**

]. Since this will lead to more conservative calculated prompt enthalpy rise and total enthalpy values, the NRC staff finds this to be acceptable.

4.2.2.1.5 Miscellaneous Modeling Scheme Considerations

The CRDA TR describes a “typical” model, but does not appear to clarify how the guidance should be applied to “atypical” models such as axially heterogeneous control rods or mixed cores. In order to verify the applicability of the CRDA modeling guidance to these situations, the NRC staff asked RAI-2 to address axially heterogeneous control rods such as rods with hafnium tips, and RAI-8 to clarify what limitations or changes would be necessary to account for cores with non-AREVA fuel.

In the response to RAI-2 (Reference 8), AREVA indicated that modeling a uniform axial control rod composition that has a worth equivalent to or greater than the original equipment control rod would be conservative. The justification provided is that control rod designs which have axially heterogeneous designs, due to hafnium tips or removal of absorber material near the bottom, tend to have lower worth in these segments. Therefore, modeling the rods as axially uniform blades with a composition consistent with the dominant (higher-worth) axial zone ensures that the reactivity addition due to the CRDA is higher. **[**

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The NRC staff agrees in principle that modeling currently known axially heterogeneous control rod designs as axially uniform control rods based on their dominant, higher worth axial zone, is conservative because this approach would conservatively bound the reactivity impact of the rod drop. **[**

]. Therefore, the NRC staff is including a condition in Section 5.0 which states that use of different control rod designs within the same core shall be captured by ensuring that the control rod geometry and composition

used in the model for each control rod bounds the worth for the physical control rod used at that location, for all axial elevations.

For mixed cores, AREVA stated in their response to RAI-8 (Reference 8) that the fuel mechanical properties and thermal hydraulic properties for non-AREVA fuel will be evaluated using NRC-approved models and correlations, as appropriate. This includes hydrogen uptake models, additive constants to support CPR correlations, and so on.

The additional information provided by AREVA was acceptable to demonstrate that the CRDA modeling description provides appropriate guidance to capture the impact of fuel designs and control rod designs other than those captured in the reference analysis, with one exception. The exception, associated with the underlying assumptions associated with control rod modeling, is addressed via a condition in Section 5.0.

4.2.2.2 CRDA Analysis Procedure

The CRDA TR provides a specific procedure for performance of the CRDA analysis, which includes a description of what conditions should be evaluated, which control rods should be selected for evaluation, and how the acceptance criteria should be verified to have been met. Section 4.2.2.2.1 discusses the at-power CRDA scenario, and the remainder of the subsections discuss the CZP CRDA scenario.

4.2.2.2.1 At-Power CRDA Scenario

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]. The SRP 4.2 Appendix B acceptance criteria for power levels above 5% indicate that the minimum critical power ratio (MCPR) is appropriate for use in determining the high temperature failure threshold. Therefore, the CRDA TR provides an assessment of the at-power CRDA conditions for the reference model **[**

]. The at-power CRDA scenario is distinguished from the CZP CRDA scenario by the presence of increased negative reactivity via the following mechanisms:

1. The presence of significant voiding in the coolant results in less moderation, so neutron spectrum skews more towards faster neutrons (i.e., the spectrum is “harder”). Consequently, the control rod absorber material is less effective at neutron absorption (i.e., rod worths are lower) and the reactivity consequence of the rod drop itself is milder.

2. The coolant is at saturated conditions, so the direct heating of the coolant can produce voiding. This produces a significant negative moderator density feedback effect that is not present for CZP conditions where the direct coolant heating does not result in a significant change in the coolant density.

3. While the magnitude of the Doppler reactivity coefficient tends to be smaller at higher fuel temperatures, the harder neutron spectrum results in a larger number of neutrons available for Doppler capture in the resonance regions.

The reference analysis presented in the CRDA TR confirms that the prompt power pulse for the at-power CRDA event is much broader **[ ]**, compared to the CZP CRDA event. The relatively slow power increase indicates that the prompt enthalpy rise will not be limiting for the PCMI failure mechanism. The CRDA analysis performed by AREVA, including an evaluation of the potential impact of suppression rods in response to RAI-7 from the NRC staff, show that the **[ ]**. This information also indirectly demonstrates that the radiological consequence and core coolability acceptance criteria are met, in that it shows that no radiological releases are expected and **[ ]**. The RPV pressure acceptance criterion was also evaluated for at-power CRDA events and was determined to have been met; further discussion can be found in Section 4.2.2.1.1.

The analyses performed by AREVA for at-power CRDA events alone are **[ ]**. However, these results are consistent with previous analyses of the CRDA event using other methodologies and the NRC staff's understanding of the relevant phenomena. Therefore, the NRC staff finds the information presented in the CRDA TR to be acceptable to demonstrate that the at-power CRDA event continues to be **[ ]**, when using the AURORA-B EM.

4.2.2.2.2 CZP CRDA Scenario: Plant Parameters

The CRDA TR discusses specific plant parameters that need to be considered for the CRDA event, and provides recommendations for values that should be used in the analysis of the CRDA event. When appropriate, sensitivity studies were performed to justify the recommendation. The key plant parameter recommendations were associated with: control rod parameters, initial conditions, and control rod pattern selection.

Control rod parameters include scram setpoints, scram delay times, scram speeds, and rod drop velocity. In all cases, values were selected based on maximizing the positive reactivity addition. The reactor scram is of relatively low importance in the CRDA event due to the fact that the Doppler reactivity feedback terminates the power excursion before the rods start inserting for a scram, so the guidance merely indicates that conservative values should be used that will delay the full insertion time as long as possible. The control rod drop velocity is a much more important parameter in that maximizing the rod drop velocity will result in as rapid an addition of positive reactivity as possible, amplifying the prompt power excursion before it is arrested by Doppler reactivity feedback. The BWR/2-6 designs incorporate control rod velocity limiters, which impose an upper limit on how quickly the control rods can fall. The CRDA TR recommends a value of 3.11 feet per second (ft/s) to bound the results of control rod velocity limiter tests performed by the General Electric Company. This justification is acceptable, however, there is no guarantee that this value would be bounding for all plants utilizing this methodology. Therefore, the NRC staff is including a condition in Section 5.0 which requires licensees to verify that their maximum control rod velocity is bounded by the recommended value, either by confirming that their control rod velocity limiters are consistent with the ones tested in the report referenced by the CRDA TR, or by referencing other test data applicable to their control rod velocity limiters.

The initial conditions recommended for use are listed in Table 9.2 of the CRDA TR. Sensitivity studies generally confirm that the initial conditions that affect the enthalpies the most are the fuel and moderator temperature, due to the non-linear nature of the Doppler reactivity feedback response as a function of temperature. The NRC staff clarified via RAI-4 (Reference 8) that the AURORA-B code system normalizes the eigenvalues to criticality, so the sensitivity studies on the initial core temperatures are not biased by the effect of the starting temperature on the overall core reactivity. Therefore, the core pressure is also set at a value consistent with cold (68 degrees Fahrenheit (ºF)) conditions. No sensitivity calculation was performed for the core pressure, but this was not necessary since the coolant density will not increase relative to cold conditions during startup. **[**

]. Therefore, the NRC staff considers use of a 68 ºF value, with the factors discussed above accounted for through the uncertainty in Doppler reactivity feedback, to be acceptable.

Sensitivity studies were performed to determine reasonable bounding values for the other initial conditions based on the assumption that the limiting conditions will occur at CZP conditions during startup. One sensitivity study that seemed to result in mixed results was the study performed on the initial core flow. The CRDA TR recommends use of a core flow of 10 percent based on the fact that during startup, the recirculation pumps will be operating at minimum speed, providing at least that much core flow. This is an acceptable justification for the purpose of bounding the prompt enthalpy rise, since the prompt enthalpy rise does not increase significantly for values above 10 percent of rated flow. However, this justification does not apply to the total enthalpy, which may exhibit large increases for higher core flows. In the response to RAI-9 (Reference 8), AREVA indicated that operating the recirculation pumps at higher speeds would generally involve the addition of heat, increasing the initial coolant temperature. The reduction in the total enthalpy rise resulting from this increase in initial coolant temperature is expected to be larger than the increase in the total enthalpy rise due to an increase in initial core flow. An additional step was added to the evaluation procedure to confirm that **[**

]. However, should this check fail, AREVA will need to perform a plant-specific evaluation to demonstrate that they have identified the limiting initial conditions for the CRDA analysis. A condition is included in Section 5 which clarifies the NRC staff's expectations in this respect.

The CRDA TR provides a description of the expected process by which a plant would be expected to determine the control rod patterns used to analyze the CRDA event. The maximum number of inoperable rods allowed at the plant being analyzed are assumed to occur **[**

], placed as close together as allowed by plant TS requirements. The CRDA TR provides an example rod pattern,

which also demonstrates that **[**
]. This kind of distribution for inoperable rods is sufficient to ensure that the radial power tilt is maximized for **[** **]**, but the exact selection of inoperable rod positions may not be sufficient to bound the potential changes in local neutronic coupling resulting from withdrawal of specific rods. Alternative inoperable rod patterns may be needed to ensure that rod drops are evaluated using a rod pattern that maximizes the reactivity addition resulting from individual rod drops. Specific distributions of withdrawn rods may result in local distributions of uncontrolled locations where a CRDA event would lead to close neutronic coupling of a significant number of fuel assemblies in adjacent uncontrolled cells. Typical quarter core symmetry and operating strategies result in relatively small rod worth variations for quadrant symmetric control rods. Therefore, the NRC staff is including a condition in Section 5.0 which requires evaluation of alternate distributions of inoperable rods, as needed, to ensure that the CRDA evaluations will include consideration of inoperable-rod distributions that maximize the change in face- and/or diagonally-adjacent withdrawn control rods as a result of withdrawal of the candidate rod, for at least one member of each quadrant symmetric control rod group. This is expected to apply primarily to situations where candidate rod locations are near the half-core boundary that defines the distribution of inoperable rods, where a candidate rod may be near an inoperable rod location.

Candidate rods are evaluated as if they are the first rod withdrawn within the given group, and **[** **]** (Section 4.2.2.2.3 discusses the selection of candidate rods). **[**

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For typical control rod groupings utilized by BWR/2-6 plants in the US, the first rod withdrawn **[** **]** results in a significant change in the local neutronic coupling between fuel assemblies, which manifests as a significant increase in reactivity. Subsequent rod withdrawals generally lead to less significant increases in local neutronic coupling, or similar changes in local neutronic coupling elsewhere in the core, so the change in reactivity is more modest. However, there may still be local neutronic characteristics for individual rods that would increase the severity of the consequences from a CRDA. Therefore, the CRDA TR states that analyzing rod drops as if they occur as the first rod pulled **[** **]** will be sufficient to bound all subsequent groups. During a normal startup sequence, removal of the first rod **[** **]** may not lead to a significant change in the core conditions because the reactor is not near criticality. When the CRDA event is analyzed in the manner described in the CRDA TR, MB2-K will normalize the eigenvalue to criticality at the beginning of the CRDA transient. The NRC staff finds this to be an acceptable assumption for the initial rod configuration, given that the reactor will be treated as if it is initially critical, and each candidate rod will be individually analyzed under conditions where the positive reactivity addition is conservatively maximized.

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Based on the above discussion, the NRC staff has evaluated the recommendations in the CRDA for plant parameters to be utilized in the analysis of the CRDA event and found them to be acceptable, with two conditions (as incorporated in Section 5.0).

4.2.2.2.3 CZP CRDA Scenario: Control Rod Selection

The potential number of scenarios for the CRDA event is very high due to the number of rods that could drop and the number of core exposures that the plant may attempt to start up from. In order to reduce the number of CRDA scenarios for explicit analysis to a manageable number, certain criteria are used to screen out scenarios that are not likely to be limiting. The CRDA TR explains how candidate rods are to be selected at specified core exposures for further evaluation based on a series of selection criteria.

First, the k-effective values at the ends of each withdrawal group for all allowed withdrawal sequences are evaluated to determine when criticality is expected to occur. In determining which groups may contain the critical rod pull, **[**

]. AREVA justifies the exclusion of rod pulls that occur beyond criticality based on the fact that the consequences due to the CRDA event are much less severe at the higher temperatures due to nuclear heating. The NRC staff agrees that the impact of the CRDA event for rod pulls in the startup range at higher core temperatures would be bounded by rod pulls at lower core temperatures (see discussion in Section 4.2.2.2.2). Therefore, the only rods considered as possible candidates are those included in the rod groups expected to contain the critical rod pull, **[**

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For each group identified in this way, the highest worth rod plus **[**

]

are flagged for further evaluation. As per the AREVA response to RAI-5 (Reference 8), this is based on the fact that the minimum effective delayed neutron fraction for BWRs is **[**

]. This is consistent with results from prior analyses and from the evaluation of lower worth rods in the reference analysis shown in the CRDA TR.

At CZP conditions, all fuel rods will have the same initial enthalpy, so the total enthalpy will be directly related to the total enthalpy rise. This would be expected to be maximized by larger positive reactivity additions. Burnup of the fuel plays a role due to the change in gap thermal conductivity. However, AREVA documented sensitivity studies based on rod power/exposure histories that demonstrated that this impact is relatively small. **[**

]. Therefore, with respect to the high temperature failure acceptance criteria, evaluation of the highest worth rod for each core statepoint of interest would be sufficient to verify that no failures would be expected, though additional rods may need evaluation if the high temperature failure acceptance criteria are not met for the highest worth rod. A follow-up response to RAI-3 describes a methodology to determine the rod internal pressure for use in applying the more restrictive high temperature failure acceptance criteria in DG-1327 (as discussed in Section 4.2.2.2.4). In order to ensure that appropriate candidate rods are identified for evaluation against the high temperature failure criteria, a condition was included in Section 5.0 to ensure that additional candidate rods are considered as necessary.

The PCMI failure acceptance criteria are dependent on the hydrogen content of the fuel cladding, which is a function of burnup (among other things). As a result, fuel failures may occur in higher burnup fuel near lower worth control rod drops near the core periphery. Revision 0 of the CRDA TR provided criteria based on **[ ],** but the correlation between the **[ ]** and the probability of PCMI failures was not clear. The NRC staff asked AREVA to provide further justification for the proposed criteria for selection of candidate rods for further evaluation. In response to RAI-5 (Reference 8), AREVA provided a new procedure to identify candidate rods for evaluation based on their **]**.

The approach being proposed by AREVA is based on the assumption that there will be a reasonably strong correlation between **[ ]**. For the initial application at a plant group or fuel loading type, rod drops will be evaluated for a range of rod worths. **[ ]**, and this information is used to establish an **[ ]** (evaluation boundary) **[ ]**. Once this threshold has been established, then it can be used to screen out rods that are not expected to experience failures. The screen-out process works by determining the minimum failure threshold for **[ ]**. If this minimum PCMI failure threshold **[ ]** the evaluation boundary **[ ]**, then the rod of interest would not be expected to result in any PCMI failures. If minimum PCMI failure thresholds are identified that are **[ ]** than the evaluation boundary, then that rod drop will be explicitly analyzed and the PCMI acceptance criteria are evaluated based on the calculated rod enthalpies and rod-specific hydrogen contents.

[

], the NRC staff is including a condition in Section 5.0 which specifies local core parameters that must be bounded by the data set used to establish the evaluation boundary. For uranium dioxide fuel, the local core parameters selected to capture the relevant influences are: design pin peaking factors **[** **]**, fuel assembly design **[** **]**, proximity to the core periphery **[** **]**, and average burnup for the 16 fuel assemblies surrounding the rod of interest **[**

]. However, ensuring that the data set is bounded by the evaluation boundary, along with the step in the AREVA procedure **[**

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The identification of candidate rods for evaluation is performed at three unique core exposures corresponding to beginning of cycle (BOC), peak hot excess reactivity, and end of cycle (EOC). While intermediate exposures are not explicitly evaluated, these three core exposures capture the most significant shifts in power/exposure distribution corresponding to gadolinia burnout and the axial shift in power from the bottom to the top of the core. In order to ensure that the intermediate exposures are adequately addressed due to the burnup dependent nature of the PCMI failure thresholds, **[**

]. The NRC staff finds this to be a reasonable approach to ensure that intermediate exposures do not lead to a combination of minimum failure thresholds and rod worths that proves to be significantly more limiting than the evaluation exposures.

As discussed in this section, AREVA provided adequate justification for reasonable assurance that the limiting results from analysis of the CRDA event will be associated with one of the candidate rods identified using the procedure as described in the CRDA TR (as modified by the response to RAI-5). Therefore, the NRC staff finds the candidate rod selection process to be acceptable for the purpose of limiting the scope of the CRDA analysis to those rods expected to be limiting. The coupled AURORA-B code system is then utilized to explicitly calculate enthalpy values for use in evaluating the acceptance criteria for the candidate rods.

4.2.2.2.4 CZP CRDA Scenario: Evaluation Against Acceptance Criteria

The CRDA TR provides a discussion of how to use the analysis results in determining whether the RIA acceptance criteria are met. The at-power CRDA event (see Section 4.2.2.2.1) and RPV pressure response (see Section 4.2.2.1.1) were **[**

]. In

general, the evaluation procedure is applicable to acceptance criteria from SRP 4.2 Appendix B and DG-1327, with the exception that the fuel rod pressure is included in the evaluation of the high temperature failure threshold from DG-1327. The primary acceptance criteria are associated with radiological consequences and core coolability.

In order to evaluate the radiological consequences, the number of fuel rod failures must be determined. This is performed by comparing the calculated prompt enthalpy rises and total enthalpies for each rod in the fuel assemblies surrounding the analyzed control rod drop scenarios against the acceptance criteria for PCMI failure and high temperature failure, respectively. Inventory release fractions are then determined by combining steady state gap release fractions (as determined by the applicable plant-specific licensing basis, usually derived from Regulatory Guide 1.183 or 1.195) and transient release fractions for the CRDA event (as calculated based on the calculated nodal enthalpies for the failed rods and the applicable regulatory guidance, SRP 4.2 Appendix B or the final regulatory guide based on DG-1327). The inventory releases can then be used to demonstrate that the CRDA releases are non-limiting, typically by comparison to the inputs used in the bounding radiological analysis of record. For example, in the reference analysis provided in the CRDA TR, a multiplier is applied to the number of rod failures to account for the difference in calculated release fractions for the CRDA event relative to the release fractions used in the plant-specific radiological consequence analysis of record. The adjusted number of rod failures can then be directly compared to the assumed number of rod failures in the radiological consequence analysis of record to verify that the analysis of record is bounding.

The PCMI failure acceptance criteria are dependent on the prompt enthalpy rise and the hydrogen content of the cladding, which can both vary independently from fuel rod to fuel rod. Therefore, the CRDA TR describes a method to determine pin-specific enthalpy rises based on the maximum and average enthalpies as determined by S-RELAP5. The initial approach presented in Revision 0 of the CRDA was based on an assumption of linearity in pin powers that did not seem to bound all possible pin power distributions. In response to RAI-6 (Reference 8), AREVA provided a different approach that determined pin-specific enthalpy rises via **[**

]. The CRDA TR provides information supporting a conclusion that the pin peaking factors are essentially constant during the majority of the time that energy is being deposited in the fuel and **[**

]. As a result, the NRC staff agrees that the approach described in the RAI-6 response is appropriate for determining pin-specific prompt enthalpy rises for the purpose of comparison to the PCMI failure acceptance criteria, when correlated with the pin-specific hydrogen contents calculated by RODEX4 for the depletion history of each pin. **[**

]. The revised approach is expected to be incorporated in the final approved version of the CRDA TR.

The same approach is used to determine pin-specific total enthalpies for use in applying the high temperature failure acceptance criteria. In Revision 0 of the CRDA TR, the **[**

] was used for the high temperature failure acceptance criterion, **[**

]. DG-1327 contains more restrictive high

temperature failure acceptance criteria for higher internal rod pressures, so AREVA submitted additional information in their response to RAI-3 (Reference 8) to address the impact of the rod power/burnup history on the internal rod pressure. If the total enthalpy is less than the minimum enthalpy threshold, then no further evaluation will be performed. If the total enthalpy exceeds the minimum enthalpy threshold, then the internal rod pressure will need to be determined. As described by AREVA, this process has two key elements: **[**

] and (2) determination of a transient fission gas release inventory based on guidance from DG-1327 and adjusting the internal rod pressure to account for the addition of the extra fission gas in the gap. When determining the final internal rod pressures, **[**

]. The NRC staff does note that the approved version of the CRDA TR needs to include a placeholder for the final approved regulatory guide in lieu of referring to DG-1327, but this approach would be acceptable for the purpose of evaluating the high temperature failure acceptance criteria.

The final set of acceptance criteria is associated with core coolability. SRP 4.2 Appendix B provides four criteria associated with the following parameters: (1) peak fuel enthalpy, (2) peak fuel temperature, (3) mechanical energy generation, and (4) change in geometry of fuel rod/pellet. DG-1327 contains two criteria that are similar to the first two criteria in SRP 4.2 Appendix B. These criteria are evaluated in the CRDA TR by direct comparison to the total enthalpies and peak fuel temperatures determined using the CRDA analysis methodology. The CRDA TR presents information to support an assertion that fuel dispersion will not occur below the peak fuel enthalpy limit given in criteria (1) as long as **[**

] in combination with criteria (1). The CRDA TR also provides a discussion that adequate cooling will be maintained **[**
]. Therefore, ensuring that there is no gross change in fuel assembly geometry through confirming that the first three criteria are met, is sufficient to ensure that (4) is met. DG-1327 supports the conclusion that the first two criteria are usually sufficient to ensure that a coolable geometry is maintained.

As a result of the above discussion, the NRC staff finds that the proposed procedure is acceptable to confirm that the acceptance criteria for the CRDA event are met. This review considered the acceptance criteria for both SRP 4.2 Appendix B and the draft acceptance criteria in DG-1327. At this time, DG-1327 has not been finalized and may be subject to change, but as long as the basis for the above findings continue to remain valid in the final regulatory guidance, then the methodology outlined in the CRDA TR should remain valid for use in demonstrating that NRC requirements are met.

4.3 Code Integral Assessment

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

integral effects tests. 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 four CCDs within the AURORA-B EM (S-RELAP5, MICROBURN-B2, MB2-K, and RODEX4) have been assessed against integral and separate effect data and found to be acceptable for performing safety analyses during the review and approval of their individual TRs as well as the base AURORA-B TR. As a result, the majority of this section of the report will focus on the specific assessments that were performed to demonstrate that the overall EM provides adequate predictions of the phenomena of interest for the CRDA event.

Several of the assessments performed to support previously approved TRs are referenced to support the ability of the AURORA-B EM to analyze the CRDA event. While these assessments were not reviewed by the NRC with the CRDA event in mind, the NRC did find them sufficient to demonstrate that the specific phenomena were captured accurately by the AURORA-B EM. The NRC staff determined that there was nothing specific to the CRDA event that would invalidate the prior NRC acceptance of the assessments to validate the AURORA-B EM's ability to model the phenomena, as described below.

- Bundle void tests – supports void distribution calculations (for at-power CRDA)
- CASMO4/MICROBURN-B2 qualification – supports calculation of neutronic response
- RODEX4 qualification – supports rod history effect calculations
- Numeric benchmarking for neutronic transients including CRDAs – supports code stability and fidelity
- Pressure drop and critical power tests – supports applicability of CPR calculations (for at-power CRDA)
- Peach Bottom turbine trip test – supports ability of AURORA-B EM to predict neutronic/thermal hydraulic coupled feedback (for at-power CRDA)

Additional model integral test assessments were provided to support the ability of the AURORA-B EM to accurately evaluate the CRDA event for startup conditions based on tests performed at the Special Power Excursion Reactor Test III (SPERT III) reactor. These tests provide a valuable assessment of the AURORA-B EM's ability to accurately capture the Doppler reactivity feedback, since the SPERT III reactor does not include moderator voiding and the power pulses are short enough to ensure that no significant heat transfer to the moderator occurs prior to the mitigation of the prompt power excursion due to Doppler reactivity feedback. As such, this assessment provides confidence that the AURORA-B EM will accurately model

the Doppler reactivity feedback in the absence of other reactivity feedback mechanisms (such as void feedback, as captured by the Peach Bottom turbine trip tests). The assessment shows that AURORA-B generally predicted the prompt power pulse from the SPERT III experiments well for a variety of conditions. **[**

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The NRC staff reviewed the previous assessments performed to support the AURORA-B EM and its constituent CCDs, and determined that they were applicable to demonstrate that specific phenomena relevant to the CRDA event are appropriately assessed. The one significant assessment gap, related to determining the Doppler reactivity feedback in the absence of any other significant reactivity feedback mechanisms, was filled by assessing AURORA-B against data from SPERT III tests of rod ejection accidents. Therefore, the NRC staff has determined that the AURORA-B EM has been satisfactorily assessed for its ability to model the relevant phenomena for the CRDA event.

4.4 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 indicates 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 consistent with the results of the accident scenario identification process.

The CRDA TR describes evaluations performed for each of the individual phenomena identified in the CRDA PIRT. In general, the evaluations led to one of the following conclusions:

1. The dominant parameters affecting the relevant phenomenon are set to bounding values, therefore, no uncertainty needs to be considered (example: minimum TS scram times associated with reactor trip reactivity).

2. Studies were performed to establish the percentage increase in the calculated enthalpy rise that would be necessary to bound the limiting end of the uncertainty (based on relevant references) in the dominant parameters affecting the relevant phenomenon (example: uncertainty in evaluation of **[** **]**).

3. Studies were performed using a reasonable range for the target parameters which may not represent a limiting range, but was sufficient to demonstrate a reasonable uncertainty (example: direct heating of moderator).

4. An uncertainty was assigned based on uncertainties determined as part of the qualification of individual CCDs in the AURORA-B EM (example: power distribution).

The bounding values associated with conclusion (1) have been discussed earlier in this SE, so no further considerations are necessary. The references cited to support conclusion (2) were reviewed to ensure that the bases for the ranges of values used for the parameters of interest continue to be valid for this application. The studies supporting conclusion (3) were reviewed to determine whether the uncertainty was supported in light of the range analyzed. For the most part, these studies demonstrated that the impact on enthalpy was minimal over a large

reasonable range of values for the parameter of interest, so analysis of a more extreme range of values would not change the results significantly. One exception was the study performed on the pellet radial power distribution, which drives the temperatures used to calculate the effective Doppler temperature. The NRC staff's considerations in determining the appropriate weighting for the effective Doppler temperature are documented in Section 4.2.2.1.4, but this sensitivity study includes a number of overly conservative weighting schemes. The uncertainty selected to represent the pellet radial power distribution bounds all evaluated schemes except for an unrealistically conservative weighting that **[** **]**. Finally, the uncertainties supported by conclusion (4) were evaluated to verify that they are applicable to the overall AURORA-B EM in the CRDA analysis, based on the modeling and nodalization discussed in the CRDA TR. This primarily concerns uncertainties derived from the MB2-K and RODEX4 CCDs. Since the core model used in the CRDA evaluation has **[** **]**, and the underlying phenomena are consistent with the prior qualification of these CCDs, the application of these uncertainties to the CRDA analysis is acceptable.

The following table summarizes the parameters evaluated, how their enthalpy uncertainties were determined, and the NRC staff's assessment of the acceptability of the approach used for the purpose of determining the expected impact on the limiting enthalpy rises for the CRDA analysis. AREVA elected to perform an assessment of all phenomena from the PIRT developed for the CRDA event, so many of these assessments primarily serve the purpose of confirming the low importance of the phenomenon of interest to the enthalpy FoM for the CRDA event.

Parameter	AREVA Analysis	NRC Assessment
Control rod worth	The rod worth was plotted against the enthalpy rise for a range of exposures and temperatures **[** **]**.	The data presented by AREVA suggests that **[** **]**. As discussed in Section 4.2.2.1.5, the variation in design control rod worth is to be addressed through explicit modeling rather than as an uncertainty (Condition 26 in Section 5.0).

Parameter	AREVA Analysis	NRC Assessment
Moderator feedback	The uncertainty was determined to be minimal based on a **[ ]** change in active channel moderator density feedback and **[ ]** bypass channel moderator density feedback.	The range of feedback variation analyzed is fairly large relative the expected change in moderator density for a CZP CRDA event due to the fact that it would take significant energy to reach saturation conditions. Therefore, the NRC staff finds this analysis to be sufficient to demonstrate that the impact on the enthalpy uncertainty is minimal.
Fuel temperature feedback	The uncertainty was determined based on a **[ ]** change in Doppler feedback. This range was selected **[ ]**.	The NRC staff noted that **[ ]**. This is an appropriate uncertainty range for use of the underlying resonance integrals determined by CASMO-4, and is consistent with the NRC approval of the base AURORA-B TR. Other code uncertainties resulting from the MB2-K computational scheme to determine the overall reactivity change are accounted for in the validation of MB2-K.
Delayed neutron fraction	The uncertainty was determined based on a **[ ]** change in the delayed neutron fraction value. This value was selected to match the value used in the NRC approved EPR rod ejection accident methodology TR.	The base AURORA-B TR used a value of **[ ]** instead of **[ ]**. As discussed later in this section, the total uncertainty proposed by AREVA contains sufficient inherent conservatism to accommodate this modest increase in the delayed neutron fraction uncertainty.
Scram reactivity	Minimum technical specification scram times are used, so no uncertainty is applied.	Use of the minimum allowed scram times is conservative, and the total worth of the control rods is well beyond that required to result in subcriticality. Therefore, the selected parameters are bounding and no uncertainties are necessary.

Parameter	AREVA Analysis	NRC Assessment
Fuel cycle design	The neutronic properties that are dependent on the fuel cycle design are explicitly captured in the CRDA analysis process or via other uncertainties, so no uncertainty is applied.	The overall neutronic parameter affected by the fuel cycle design is the fuel reactivity at specific statepoints due to the core design and depletion. **[** **]**. Therefore, the fuel cycle design is appropriately addressed through other uncertainties and analysis process requirements.
Heat resistances	The uncertainty was determined to be **[** **]** based on a **[** **]** change in gap width and ± 20 percent change in the fuel heat transfer coefficient.	The analyzed range of fuel heat transfer coefficients is much larger than the **[** **]** uncertainty accepted in the base AURORA-B TR, while the gap width variation is about **[** **]** considered in the base AURORA-B TR. The prompt enthalpy rise is not affected (due to the adiabatic nature of this phase) and the impact on the total enthalpy is based on two competing phenomena: heat transfer from the fuel pellet to the coolant (smaller gap = more rapid heat transfer to the coolant = lower total enthalpy) and the temperature used to compute the Doppler reactivity feedback (smaller gap = lower pellet surface temperature = weaker Doppler reactivity feedback = larger power pulse = higher total enthalpy). **[** **]**, so the NRC agrees that it is reasonable to expect that the impact due to uncertainties in the gap width will not be significant. However, this conclusion would need to be verified for significantly different fuel rod geometries and manufacturing tolerances. A condition was added to Section 5.0 to ensure that this will be the case.

Parameter	AREVA Analysis	NRC Assessment
Heat transfer	For the CZP CRDA event, the power pulse is terminated prior to any significant heat transfer from the fuel to the coolant, so the effect on reactivity feedback is insignificant. From a heat transfer point of view, use of steady state correlations for transient conditions has been shown to be conservative. Therefore, no uncertainty is applied.	The NRC staff agrees with the assessment of the impact of fuel-to-coolant heat transfer on the Doppler reactivity feedback. The SE approving the base AURORA-B TR agrees on a similar conclusion regarding the conservatism of steady-state heat transfer correlations. Therefore, no uncertainties are needed.
Heat capacities	The heat capacities for the fuel and cladding are established based on the RODEX material properties, which are consistent with those from **[** **]**, while the heat capacity of the cladding is not important for the CRDA event. Therefore, no uncertainties are applied.	**[** **]** contains an assessment by the NRC of the **[** **]**. The heat capacities for uranium dioxide fuel, especially at the relatively low fuel temperatures expected for the CZP CRDA event, show an excellent match to the experimental data. The cladding heat capacity does impact the total enthalpy and is shown in **[** **]** as having a potential uncertainty of a few percent. However, the impact due to a change in cladding heat capacity of a few percentage, for a few seconds, would not result in a significant change in total enthalpy because residual heat deposition and redistribution is the primary driver for the total enthalpy. Therefore, the NRC staff found it acceptable to consider the impact of uncertainties in the heat capacities of the fuel and cladding to be minimal.
Energy deposition	The uncertainty was determined to be **[** **]** based on a ± 20 percent change in the energy deposition fractions to the coolant.	The direct deposition of energy in the coolant is characterized by the moderator density and the gamma ray/neutron flux entering the coolant. The former is bounded by the initial conditions, and the latter is a direct function of the power level. As such, this uncertainty is associated with the adequacy of the energy partitioning. This range is judged to be reasonably large, given the acceptable performance of AURORA-B for several assessments in the base AURORA-B TR that are driven largely by moderator feedback. Therefore, the NRC staff concludes that the results of the study support AREVA's conclusion.

Parameter	AREVA Analysis	NRC Assessment
Pellet radial power distribution	This parameter represents the uncertainty in the weighting used to determine the Doppler effective temperature passed to MB2-K for the reactivity calculation. A series of different perturbations of the weighting factors are applied, and a **[ ]** uncertainty was selected **[ ]**.	Section 4.2.2.1.4 discusses the NRC staff's evaluation of the weighting factors selected for use in the analysis. The NRC staff found the weighting to be slightly conservative for the CZP CRDA event, so applying a further uncertainty is not necessary. Nevertheless, the inclusion of this uncertainty provides some additional assurance against an unexpected interplay between the pellet temperature profile and the Doppler reactivity feedback, which is very unlikely given the short time to peak enthalpy values relative to the time constant for heat transfer.
Rod peaking factors	The local peaking factor uncertainty from MB2-K is applied directly as an uncertainty to the enthalpy rise.	The enthalpy rise would be expected to be more or less proportional to the power generation. In fact, higher temperatures in the peak power heat structure would accelerate heat transfer to the coolant, thus reducing the total enthalpy slightly. The peak power rod does not contribute to the Doppler reactivity feedback calculation. As a result, the NRC staff finds that direct application of the local peaking factor uncertainty to the enthalpy is acceptable.
Cladding hydrogen content	The cladding hydrogen content is not used as a relevant parameter in the CRDA analysis. Rather, the value is used to determine the fuel cladding failure threshold for PCMI. Based on NRC guidance in the application of the current acceptance criteria for PCMI failure, a best-estimate model is sufficient. As a result, no uncertainties are necessary.	The NRC staff agrees with the AREVA characterization of the discussion in the memo presenting the technical justification for the current SRP 4.2 Appendix B acceptance criteria. The relatively low hydrogen content as a result of pickup during irradiation is not expected to affect the heat transfer properties of the cladding significantly enough to affect the CRDA analysis results.

Parameter	AREVA Analysis	NRC Assessment
Power distribution	The initial core power is conservative, and the uncertainty due to core power distribution from MB2-K is applied directly to the enthalpy.	The results from sensitivity studies show that the recommended initial core power is conservative for the CRDA analyses. The peak rod enthalpy rise will not be proportional with an increase in fuel assembly reactivity, due to the non-linearity in Doppler reactivity feedback as a function of fuel temperature. However, the proposed uncertainty of **[** **]** is much higher than **[** **]**. The NRC staff expects this increase to be sufficient to account for the difference in expected enthalpy behavior compared to fuel reactivity increase.
Coolant initial conditions	All initial conditions were selected to conservatively bound the possible range of operating conditions, so no uncertainty is necessary.	The NRC staff considerations associated with the initial conditions are documented in Section 4.2.2.2.2. The NRC staff agrees that the initial conditions are conservative, therefore, no uncertainty is necessary.
Fuel rod internal pressure	The fuel rod internal pressure is primarily used in the application of the high temperature failure threshold, which is a function of the difference between fuel rod internal pressure and the system pressure. The response to RAI-3 describes a conservative approach for determining this internal pressure, and also provides information demonstrating that the impact of fission gas buildup on the gap conductivity does not result in a significant impact to enthalpy. Therefore, no uncertainty is necessary.	The NRC staff agrees that the approach for determining the transient fission gas release and combining it with a pre-transient fission gas distribution **[** **]**, is conservative. **[** **]**. The response to RAI-3 also demonstrates that the gap conductivity change due to significant changes in fission gas migration to the gap **[** **]** would not have a large effect on the enthalpy. This effect is already captured in the fuel rod models in AURORA-B used to analyze the CRDA event, and any uncertainty would be even smaller. Therefore, the NRC staff finds it acceptable to treat the uncertainty associated with fuel rod internal pressure as minimal.

To apply these uncertainties, the CRDA TR combines all uncertainties using **[**

]. This is sufficient to accommodate an increase of **[** **]** in the delayed neutron fraction uncertainty plus control rod manufacturing tolerances of about 2.5 percent, which is much higher than the NRC staff would expect from normal manufacturing processes.

In order to obtain greater confidence in the appropriateness of the recommended value for the total uncertainty, the NRC staff asked for further validation of the recommended uncertainty. In response to RAI-10 (Reference 8), AREVA provided the results of uncertainty analyses performed using the same **[** **]** process used to determine the uncertainties for the base AURORA-B TR. **[**

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As a result of the above discussion, the NRC staff considered the proposed approach for determining appropriate uncertainties to use **[** **]** for the calculated enthalpy rises used in applying the acceptance criteria for the CRDA event. The approaches used to evaluate specific uncertainties are discussed in the table beginning on page 30 of this SE, and the subsequent discussion addresses **[**

]. Therefore, the NRC staff concludes that the **[** **]** provided in Section 8.7.4 of the CRDA TR to account for uncertainties is appropriate. Since this assessment is based on offsetting considerations based on the information submitted, the NRC staff is including a condition in Section 5.0 requiring NRC approval for use of a reduced total uncertainty.

5.0 LIMITATIONS AND CONDITIONS

As discussed previously in this report, limitations and conditions have been applied to use of the AURORA-B EM as part of its initial approval for application to AOOs. In some cases, the CRDA TR provided justification that a limitation or condition does not apply for the specific application

described therein. In other cases, new limitations or conditions were added regarding the use of AURORA-B to analyze the CRDA event.

The summary of all limitations and conditions applicable to the use of the AURORA-B EM to perform CRDA analyses for BWR/2-6, in a manner consistent with the base AURORA-B TR, are stated below. Captions in italics indicate the source. The numbering scheme uses the base AURORA-B TR condition numbers (1-26) that are still applicable, supplemented by those that need to be applied to CRDA analyses. Note that the summary below is only intended to list conditions specific to the use of AURORA-B for analysis of the CRDA event. Therefore, all limitations and conditions from the base AURORA-B TR that have been adequately addressed in the CRDA analysis methodology, or are not relevant to the primary phenomena in the CRDA event, have been omitted. Some of the omitted limitations and conditions from the base AURORA-B TR may remain valid for other transients, so the base AURORA-B TR should be used as the starting point for limitations and conditions on the general use of the AURORA-B EM.

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 license amendment requests (LARs). In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2. *(This is Condition 1 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

14. The scope of the NRC staff's approval of AURORA-B does not include the ABWR design. *(This is Condition 14 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

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. *(This is a revised version of Condition 20 of the SE for the base AURORA-B TR, rewritten to be specific to the CRDA application. It remains applicable to CRDA analyses for BWRs/2-6.)*

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. *(This is Condition 21 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

22. As discussed in Section 3.3.1.5 and Section 4.0 of Reference 3 (the SE for the base AURORA-B TR), the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EM. 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". 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. *(This is Condition 22 of the SE for the base AURORA-B TR. It remains applicable to at-power CRDA analyses for BWRs/2-6.)*

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. *(This is Condition 23 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

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

    ] areas discussed in Section 4.0 of Reference 3 (the SE for the base AURORA-B TR) without prior NRC approval. *(This is Condition 24 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

25. The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of Reference 3 (the SE for the base AURORA-B TR). *(This is Condition 25 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

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

    ] areas discussed in Section 4.0 of Reference 3 (the SE for the base AURORA-B TR). *(This is Condition 26 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

27. The control rod model at each location in the core used for CRDA analyses with the AURORA-B EM shall use a control rod geometry and composition that is verified to bound the control rod worth for the physical control rod used in that location, for all axial elevations. *(This is a new condition from this report.)*

28. Licensees utilizing AURORA-B to perform CRDA analyses using the methodology described in this TR shall confirm that the recommended maximum rod velocity of 3.11 ft/s is conservative for their control rods. *(This is a new condition from this report.)*

29. If the check to verify that the total enthalpy is limiting at 10 percent core flow CZP conditions by **[**

    ] fails, AREVA shall perform a more comprehensive evaluation to verify that they have identified the limiting initial conditions for that plant. This evaluation should consider a range of flow values and corresponding plant-specific minimum temperatures that is sufficiently broad to clearly identify the combination of initial conditions which maximizes the total enthalpy for the limiting rod. *(This is a new condition from this report.)*

30. When individual control rods are evaluated using the CRDA analysis methodology, if necessary, alternate distributions of inoperable rods should be utilized to ensure inclusion of at least one evaluation within each group of 4 quadrant symmetric control rods that maximizes the change in face- and/or diagonally-adjacent uncontrolled cells as a result of the candidate control rod withdrawal. *(This is a new condition from this report.)*

31. The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worths must be verified to bound the following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest. *(This is a new condition from this report.)*

32. If the highest worth rod at a given core statepoint results in a total enthalpy that is higher than the minimum high temperature failure threshold (i.e., lowest threshold for all rod internal pressures), additional rods must be considered for evaluation. This may be done by evaluating the next highest worth rods at the core statepoint of interest until the minimum high temperature failure threshold is met, or by using an approach analogous to the evaluation boundary curve used for the PCMI failure threshold (as subject to condition 29). *(This is a new condition from this report.)*

33. If the methodology described in ANP-10333 is used to analyze the CRDA event with a fuel assembly design that has a different fuel rod geometry and/or manufacturing tolerances than the one used as a basis for the sensitivity study on gap width, the sensitivity study shall be repeated for the new fuel assembly design, using bounding values consistent with the uncertainty range for **[** **]** limiting increase in the peak total enthalpy, the total uncertainty shall be increased accordingly for total enthalpies calculated based on the new fuel assembly design. *(This is a new condition from this report.)*

34. The uncertainty designated in the CRDA TR of **[** **]** for the enthalpy rises calculated using the CRDA analysis methodology may not be reduced without prior NRC approval. *(This is a new condition from this report.)*

6.0 CONCLUSIONS

In the CRDA TR, AREVA presented new models and methods to extend the applicability of the AURORA-B EM for evaluation of the CRDA event. This SE addresses the application of the CRDA TR only to BWRs/2-6; the NRC is not currently reviewing the AURORA-B EM for applicability to ABWRs. The following conclusions are provided here in summary as they apply to BWR/2-6 submittals.

The CRDA TR presents a description of the CRDA event, the relevant phenomena, the applicable FoMs, and a ranking of the phenomena for any applicable FoMs. In some cases, the base AURORA-B TR was cited as the source for the assessment of the AURORA-B EM concerning specific FoMs, so the NRC staff confirmed that the prior assessment would be applicable to the CRDA event. In one case, for the fuel rod enthalpy, the phenomena ranking was verified against prior precedents (for CRDAs and RIAs in general) as well as the NRC staff's technical understanding of the relevant phenomena.

The application of the AURORA-B EM for the purpose of analyzing CRDA events involved the addition of several enhancements to the AURORA-B CCDs, as described in the CRDA TR: pin power reconstruction at cold conditions, peak rod heat structure models, a hydrogen pick-up model, and additional information interchange between the CCDs. The NRC reviewed the new models as well as the models previously approved as part of the base AURORA-B TR, and found them to be acceptable for use in analyzing the CRDA event.

The CRDA TR also presents a procedure for analysis of the CRDA event, which includes modeling guidance and evaluation guidance. This formed the bulk of the NRC staff review of the CRDA TR, and included a review for acceptability of model nodalization guidance, modeling input specifications, recommended initial conditions, control rod evaluation procedure, and acceptance criteria. **[**

]. As a result of the NRC staff considerations, conditions (27) through (32) were identified as being necessary to ensure that relevant scenarios are bounded by the proposed CRDA analysis methodology.

In order to demonstrate the capability of the AURORA-B EM to analyze the CRDA event, assessments were made against numeric benchmarks, separate effects tests, and integral tests. In most cases, these assessments were already performed as part of the base AURORA-B TR. One additional assessment was added, for tests performed at the SPERT III reactor to simulate rapid rod withdrawal scenarios. The data from this assessment was valuable in that it provided confidence that the AURORA-B EM could accurately predict the Doppler-only component of the reactivity feedback. The prior assessments were primarily numeric benchmarks or tests which involved significant moderator reactivity feedback, which is not the case for the CZP CRDA event.

Finally, the CRDA TR presented an analysis of the uncertainties associated with the proposed CRDA analysis methodology. The approach used by AREVA is not one that would be appropriate for general application. However, the NRC staff considered the applicability of the various assumptions that must be valid for this approach as well as inherent conservatisms in the application to the CRDA event, and requested some further validation based on more robust statistical analysis methodologies. Based on the staff considerations and information provided by AREVA, the NRC staff determined that the recommended uncertainty to be applied **[**

] the enthalpy rise values calculated in the CRDA analysis was reasonable. Since the total uncertainty was accepted based on consideration of offsetting impacts, condition (34) was added to disallow reduction of the uncertainty without prior NRC approval. In addition, the uncertainty associated with the gap width would need to be re-evaluated for different fuel rod geometries and manufacturing tolerances. Condition (33) was included to ensure that this happens whenever AREVA makes these kinds of changes.

In summary, the NRC staff finds that the assessment of the AURORA-B EM, as described in the CRDA TR and responses to NRC staff RAIs, adequately demonstrates that AURORA-B is suitable to analyze the CRDA event by demonstrating acceptable performance in each of the highly ranked phenomena. In addition, the NRC staff finds that the procedure described in the CRDA TR for performance of the CRDA analyses provides appropriate guidance to identify and analyze the limiting scenarios. Since the CRDA event is relatively insensitive to thermal hydraulic performance of the plant and appropriate guidance has been presented to address the relevant factors, the NRC approval of the AURORA-B EM for CRDA analysis purposes extends to all forced circulation BWR plant types (BWR/2 through BWR/6 plants) for operating conditions up to and including EPU conditions with expanded power and flow windows. Additionally, NRC approval of the AURORA-B EM for analysis of the CRDA event is contingent on adherence to the limitations and conditions set forth in Section 5.0.

7.0 REFERENCES

1. ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," March 2014. (ADAMS Accession Nos. ML14098A332/ML14098A333 (Publicly Available/Non-Publicly Available)).

2. ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," December 2009. (ADAMS Accession No. ML100040158 / ML100040159 (Publicly Available/Non-Publicly Available)).

3. Draft Safety Evaluation for ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient And Accident Scenarios," August 2017 (ADAMS Accession No. ML17207A495 (Non-Publicly Available)).

4. NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition." (ADAMS Accession No. ML070810350 (Publicly Available)).

5. Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," November 2016. (ADAMS Accession No. ML16124A200 (Publicly Available)).

6. EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," October 1999. (ADAMS Accession Nos. ML003698495/ML003698553 (Publicly Available/Non-Publicly Available)).

7. Final Safety Evaluation for BAW-10247PA, Revision 0, Supplement 1P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," approved May 17, 2017. (ADAMS Accession Nos. ML17129A006/ML17096A503 (Publicly Available/Non-Publicly Available)).

8. ANP-10333Q1P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Responses to NRC Request for Additional Information," April 2017. (ADAMS Accession Nos. ML17100A171/ML17100A174 (Publicly Available/Non-Publicly Available)).

9. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a.

Attachment: Resolution of Comments

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