0414-12-F04 (Rev. 002, 01/15/2018)

framatome

AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)

ANP-10333NP-A
Revision 0

March 2018

Framatome Inc.

NRC-IC-18-019
T4.12.1
Rec'd 5/21/18 HHE

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

May 10, 2018

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

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT ANP-10333P, REVISION 0, "AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO CONTROL ROD DROP ACCIDENT (CRDA)" (CAC NO. MF3889/EPID: L-2014-TOP-0005)

Dear Mr. Peters:

By letter dated March 31, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14098A331), Framatome Inc. (Framatome) (formerly AREVA Inc.) submitted Topical Report (TR) ANP-10333P, Revision 0, " AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. By letter dated October 25, 2017 (ADAMS Accession No. ML17240A268), an NRC draft safety evaluation (SE) regarding our approval of TR ANP-10333P, Revision 0, was provided for your review and comment. By letter dated December 15, 2017 (ADAMS Accession No. ML17353A223), Framatome provided comments on the draft SE. The NRC staff's disposition of the Framatome comments on the draft SE are discussed in the attachment (ADAMS Accession No. ML18128A004) to the final SE enclosed with this letter.

The NRC staff has found that TR ANP-10333P, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

In accordance with the guidance provided on the NRC website, we request that Framatome publish approved proprietary and non-proprietary versions of TR ANP-10333P, Revision 0, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

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

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

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

Sincerely,

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

Project No. 728
Docket No. 99902041

Enclosure:
Final Safety Evaluation (Non-Proprietary)

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.

Enclosure

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. [ ]

   [ ], 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:* **[**

   ].

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.

---

[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. [

].

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. [

].

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. [

].

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. [

].

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. [

].

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

[

]. 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 (°F)) 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 °F 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). [

].

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.

[

].

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, [

].

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 [ ].

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. [

].

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. [

].

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

Principal Contributor: Scott Krepel, NRR/DSS/SNPB

Date: May 10, 2018

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT ANP-10333P, REVISION 0, "AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO CONTROL ROD DROP ACCIDENT (CRDA)"

FRAMATOME, INC.

PROJECT NO. 728/DOCKET NO. 99902041

This Attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by Framatome Inc. (formerly AREVA Inc.) on the draft safety evaluation for Topical Report ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)."

Page No.	Line(s) No.	Proposed Change and Reason	NRC Response
2	20-25, 27-28, 33-34, 36, 42-44, 47-48	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
2-5	Section 2.0	The Limitations and Conditions listed from the AURORA-B TR SE (ANP-10300P, Revision 0) are not consistent with the final SE Limitations and Conditions. Listed Limitations and Conditions should be updated to reflect final Limitations and Conditions from the AURORA-B TR SE.	The Limitations and Conditions were updated to reflect those from the ANP-10300P final SE.
3	11-15, 18, 20-21, 24-26, 33-37,42-44	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
4	11,13	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
5	1-3, 9-11	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
22	1	Replace "[ ]" with "[ ]" for clarification.	NRC staff agrees with suggested editorial change. Final SE modified accordingly.
22	48	Replace "[ ]" with "[ ]" for clarification.	NRC staff agrees with suggested editorial change. Final SE modified accordingly.

Attachment

Page No.	**Line(s) No.**	**Proposed Change and Reason**	**NRC Response**
37	44	Replace "*different*" with "*reduced*" for consistency.	NRC staff agrees with suggested editorial change. Final SE modified accordingly.
38-40	Section 5.0	The Limitations and Conditions listed from the AURORA-B TR SE (ANP-10300P Revision 0) are not consistent with the final SE Limitations and Conditions. Listed Limitations and Conditions should be updated to reflect final Limitations and Conditions from the AURORA-B TR SE.	The Limitations and Conditions were updated to reflect those from the ANP-10300P final SE.
39	7-9, 18-20	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
40	17-18	Proprietary markups have been added to reflect AURORA-B AOO SE and are highlighted in yellow.	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
41	28	Replace "*modification*" with "*reduction*" for consistency.	NRC staff agrees with suggested editorial change. Final SE modified accordingly.

March 31, 2014
NRC:14:014

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

Request for Review and Approval of ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application **to Control Rod Drop Accident (CRDA)"**

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP Inc.) to H. D. Cruz (NRC), "Request for Review and Approval of ANP-10300P, Revision 0, 'AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient Accident Scenarios'," NRC:09:134, December 23, 2009.

Ref. 2: Letter, Ronnie L. Gardner (AREVA NP Inc.) to H. D. Cruz (NRC), "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," NRC:09:133, December 22, 2009.

Ref. 3: BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

Ref. 4: NRC Memorandum, Technical and Regulatory Basis for the Reactivity Initiated Accident Acceptance Criteria and Guidance, January 2007, (NRC ADAMS Accession Number ML070220400).

AREVA Inc. (AREVA) requests the NRC's review and approval of the Topical Report ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," dated March 2014, for referencing in licensing actions.

This report presents a methodology for the evaluation of Control Rod Drop Accident (CRDA) for boiling water reactors (BWRs). The methodology builds on the AURORA-B AOO Methodology currently under review by the NRC (Reference 1) and consists of the thermal-hydraulic system code S-RELAP5, the Advanced Neutron Kinetics Method for BWR Transient Analysis (MB2-K), and the NRC approved advanced fuel performance code RODEX4 (Reference 3.) This methodology implements improvements in analysis capabilities to address items identified as deficiencies in the Reference 4 NRC Memorandum and provides a method to address the Interim Acceptance Criteria presented in Appendix B of SRP 4.2 Revision 3, as well as the anticipated final acceptance criteria. In addition to addressing the fuel failure thresholds, the methodology also incorporates the application of enthalpy dependent release fractions for the radiological consequence. The topical addresses both the gas release factions of the interim acceptance criteria as well as the proposed revised release fractions for Reg. Guide 1.183.

This topical report is part of AREVA's response to NRC Information Notice 2009-23: Nuclear Fuel Thermal Conductivity Degradation. This information notice states that previous fuel performance codes did not model the impact of irradiation on fuel thermal conductivity adequately. The RODEX4 fuel performance code in BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," (Reference 3) contains a nuclear fuel thermal conductivity model which accurately reflects the impact of

AREVA INC.

3315 Old Forest Road, Lynchburg, VA 24501
Tel.: 434 832 3000 - www.areva.com

irradiation. Also, the Supplement 1 to BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," (Reference 2) is currently under review by the NRC, and will extend capabilities to include a hydrogen up-take model upon approval. The updated S-RELAP5 Models and Correlations Code Theory Manual and the MB2-K Theory Manual will be provided for information in a follow-on transmittal to support review of this Topical Report.

AREVA considers some of the material contained in the enclosed documents to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the report are found in Enclosures 1 and 2, respectively. Enclosure 3 is the notarized Affidavit.

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

There are no commitments contained within the enclosures to this letter.

If you have any questions related to this submittal please contact Mr. Alan B. Meginnis, Product Licensing Manager, by telephone at (509) 375-8266, or by e-mail at Alan.Meginnis@areva.com.

Sincerely,

Philip A. Opsal for Pedro Salas
Philip A. Opsal

Pedro Salas, Director
Regulatory Affairs
AREVA Inc.

Attachments:

A. NRC Prioritization Matrix

Enclosures:

1. Proprietary Version of ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)"
2. Non-Proprietary Version of ANP-10333NP, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)"
3. Notarized Affidavit

cc: J. G. Rowley
Project 728

ATTACHMENT A:
NRC Prioritization Matrix

TR Prioritization Scheme Matrix for Metric and Resources			
Title: ANP-10333P, Revision 0, "AURORA-B Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)"			
Expect submitting FY	**TAC**	**PM**	**Today's Date: 3/31/2014**
Technical Review Division(s)		**Technical Review Branch(s)**	
Factors	**Select the Criteria That the TR Satisfies**	**Points can be Assigned for Each Criteria**	**Assigned Points**
TR Classification (Select one only)	Resolve Generic Safety Issue (GSI).	6	3
	Emergent NRC Technical Issue.	3	
	New technology improves safety.	2	
	TR Revision reflecting current requirements or analytical methods.	2	
	Standard TR.	1	
TR Applicability (Select one only)	Potential industry-wide applications.	3	2
	Potentially applicable to entire groups of licensees.	2	
	Intended for only partial groups of licensees.	1	
TR Implementation Certainty (Select one only)	Industry-wide Implementation expected.	3	1
	Expected implementation by an entire group of licensees (BWROG, PWROG, BWRVIP, etc.) who sponsored the TR.	2	
	Docketed intent by U.S. plant(s) but no formal LAR schedule yet.	1	
	No U.S. plant(s) have indicated strong intent on docket to implement yet.	0	
Tie to a LAR (Select if applicable)	A SE is requested by a certain date (less than two years) to support a licensing activity or renewal date (note it in Comments).	3	0
Review Progress (Points are cumulative as applicable)	Accepted for review.	0.3	0
	RAI issued.	0.5	0
	RAI responded.	1.2	0
	SE drafted.	2.0	0
Management (LT/ET) discretion adjustment		-3 to +3	
Total Points (Add the total points from each factor and total here):			6

Comments:

The 3 points for "TR Classification" is justified as the Methodology in ANP-10333P in that it provides a method to address revised criteria for the Control Rod Drop and it implements the RODEX4 models in the BWR CRDA Analysis. This methodology is needed to demonstrate compliance with the current Interim acceptance criteria of SPR 4.2 Appendix B, the anticipated final criteria, as well as the anticipated revision to RG 1.183 and 1.195.

.

The 2 points for "TR Applicability" are justified because AREVA could apply this methodology in a fuel transition for all BWRs.

The 1 point for "TR Implementation Certainty" is justified based on the assumption that all US BWRs will be required to meet the revised criteria of SRP 4.2 after it is published. This new methodology will be required for AREVA analyses to demonstrate compliance with the new criteria.

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the report ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," dated March 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 24th

day of March, 2014.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

NRC614:021
received 12/17/14

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

December 10, 2014

Mr. Pedro Salas, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: ACCEPTANCE FOR REVIEW OF TOPICAL REPORT ANP-10333P, REVISION 0, "AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO CONTROL ROD DROP ACCIDENT" (TAC NO. MF3889)

Dear Mr. Salas:

By letter dated March 31, 2014 (Agencywide Documents Access and Management System Accession No. ML14098A331), AREVA NP Inc. (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) ANP-10333P, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)." The NRC staff has performed an acceptance review of TR ANP-10333P. We have found that the material presented is sufficient to begin our comprehensive review. The methodology of ANP-10333P, Revision 0, is predicated on NRC staff approval of ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios." Therefore, the NRC staff technical review of ANP-10333P, Revision 0, cannot begin until, at a minimum, a draft safety evaluation (SE) of ANP-10300P, Revision 0, which defines the conditions and limitations of its application, is completed.

Due to the uncertainty of the creation of the draft SE for ANP-10300P, Revision 0, the NRC's current review priorities, and available technical resources; we are uncertain when we expect to perform an audit, issue our requests for additional information, and issue a draft SE for the review of TR ANP-10333P, Revision 0. The NRC staff estimates that the review will require approximately 480 staff hours including project management and contractor time once the review is started. The estimated review costs were discussed and agreed upon in an email between Gayle Elliott, AREVA Product Licensing Manager, and the NRC staff on October 10, 2014.

Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review. You did not request a fee waiver; therefore, NRC staff hours will be billed accordingly.

As with all TRs, the SE will be reviewed by the NRC's Office of the General Counsel (OGC) to determine whether it falls within the scope of the Congressional Review Act (CRA). During the course of this review, OGC considers whether any endorsement or acceptance of a TR by the NRC amounts to a rule as defined in the CRA.

If this initial review concludes that the SE, with its accompanying TR, may be a rule, the NRC will forward the package to the Office of Management and Budget (OMB) for further review and consideration. Any review by OMB would impact the schedule for the final issuance of the SE.

If you have questions regarding this matter, please contact Jonathan G. Rowley at (301) 415-4053.

Sincerely,

Anthony J. Mendiola, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

NRC-IC-16-
REC'D 12-19-2016
[signature]

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

December 9, 2016

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: AREVA INC. TOPICAL REPORT ANP-10333P/NP, REVISION 0, "AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO CONTROL ROD DROP ACCIDENT (CRDA)" (CAC NO. MF3889)

Dear Mr. Peters:

By letter dated March 31, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14098A331), AREVA INC. (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval Topical Report (TR) ANP-10333P/NP, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On November 9, 2016, Alan Meginnis, AREVA Product Licensing Manager, and I agreed that the NRC staff will receive the response to the enclosed RAI questions within 60 days from the date of this letter.

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

Sincerely,

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

Project No. 728

Enclosure:
RAI Questions

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO TOPICAL REPORT ANP-10333P/NP, REVISION 0

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

APPLICATION TO CONTROL ROD DROP ACCIDENT (CRDA)"

AREVA INC.

(CAC NO. MF3889)

Topical Report (TR) ANP-10333P/NP, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," presents a methodology and supporting evaluations to extend the use of the AURORA-B code system to analysis of the CRDA. The AURORAB code system is documented in TR ANP-10300P/NP, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient Accident Scenarios." The most significant regulatory requirement that the CRDA evaluation must satisfy is associated with General Design Criteria 28 in Appendix A to Title 10 of the *Code of Federal Regulations,* Part 50. The Standard Review Plan, NUREG-0800, provides interim acceptance criteria for reactivity insertion accidents (including CRDA events) in Appendix B to Section 4.2.

Potential Deviations

Two of the most important parameters used to determine if the interim acceptance criteria are met (fuel rod enthalpy and fuel temperature) are directly dependent on the magnitude of the power burst resulting from the CRDA. As a result, any potential deviation between the physical reality and the CRDA evaluation that may lead to a reduction in the limiting reactivity response in the CRDA analysis needs to be addressed. The NRC staff has identified some such potential deviations and determined that the following information is required in order to complete the evaluation.

RAI-1

The primary mitigation mechanism for the CRDA event is the Doppler reactivity effect as the fuel temperature increases. The TR explains that this effect **[**

], which were set to maintain consistency with the steady state core simulator code (MICROBURN-B2). The CRDA event is a very fast transient that primarily consists of a fuel power/temperature response and its mitigation by negative reactivity due to Doppler feedback. As such, it would be expected to be more

sensitive to the accuracy of the MB2-K treatment of the variation in the strength of the Doppler effect in the fuel rod due to variation in the fuel temperature. The TR provides results from a sensitivity study in Section 8.7.3.11, but it is not clear how the study supports the selected weighting factors or the uncertainty associated with the pellet radial power distribution. Please provide a discussion of the technical basis supporting the weighting factors and the uncertainty selected for use with the CRDA analysis methodology, including the following:

a. Changes in the radial fuel temperature distribution during the power excursion associated with the CRDA, and

b. The impact of fuel geometry changes due to irradiation (in particular, pellet growth and cracking) on the radial fuel temperature distribution.

RAI-2

Page 8-24 of the TR discusses the technical basis for control rod modeling and its uncertainty. This discussion does not appear to address the fact that some control rods are designed with axial variations in neutron-absorbing material. Such variations may affect the reactivity insertion curve during a CRDA. Please provide a discussion of how axial variations in neutron-absorbing material used in the control rod blade will affect the reactivity insertion curve, including any limitations to the applicability of the recommended modeling approach and uncertainties.

RAI-3

Page 8-20 of the TR discusses the power history studies to justify the use of nodal average powers in constructing the rod power history effects. The approach used appears to result in lower exposures for lower powered histories and higher exposures for higher powered histories. It is not clear if the observed changes in the enthalpy rise are due to the change in average power or the change in exposure. The latter effect is not relevant because the CRDA analysis is performed at the cycle exposures which are considered to be most likely to be limiting. If the intent of this study is to demonstrate that use of nodal average powers in constructing the rod power histories is acceptable, [

] used to arrive at the same fuel exposure.

RAI-4

The sensitivity study documented in Section 8.7.2.3 for the core initial coolant temperature consists of a series of perturbations on the core initial coolant temperature. An increase in the core initial coolant temperature would result in a reduction in reactivity due to the corresponding increase in fuel temperature and the Doppler effect. If this was not compensated for in some other way (e.g., rod pattern adjustment), then the sensitivity studies may incorporate a less critical core as the starting point, which could reduce the severity of the prompt power pulse. Please provide a discussion of the effect of changes to the core initial coolant temperature on the initial reactivity of the core, and how they are captured by the sensitivity study.

<u>**Scenarios Selection**</u>

Some of the details provided in the TR for the approach used to select CRDA scenarios for analysis do not include a justification that the approach is appropriate for its intended purpose. If the CRDA analysis is performed for scenarios that do not bound the worst case scenario, then

a non-conservative result will be used to demonstrate that the acceptance criteria for the CRDA event are met. Therefore, the NRC has determined that additional information is needed to clarify how the selection process outlined in the TR to select the appropriate scenarios for analysis will bound all possible scenarios.

RAI-5

Page 7-8 states that the rod drop with the highest static rod worths at three exposures for the cycle are used in the CRDA evaluation, along with other candidate rods identified to evaluate the impact of the CRDA on fuel rods with high exposure and cladding content. The pellet/cladding mechanical interaction (PCMI) failure threshold is dependent on the hydrogen content in the cladding, so fuel with higher exposure may fail the acceptance criteria even if the prompt enthalpy rise is smaller than lower exposure fuel. In order to address this possibility, selection criteria are provided to guide the selection of additional rods as necessary. It is not clear how the proposed selection criteria will ensure that any potentially limiting rods will be identified for a broad range of possible cladding hydrogen contents, fuel types, and plant configurations. Please describe how the selection criteria will be effective in identifying suitable candidate rods for analysis that will ensure that the acceptance criteria are met, especially for fuel with cladding hydrogen content in the range where the failure threshold rapidly decreases (75 to 150 parts per million).

RAI-6

Section 7.6 discusses the approach used to determine rod enthalpy increases for individual fuel rods, which is then used to determine how many rods will experience PCMI failure for fission gas inventory release purposes. The text is not clear regarding how the **[**

]. Please describe why this assumption **[**

] would be expected to yield bounding results of fission gas inventory releases for all possible fuel lattices, including those with strong poisons that have not yet fully burned out or those that have experienced strongly asymmetric operating conditions (e.g., adjacent control rod insertion).

RAI-7

Section 7.7 describes the evaluation of the CRDA event for at-power conditions using the critical power ratio (CPR). The proposed approach seems sufficient for typical analyses, but it is not clear how broad the generic applicability of the analysis is. Based on the discussion in the TR, it appears that the [ ]. If that is true, please describe how unusual operating conditions may affect this determination, such as insertion of suppression rods which result in a radial asymmetry in the core power distribution.

RAI-8

The TR does not appear to discuss the applicability of this methodology to mixed cores. Please describe any limitations or changes necessary to account for cores with non-AREVA fuel, including evaluation parameters not directly involved in the CRDA calculation such as the cladding hydrogen content.

Sensitivity studies were performed on various input parameters for the CRDA calculation. In some cases, these studies were used to support use of a bounding value for the CRDA analysis. In other cases, the study results were used to support a value for the uncertainty in the enthalpy rise. These studies and their results are used to provide reasonable assurance that the results of the CRDA analysis will bound real-world conditions. The NRC identified some cases where the sensitivity study approach or the conclusions did not clearly support the intended purpose, so further information is necessary.

RAI-9

A number of the conclusions derived from the sensitivity studies do not appear to be supported by the actual results from the calculations performed for the studies. For example, Section 8.7.2.5 discusses the sensitivity of the CRDA analysis results to the initial core flow. The text states that the prompt enthalpy rise decreases as the initial core flow increases, while the total enthalpy increases as the initial core flow increases. This is used to support the use of a minimum core flow as a bounding value for determining the prompt enthalpy rise. However, no clear recommendation is given for evaluation of the total enthalpy, and the sensitivity studies show that the limiting value for the prompt enthalpy rise was calculated for an initial core flow just above the recommended minimum value. Please provide further clarification for the behavior of the prompt enthalpy rise as a result of variations in the initial core flow, and provide guidance on the appropriate initial core flow to use when evaluating the total enthalpy.

RAI-10

Section 8.7.4 states [

]. Please provide a justification for the appropriateness of the statistical approach used.

Time Step Size

The guidelines from TR ANP-10300 were reviewed against the guidance provided in this TR. For the most part, the NRC staff assumed that the general AURORA-B guidance would also be appropriate for the CRDA analysis. [

]. The TR was not clear on how the time step size guidance supported this goal, so additional information is necessary.

RAI-11

The TR recommends a different time step size for some CRDA evaluations due to the nature of the transient. Sensitivity studies were performed to determine the impact of any further changes in time step size. However, it is not clear how the specific value recommended in the TR was determined. The documentation of the sensitivity studies only discusses calculations performed at a different time step size. Please clarify if the discussion on the time step sensitivity in Section 8.7.1 was intended to characterize the sensitivity study as showing that a further change in time step size beyond the recommendation in Table 7.5 would not yield a significant change in calculated enthalpy. If the NRC staff interpretation is in error, please provide sufficient information to enable an understanding of how the sensitivity study relates to the final recommendation on time step sizes.

A
AREVA

April 6, 2017
NRC:17:018

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

Response to Request for Additional Information Regarding Topical Report ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)"

Ref. 1: Letter, Pedro Salas (AREVA) to Document Control Desk (NRC), "Request for Review and Approval of ANP-10333P, Revision 0, 'AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)'," NRC:14:014, March 31, 2014.

Ref. 2: Letter, Jonathan Rowley (NRC) to Gary Peters (AREVA), "Request for Additional Information RE: AREVA Inc. Topical Report ANP-10333P/NP, Revision 0, 'AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)' (CAC NO. MF3889)," December 9, 2016.

AREVA Inc. (AREVA) requested the NRC's review and approval of Topical Report (TR) ANP-10333P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)" in Reference 1. The NRC provided a Request for Additional Information (RAI) regarding this TR in Reference 2. The responses to this RAI are included in this letter as Enclosure 1 (ANP-10333Q1P, Revision 0).

The information provided in Enclosure 1 has been developed using a new version of AURORA-B which contains updates to both the S-RELAP5 and MB2-K modules. As a result of the necessary code changes and other issues summarized in Table A.1 of Enclosure 1, information needs to be updated in the base TR to ensure the final methodology is sufficiently captured by the approved version of the TR. To facilitate the updating of the TR, AREVA will revise the TR to Revision 1 of ANP-10333P such that the approved TR will be ANP-10333PA, Revision 1. AREVA requests that the NRC acknowledge this in the safety evaluation (SE) for this TR.

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

There are no new commitments within this letter or its enclosures.

AREVA INC.

3315 Old Forest Road, Lynchburg, VA 24501
Tel.: 434 832 3000 - www.areva.com

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

Sincerely,

Gary Peters, Director
Licensing & Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures:

1. Proprietary copy of 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"
2. Non-Proprietary copy of ANP-10333Q1NP, 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"
3. Notarized Affidavit for Withholding of Proprietary Information

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the report 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," dated April 2017 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 5th

day of April, 2017.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

ANP-10333NP
Revision 0

AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)

March 2014

AREVA Inc.

AREVA

AREVA Inc.

ANP-10333NP
Revision 0

AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)

sja

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

Contents

Tables

Figures

Nomenclature

ABWR	advanced boiling water reactor
ASME	American Society of Mechanical Engineers
AST	alternate source term
BOC	beginning of cycle
BPWS	banked position withdrawal sequence
BWR	boiling water reactor
CCD	component calculational device
CET	component effects test
CFR	code of federal regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
CZP	cold zero power
EM	evaluation model
EMDAP	evaluation model development and assessment process
EPRI	Electric Power Research Institute
EPU	extended power uprate
FIST	full integral simulation test
FMA	foundation methodology assessment
FMCRD	fine motion control rod driver
FoM	figure of merit
GDC	general design criteria
IET	integral effects test
LHGR	linear heat generation rate
LOCA	loss of coolant accident
LPRM	local power range monitor
LTR	licensing topical report
LWR	light water reactor
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U.S.
PCMI	pellet clad mechanical interaction
PHE	peak hot excess reactivity
PIRT	phenomena identification and ranking table

PTD	plant transient data
PWR	pressurized water reactor
RIA	reactivity insertion accident
RLBLOCA	realistic large break loss of coolant accident
RPS	reactor protection system
SE	safety evaluation
SET	separate effects test
SRP	standard review plan
U. S. NRC	Nuclear Regulatory Commission, U. S.
ΔH	transient change in enthalpy
ΔH_p	prompt enthalpy rise
ΔH_tot	total enthalpy rise

Abstract

The AREVA methodology to analyze the boiling water reactor (BWR) Control Rod Drop Accident (CRDA) is presented. The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with criteria for the BWR CRDA event as presented in the U. S. NRC Standard Review Plan Section 15.4.9 (Reference 1). This report presents the CRDA requirements, followed by the code and model requirements, the application methodology, and methodology uncertainties.

1.0 Introduction

The AREVA methodology to analyze the boiling water reactor (BWR) control rod drop accident (CRDA) is presented. The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with regulatory criteria for the BWR CRDA event as presented in the U. S. NRC Standard Review Plan Section 15.4.9. (Reference 1). The methodology is structured such that it will support changes in the acceptance criteria.

The AREVA methodology for the CRDA evaluation includes both generic evaluations and cycle-specific analysis. Generic studies are used to address at power conditions and system pressurization. The cycle specific analysis includes the determination of candidate control rods that could challenge fuel failure criteria and the subsequent evaluation of these candidate rods with a three-dimensional neutron kinetics and thermal-hydraulics code system.

This methodology has been developed to support recent changes in the CRDA acceptance criteria and evaluation process as reflected in the Interim Acceptance Criteria and Guidance of Appendix B of SRP 4.2 (Reference 6).

This methodology is based on extending the model qualification of the AURORA-B system described in Reference 9. The term AURORA-B CRDA will be used throughout this document to distinguish between the AURORA-B Base Evaluation Model described in Reference 9 and the methodology presented in this document.

2.0 Summary

The models and methodology described and documented herein provides a means to show compliance with interim acceptance criteria for the BWR CRDA event as established in SRP 4.2 Appendix B (Reference 6).

The range of applicability includes BWR plant types, control blades, and fuel designs for which the AREVA lattice physics and nodal simulator methods have been validated or will be validated for. { The CASMO-4/MICROBURN-B2 methodology described in the NRC approved Topical Report, EMF-2158PA, Revision 0, (Reference 17), identifies the specific commercial reactors included within the topical in Section 7.2 . These reactors are identified as core size and the core lattice of either C or D. Application to reactors not included in the topical is demonstrated in accordance with Item 1 of Section 6 of SER for Reference 17. Benchmarking of the CASMO-4/MICROBURN-B2 methodology consistent with Item 1 of Section 6 of the SER, allows for modeling of BWRs equipped with external recirculation pump systems (BWR/2 plants), jet-pump recirculation systems (BWR/3 through BWR/6 plants), and internal recirculation pump systems (similar to ABWR plants). }

Evaluation of the event requires the establishment of initial conditions covering the range of cycle operation (exposure) and domain of coolant conditions (flow rate, pressure and temperature).

The cycle is evaluated to determine exposure points, target rod patterns, coolant conditions and candidate rods which when dropped could generate a power pulse that could potentially cause cladding failure. The selection of the initial conditions and candidate rods for evaluation is described in Sections 7.3 and 7.4.

The candidate rods are subsequently evaluated with the transient code which includes the following capabilities:

- Three-dimensional neutron kinetics which properly models the power excursions, including changes in reactivity for the dropped rod, the proper treatment of the Doppler effect in the fuel, the power peaking during the transient, and the reactivity change associated with the change in moderator conditions.
- Transient thermal energy transport in the fuel pellet capable of accommodating the rapid deposition of energy from the power excursion, and to properly represent the subsequent transport of thermal energy from the fuel pellets to the coolant and changes to the coolant state, momentum and energy.

- A control system model that properly inserts the non-faulted control rods on the receipt of a RPS trip signal.
- Determination of fuel enthalpy or CPR during the event.

The selection of the candidate rods is based on static rod worth as determined with the CASMO4/MICROBURN-B2 methodology (NRC approved Topical Report EMF-2158(P)(A) Reference 17). The location of the rod relative to fuel with potentially high hydrogen content in the cladding is also considered due to the impact on acceptance criteria.

Following selection of the candidate rods, the rod drops are then evaluated with the AURORA-B code system and the results are compared against the CRDA acceptance criteria. The methodology incorporates a hydrogen pickup model to allow assessment of the PCMI failure criterion as defined in the Interim Acceptance Criteria and Guidance of Appendix B of SRP 4.2 (Reference 6).

If fuel failures are determined, the radiological consequences are evaluated by comparing the fission product inventory release determined with revised enthalpy dependent release fractions against the actual fission product inventory release used in the plant licensing basis.

This methodology replaces the prior AREVA methodology for the CRDA (Reference 41) approved in 1983 with modern methods for fuel and plant analysis.

The methodology and documentation presented, identifies the various and current regulatory requirements applicable for the BWR CRDA event and identifies how these requirements are addressed and complied with.

3.0 Regulatory Requirements Summary

3.1 *Regulatory Requirements*

The acceptance criteria for the CRDA are based on General Design Criterion (GDC) 13 and 28 (Reference 2) and radiation dose limits from 10 CFR 100.11 and 10 CFR 50.67 (References 3 and 2). It is noted that specific acceptance criteria for the reactivity initiated accident are under review and that interim criteria are provided in SRP Section 4.2 Appendix B (Reference 6).

The specific acceptance criteria from NUREG-0800 Chapter 15.4.9 (Reference 1) are:

1. General Design Criterion (GDC) 13, as to 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.

2. Acceptance criteria are based on GDC 28 requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor result in sufficient damage to impair significantly core cooling capacity.

   Regulatory positions and specific guidelines necessary to meet the relevant GDC 28 requirements are in SRP Section 4.2 Appendix B and have been identified as interim criteria for the Reactivity Insertion Accident. These interim criteria are summarized in Table 3.1.

   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.

3. 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.

Final revised SRP design basis acceptance criteria are anticipated to be issued in the near future. The actual methodology is structured to support changes in the final criteria.

Table 3.1 Interim Criteria Summary

Criteria			SRP 4.2 Appendix B Revision 3
Fuel cladding failure	High Temperature	Zero Power	170 cal/g total enthalpy internal rod pressure below system pressure 150 cal/g total enthalpy internal rod pressure exceeding system pressure
		Power > 5%	Local heat flux exceed thermal design limit CPR
	PCMI	Cladding Hydrogen content (ppm)	
		<75	150 cal/g prompt enthalpy
		75 to150	Linear decrease from 150 cal/g at 75 ppm to 60 cal/g at 150 ppm
		150 to 300	Linear decrease from 60 cal/g at 150 ppm to 50 cal/g at 300 ppm
Core coolability	Peak radial average enthalpy		<230 cal/g
	Peak Fuel Temperature		Below incipient fuel melting conditions
	Mechanical Energy	Non-molten fuel to coolant interaction	Address with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity
		Rod burst	
	Cool-able geometry	Fuel pellet and cladding fragmentation	No loss of cool-able geometry
		Fuel rod ballooning	
Radiological			10 CFR 100.11 or 10 CFR 50.67 radiation dose limits met at exclusion area boundary and at the outer boundary of the low population zone

3.2 *Compliance with NUREG-0800 Chapter 15.0.2*

NUREG-0800 SRP 15.0.2 "Review of Transient and Accident Analysis Methods" (Reference 7) describes the review process and acceptance criteria for analytical models and computer codes. This section summarizes the compliance with the requirements identified in the SRP 15.0.2.

Several elements are identified related to documentation and model requirements in SRP 15.0.2. The documentation includes the following components:

A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

   A comprehensive description of the AURORA-B base EM code system is provided in ANP-10300P (Reference 9). The actual model requirements for the CRDA event are established in Section 5. The model description is provided in Section 7 along with roadmap for the CRDA analysis.

B. A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

   The scenario description and discussion of phenomena interactions are provided in Section 4. The result of the phenomena identification and ranking are reflected in the evaluation model requirements in Section 5.

C. A comprehensive description of the code assessment.

   A summary of the code assessment data base is presented in Section 6.0. The results of the code assessment are provided in Section 8.0.

D. Determination of code uncertainty for a sample plant accident calculation.

   The code uncertainty bases are established within Section 8 and the application for a sample plant accident calculation is provided in Section 9.

E. A comprehensive theory manual.

   The AURORA-B code system for the CRDA evaluation contains multiple computational devices. The theory manuals for S-RELAP5, MB2-K and RODEX4 are References 19, 20, and 22.

F. A detailed user's manual.

   The user manuals for S-RELAP5, MB2-K and RODEX4 are References 57, 68, and 69. The interface requirements for interaction of the computational devices are provided in Section 9.0 along with process and procedures to ensure correct transfer of information between the various computational devices.

G. A description of the Quality Assurance program under which the evaluation model was developed and assessed, and the corrective action program that will be used to address any error that might be discovered.

The AREVA Quality Assurance program, which complies with the requirements of Appendix B to 10 CFR Part 50 is identified in Section 10.0.

4.0 Scenario Identification

4.1 *Regulatory Basis*

From SRP 15.0.2 II.1.B the requirement for the accident scenario is presented which requires:

A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

The following section identifies the accident characteristics and the sequence of events. Detailed discussion of the phenomena identification and ranking is presented in Section 5.0 with the model requirements.

4.2 *Characteristics*

The BWR Control Rod Drop Accident (CRDA) is the result of a postulated event in which a fully inserted high worth control rod becomes decoupled from its drive mechanism. The drive mechanism is withdrawn but the decoupled control rod is assumed to be stuck. At a later optimum moment, the control rod drops at its maximum speed to the position of the control rod drive mechanism. This results in the insertion of large positive reactivity into the core resulting in a prompt critical power increase which is terminated by negative Doppler reactivity from the fuel temperature increase. With the large power excursion, the Reactor Protection System will initiate an insertion of the withdrawn control rods. The power pulse width for a BWR CRDA is approximately 45 to 75 milli-seconds such that the power has increased and decreased prior to the beginning of movement of the control rods in response to the scram. Although a small portion of the fission energy is directly deposited in the moderator, no significant change in the reactivity due to moderator feedback occurs until thermal conduction of heat from the fuel to moderator begins. Once the thermal energy transfer begins the local moderator temperature increases and creates substantial local voiding. This voiding results in additional negative reactivity insertion. Some fluctuation in the power from the void feedback may occur until the scram is completed which terminates the event.

The general characteristics of a CRDA are summarized:

- A rapid insertion of positive reactivity (the worth of the dropped rod)
- A rapid increase in localized power along with fuel temperature

- Insertion of negative reactivity through the broadening of resonance absorption cross section (Doppler effect)
- Rapid decrease in power generation in response to the negative reactivity
- Heat transfer from the fuel to coolant resulting in:
    - Small decrease in the negative Doppler reactivity with decreasing fuel temperature
    - Large increase in negative reactivity from moderator voiding
- Insertion of negative reactivity of SCRAM control rods

The analysis of CRDA is generally divided between two operating regimes: the startup range and the power range.

<u>CRDA in the Power Range</u>

In the power range, there are reactor characteristics that reduce the severity of the CRDA. The first is the consequences of the void distribution of the operating reactor which results in a smaller reactivity insertion rate for the dropped rod. With the presence of voids, and the coolant at saturated conditions, the direct heating of moderator has a greater potential to decrease the moderator density in the power range compared to the cold conditions for the startup range and the increase in voids results in smaller rod worths. Other characteristics are related to the Doppler Effect. The Doppler feedback in the power range occurs much more quickly relative to the magnitude of the power increase compared to that in the startup range. (In the startup range, the actual power must increase by several orders of magnitude prior to significant Doppler feedback.) Finally, with the presence of voids in the moderator the fuel to water ratio increases and the spectrum hardens. With the hardened spectrum the resonance absorption of neutrons increases.

These characteristics all tend to reduce the severity of the CRDA in the power range.

<u>CRDA in the Start-Up Range</u>

During the startup sequence, the rod patterns employed are permitted by constraints on rod movements by the technical specification restrictions on the order in which control rods are withdrawn including the maximum number of bypassed control rods. The Banked Position Withdrawal Sequence (Reference15) is an example of a set of restrictions intended to reduce the maximum rod worth that is used by most BWRs. These type of withdrawal sequences are

typically enforced with rod pattern control systems. The AREVA CRDA methodology presented herein can be applied to any specified rod withdrawal sequence.

Sequence of Events and System Response

The sequence of events for the CRDA may be summarized as follows (this sequence has been annotated from that given in Reference 14):

1. At some time during the withdrawal of control rods from the reactor, a complete (but not necessarily sudden) rupture, breakage, or disconnection of a random fully inserted control rod drive from the companion control blade occurs.
2. The control rod drive associated with the disconnected control blade is withdrawn as a part of the start-up process and the control rod blade sticks at the fully inserted position, rather than follow the moving control rod drive. This fault is not detected by the plant operators.
3. At some later time, under critical reactor conditions, the rod pattern causes the decoupled rod to have the maximum worth from fully inserted to the position of its drive. The faulted control blade drops at that time to the location of the companion control rod drive.
4. A rapid increase in reactor power near the location of the dropped control blade occurs, and a correspondingly sudden increase in fuel temperature in the nearby fuel assemblies. The fuel temperature reactivity feedback (Doppler) terminates the initial power burst.
5. The reactor protection system trips on a high flux signal from the Source Range Monitor, the Intermediate Range Monitor, or the Average Power Range Monitor; although the trip is generally assumed to occur at a reactor power of 120%, rather than explicitly representing the various trip systems.
6. All of the withdrawn control rods aside from the decoupled control rod insert into the core (scram) at the technical specification scram speed.
7. When all of the control rods, except for the faulted control rod, are fully inserted, the reactor is assured to be sub-critical due to the Shutdown Margin requirement. This state represents the termination of the accident.

Event Evaluation

Evaluation of the event requires the establishment of initial conditions covering the range of cycle operation (exposure) and domain of coolant conditions (flow rate, pressure and temperature).

The cycle is evaluated to determine exposure points, target rod patterns, coolant conditions and candidate rods which when dropped could generate a power pulse that could potentially cause cladding failure.

The candidate rods are subsequently evaluated with the transient code which includes the following capabilities:

- Three-dimensional neutron kinetics which properly models the power excursions, including changes in reactivity for the dropped rod, the proper treatment of the Doppler effect in the fuel, the power peaking during the transient, and the reactivity change associated with the change in moderator conditions
- Transient thermal energy transport in the fuel pellet capable of accommodating the rapid deposition of energy from the power excursion, and to properly represent the subsequent transport of thermal energy from the fuel pellets to the coolant and changes to the coolant state, momentum and energy
- A control system model that properly inserts the non-faulted control rods on the receipt of a RPS trip signal
- Determination of fuel enthalpy or CPR during the event

The result of the transient are then evaluated against the CRDA criteria to assess fuel failures, core coolability, system over pressurization, and radiological consequences.

5.0 Evaluation Model Requirements

5.1 *Regulatory Basis*

From SRP 15.0.2 II.1.B the requirement for the accident scenario is presented which includes: A description of the accident scenario including initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and interactions between systems and components, if appropriate, that affect the outcome of the accident.

5.2 *Model Requirements*

The accident characteristics and sequence of events are presented in Section 4. This section presents a detailed discussion of the phenomena identification and ranking. The format used for the subsequent sub-sections is based on Element 1 of Reference 8. Since this methodology development is based on the AURORA-B base EM (Reference 9), the EMDAP process is focused on the changes to the AURORA-B base EM and extension of the validation for CRDA.

5.2.1 Analysis Purpose, Transient Class, Plant Class, and Fuel Designs

The purpose of the analysis is to ensure that the regulatory requirements for the BWR CRDA event are met.

The transient class for which the methodology described within this topical is defined in SRP 15.4.9 Spectrum of Rod Drop Accidents (BWR). In some instances the licensing basis for a plant may not adhere to the SRP, or may refer to prior revisions of the SRP. It is noted that revision of the acceptance criteria for the CRDA is ongoing at the time of document preparation.

The BWR plant types, control blades, and fuel designs for which the methodology is applicable, should be consistent with those for which the AREVA lattice physics and nodal simulator methods have been validated. This includes BWRs equipped with external recirculation pump systems (BWR/2 plants), jet-pump recirculation systems (BWR/3 through BWR/6 plants), and internal recirculation pump systems (similar to ABWR plants). A detailed description of the BWR plant types is provided in Reference 14. With respect to the CRDA, the primary difference between reactor designs is the incorporation of Fine Motion Control Rod Drive (FMCRD). The control blade design for FMCRD does not incorporate the velocity limiter. BWR2-6 plants all have velocity limiters.

5.2.2 Figures of Merit

The Figures of Merit (FoM) evaluated are parameters that demonstrate compliance with applicable acceptance criteria defined in the licensing basis of each plant. Specifically, the FoMs (also known as acceptance criteria measures) considered are those necessary to demonstrate compliance with the acceptance criteria for the CRDA. Ideally the FoM's would correspond identically to the regulatory requirement however, regulatory guidance permits "surrogate" standards for demonstrating compliance. The FoMs include prompt fuel enthalpy rise, total fuel enthalpy, MCPR, system pressure, fission product inventory release fractions, and core coolability. (The fission product inventory release is used as a "surrogate" FoM for the event dose consequences.) Evaluation of the MCPR and system pressurization FoM's are included in the AURORA-B base EM.

Figure of Merit 1 Fuel Enthalpy

Fuel enthalpy is used in the determination of cladding failure and gap release fractions. Two enthalpy values are considered; the prompt enthalpy rise and the total enthalpy. The prompt fuel 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 simply the maximum enthalpy experienced during the event. The enthalpy figure of merit is used in two ways for evaluating the CRDA. The first is to determine if fuel cladding failure criteria have been exceeded and the second is used with other methods to determine radiological consequences of the CRDA if fuel cladding failure occurs.

1. The prompt fuel enthalpy rise is evaluated and compared to the criteria for PCMI (Pellet Clad Mechanical Interaction) cladding failure. The total enthalpy is compared against the high temperature cladding failure criteria.
2. If the loss of fuel cladding integrity has occurred, independent of the driving mechanism (PCMI, high temperature, MCPR) this figure of merit in conjunction with other methods can be used in the determination of the gap inventory release fraction for the failed rods.

Figure of Merit 2 MCPR

Evaluation of MCPR is performed for at power conditions to determine fuel failure. If a rod exceeds the critical power it is assumed failed and will be included in the radiological consequence evaluation. This figure of merit is addressed in the AURORA-B base EM (Reference 9).

Figure of Merit 3 Peak System Pressure

Evaluation of peak system pressure is made to assure the peak system pressure remains below applicable ASME limits for the CRDA. This figure of merit is addressed in the AURORA-B base EM (Reference 9).

Figure of Merit 4 Fission Product Inventory Released

The total fission product inventory released from fuel rods which have experienced cladding failures is used for comparison with plant licensing basis for the CRDA. The inventory released is determined by the gap inventory release fractions which are composed of an enthalpy dependent release term and a steady state term. The inventory released can then be used to determine an equivalent number of failed fuel rods relative to using only the steady state release term for comparison with a specific number of rod failures. Alternatively, the inventory released can be compared directly to the plant licensing basis source term for the CRDA.

Figure of Merit 5 Core Coolability

Evaluation of core coolability is addressed using maximum deposited enthalpy, peak fuel temperature, the maximum cladding temperature and the power pulse characteristics.

5.2.3 Identify Systems, Components, Phases, Geometries, Fields, and Processes That Will Be Modeled

The evaluation model is the same as that used for the analysis of Anticipated Operational Occurrences with the extension of qualification to cold reactor conditions, and large rapid insertions of positive reactivity. The systems, components, phases, geometries, fields, and processes modeled are unchanged from Reference 9.

5.2.4 PIRT Summary

The figures of merit were reviewed with respect to the existing capabilities of the AURORA-B base EM. The ability of the AURORA-B base EM to assess the MCPR FoM and the peak system pressure FoM is documented in Reference 9. Therefore, the CRDA PIRT is generated for the enthalpy FoM. The AURORA-B base EM PIRT (Reference 25) and NRC PWR rod ejection PIRT (Reference 16) were used in the PIRT development.

The Fission Product Inventory Release FoM and the Core Coolability FoM are based on the determination of fuel failures which are caused by exceeding an enthalpy criteria or MCPR.

This PIRT is specific to the evaluation of the CRDA event. The generation of the PIRT involved review of existing industry PIRTs for the CRDA and PWR rod ejection.

Ranking of the phenomena was completed with respect to their influence on the enthalpy FoM.

Rankings are high (H), medium (M), low (L) or not-ranked (N).

Table 5.1 Control Rod Drop Accident Analysis PIRT

Subcategory	Phenomenon	R	Rationale
Calculation of power history during pulse	Control rod worth	H	Determines the amount of reactivity insertion.
	Rate of reactivity insertion	M	Within limits, the outcome is insensitive to the rate of reactivity insertion.
	Moderator feedback	M	The moderator temperature rise is small and corresponding reactivity effect is small, but not negligible.
	Fuel temperature feedback	H	The fuel temperature feedback causes the power excursion to turn around and essentially limits the energy deposition.
	Delayed-neutron fraction	H	Determines when prompt criticality is reached
	Reactor trip reactivity	L	It is important to have the rods trip to terminate the event but the effect is minor relative to the pulse.
	Fuel cycle design	H	Determines control rod worth and core loading
	Direct Energy deposition to moderator	L	The fraction of energy deposited directly in the moderator is small.
Calculation of rod fuel enthalpy increase during pulse (including cladding temperature)	Heat resistances in fuel, gap, and cladding	M	Per Reference 16, at a maximum, 25 percent of the deposited energy is conducted out and does not contribute to the fuel enthalpy.
	Transient cladding-to-coolant heat transfer coefficient	M	Per Reference 16, at a maximum, 25 percent of the deposited energy is conducted out and does not contribute to the fuel enthalpy.
	Heat capacities of fuel and cladding	M	Enthalpy is the integral of heat capacity and temperature. Enthalpy and enthalpy increases are both highly important. Small variations in the heat capacities have little impact on the overall event.
	Pellet radial power distribution	M	This element is rated lower because it is only part of the overall heat transfer process.
	Rod-peaking factors	H	Determines how much energy is directed to the peak location.
	Pellet and Cladding dimensions	L	Reflected in the heat resistances
Initial conditions	Burn-up distribution	N	
	Cladding hydrogen content	M	Determines enthalpy failure criteria
	Power	M	Affects reactivity feedback
	Coolant temperature and pressure	M	Affects moderator properties
	Core Flow	M	Affects moderator properties during transient
	Fuel rod internal pressure	L	Determines high temperature failure threshold

5.2.4.1 Control Rod Worth

The control rod worth determines the amount of reactivity insertion. A strong correlation exists between the control rod worth and the peak fuel enthalpy. The worth depends on fuel cycle design, cycle lifetime, and initial conditions. The initial conditions are required to be a reasonable representation of the limiting conditions allowed by plant Technical Specifications with respect to inoperable control rods that maximize the worth of the dropped rod.

The MICROBURN-B2/CASMO-4 methodology described in the NRC approved Topical Report, EMF-2158, Revision 0, (Reference 17), is fully capable of determining the dropped control rod worth, and establishing the appropriate initial conditions for power, flow, temperature, rod pattern and state in cycle.

5.2.4.2 Rate of Reactivity Insertion

The rate of reactivity insertion is controlled by the actual rod worth and the rate at which the control rod drops from the core. For BWRs 2 through 6, the control rod drop velocity is maintained with the control rod velocity limiter. Based on the results of velocity limiter tests, a conservative maximum rod velocity of 3.11 ft/s is used from the Reference 10 Appendix Velocity Limiter Tests (Page A-1). This was determined to be a bounding conservative velocity for BWRs 2 through 6.

The methodology will use a bounding rod drop rate or provide measured data which can support a slower drop rate.

5.2.4.3 Moderator Feedback

For the CRDA, moderator feedback is the change in the reactivity feedback from moderator temperature and density changes in the active, by-pass and water channels/rods. This also includes the changes in void fractions in the coolant. These changes are a result of direct energy deposition to the coolant and heat transfer from the cladding. Both conditions are specifically modeled in the 3-D kinetics code with thermal hydraulics feedback.

The methodology will model both moderator temperature and moderator void feedback. This capability is demonstrated in the AURORA-B base EM.

5.2.4.4 Fuel Temperature Feedback

Fuel temperature feedback is the reactivity feedback from fuel temperature changes. This results from the heating of fuel and the associated neutronic effects, in particular the Doppler effect. The Doppler reactivity feedback is the primary mechanism which reverses the power transient.

The evaluation module will model the Doppler feedback as a function of the fuel temperature changes.

5.2.4.5 Effective Delayed-Neutron Fraction

The effective delayed neutron fraction (β_{eff}) determines when prompt criticality is reached.

The evaluation model will include the effective delayed neutron fraction.

5.2.4.6 Reactor Trip Reactivity

Reactor trip reactivity is the negative reactivity associated with insertion of control rods after receipt of a reactor trip signal and the rate at which it is inserted. The actual event is turned around by Doppler prior to the start of motion of the rods being inserted. Consequently it will ultimately terminate the event in that all control rods, except the dropped rod, are inserted into the core to bring it to a subcritical state.

The actual magnitude of the power pulse is insensitive to the reactor trip reactivity. Therefore, the model will use minimum plant Technical Specification scram rates.

5.2.4.7 Fuel Cycle Design

The fuel cycle design includes those important design elements that determine the neutronic properties of the core at event initiation, such as the loading pattern, control history (control rod pattern, spectral shift), burnup, and exposure.

The evaluation model will represent the reactor core characteristics with respect to fuel loading and operation history. This capability is demonstrated in the AURORA-B base EM.

5.2.4.8 Heat Resistances in High Burnup Fuel, Gap, and Cladding

The resistances offered by the fuel, gap, and cladding to the flow of thermal energy from regions of high temperature to regions of lower temperature. The resistance is dependent upon the path length and thermal conductivity, which change with burnup.

The model must represent the exposure dependent heat transfer properties of the fuel, gap, and cladding. This capability is demonstrated in the AURORA-B base EM.

5.2.4.9 Transient Cladding-to-Coolant Heat Transfer Coefficient

The correlation that determines transport of energy at the interface by one or more of the following modes: forced convection-liquid, nucleate boiling, transition boiling, film boiling, or forced convection-vapor.

Due to the rapid nature of the power burst, there is limited change in heat transfer during the power burst. However, subsequent to the power burst the heat transfer from the rod to the coolant will be represented.

5.2.4.10 Heat Capacities of Fuel and Cladding

The respective quantities of heat required to raise the fuel and cladding one degree in temperature at constant pressure. The heat capacity of the fuel defines the amount of absorbed energy prior to an increase in temperature. The evaluation model will represent the material heat capacities.

5.2.4.11 Direct Energy Deposition to Moderator

Direct energy deposition into the active channel fluid occurs as the fluid attenuates γ-rays and neutrons that escape the fuel rods. The amount of energy deposited in the fluid is a function of the local fluid density and locally generated fission power of the fuel. The partition of the energy deposition which is deposited directly in the fuel pellet and that deposited in the moderator and structural components may affect the reactivity feedback mechanism.

The evaluation model will partition the energy deposition between the fuel and moderator/coolant within the core.

5.2.4.12 Pellet Radial Power Distribution

The radial distribution of the power produced in the fuel rod describes the change in power intensity from the center of the pellet to its outermost edge. The pellet radial power profile could affect the rate of energy transferred from the fuel pellet to coolant and the pellet temperature distribution for Doppler effects.

5.2.4.13 Rod-Peaking Factors

Rod-peaking factors represent the nodal rod or pellet power distribution within an assembly during the transient. As the rod is withdrawn there will be a significant change in the pin peaking factors.

The peaking factor is utilized to determine the pin enthalpy and critical power. The peaking includes the effects of power tilts that may occur across the bundles during the CRDA. The pin peaking factor change during the transient must be properly modeled.

5.2.4.14 Pellet and Cladding Dimensions

Uncertainty in the pellet and cladding dimensions are well known at beginning of life. The pellet and cladding dimensions change with exposure. The primary impact of dimensional changes is on the thermal heat transfer from the fuel to the coolant. The impact of the variance in the pellet and cladding dimensions are reflected in the thermal mechanical properties of the fuel rod.

5.2.4.15 Burnup Distribution

The local radial burnup distribution is homogenized in the cross section lookup process. The power distribution uncertainty of the kinetics includes the impact of the radial burnup distribution. The burnup distribution is reflected through the core power distribution, the control rod worth, and the fuel rod thermal-mechanical properties.

5.2.4.16 Cladding Hydrogen Content

The content of hydrogen in the cladding at the start of the transient is used in the determination of the failure criteria. This does not have an impact on the actual kinetic response, but impacts fuel cladding failure threshold for PCMI.

The methodology includes a method to calculate the cladding hydrogen content at the beginning of the event.

5.2.4.17 Power Distribution

The total core power and core power distribution at the start of the event. The initial power distribution is driven by the time in cycle and the initial control rod pattern.

The evaluation model will adequately represent the core power and power distribution at the start of the event.

5.2.4.18 Coolant Initial Conditions

The coolant initial conditions include pressure, temperature and the flow distribution (active channel, bypass, and water rod flows) at the start of the transient.

The methodology will address the spectrum of initial conditions for the event. The initial conditions in the power range are addressed in the AURORA-B base EM.

5.2.4.19 Total Core Flow

The total core flow refers to the total core flow at the beginning of the event. The actual flow value is not expected to change during the event.

The evaluation model will be capable of modeling the localized flow re-distribution during the event. This is addressed in the AURORA-B base EM.

5.2.4.20 Fuel Rod Internal Pressure

The fuel rod internal pressure relative to the system pressure is used to determine the high temperature failure criteria. This parameter is only important if the less restrictive high temperature criteria are applied.

6.0 Assessment Data Base Summary

6.1 *Regulatory Basis*

From SRP 15.0.2 II.1.C the requirement for the documentation requires a comprehensive description of the code assessment data base.

6.2 *Assessment Data Base*

A summary of the Assessment Data Base is presented in this section. The actual assessment results are provided in Section 8. The comprehensive assessment data base and the evaluation of the AURORA-B base EM are provided in Section 4 and 6 of Reference 9. For evaluation of the CRDA, the assessment base is expanded to include modeling of the SPERT III RIA experiments. For the at-power CRDA, the assessment data base information provided for the AURORA-B base EM (Reference 9) includes reactivity insertion events. For the CRDA in the start-up range, a sub-set of the assessment data supplied for the AURORA-B base EM is applicable. A summary of the Assessment Data Base is presented in this section for those items identified as important for the CRDA evaluation. The actual assessment results are provided in Section 8.

Table 6.1 is the assessment matrix which shows the component effects tests that were selected and the highly ranked PIRT phenomena addressed.

Core State Conditions

The capability of the CASMO4/MICROBURN-B2 code system to provide adequate cross sections, fuel burnup, spectral history information, and static control rod worth is provided in the LTR for CASMO4/MICROBURN-B2 (Reference 17). The assessment base for the CASMO4/MICROBURN-B2 consist of several reactors and cycles of measured data for comparison of measured TIP signals and cold and hot eigenvalue benchmarking. The assessment base also includes gamma scan data as well as calculations with higher order methods.

Numerical Kinetics Benchmarks

Numerical benchmarks provide a useful test of the equations and numerical solutions because the benchmarks are mathematical problems whose solutions are well known. Specifically, they

demonstrate the performance of the kinetics equations to simulate industry-standard problems that are specifically designed to test spatial and temporal equations numerical solutions. The assessment data base for the numerical benchmark include the 2-D TWIGL seed-blanket reactor problem (Reference 30), the LMW 3-D PWR delayed critical transient problem (Reference 31) and the LRA 2D and 3D BWR Control Blade Drop Transients (Reference 32).

Integral system test of Turbine Trip Event

The assessment of the capability of the AURORA-B base EM to model global fast transient conditions for an actual reactor utilizes the Peach Bottom Turbine Trip tests (Reference 37 and 39). This provides an assessment of 3D power distribution, scram reactivity of control rods, void/reactivity relationship, fuel rod thermal & mechanical properties and Doppler reactivity.

Integral test of a CRDA

A series of reactivity tests were performed as part of the Special Power Excursion Test (SPERT) III E-core program (Reference 42). The actual tests were conducted in the 1960s as described in Reference 43. These experiments involved the rapid reactivity insertions ranging from $0.5 to $1.3. The experiments were performed at cold startup, hot startup, hot standby, and operating power.

The capability of the AURORA-B system to simulate the kinetic response to the large insertion of reactivity is evaluated with this assessment.

Hydrogen Pickup Model

The hydrogen pickup model used is described in Reference 58 “Response to question 1b” on pages 10 through 21. The assessment of the model against measured data is provided in Figure 2 of Reference 58 (page 18).

Table 6.1 AURORA–B CRDA Evaluation Model Assessment Matrix

[

]

7.0 Evaluation Model Description

7.1 *Regulatory Basis*

From SRP 15.0.2 II.1.A, requires that the documentation contain an overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

A comprehensive description of the AURORA-B base EM code system is provided in ANP-10300P (Reference 9). The actual model requirements for the CRDA event are established in Section 5. The following Sections contain descriptions of the evaluation model and process for analysis of the CRDA and includes a roadmap for the CRDA analysis. This is collectively referred to as the CRDA methodology.

7.2 *CRDA Evaluation Methodology*

The AREVA methodology for CRDA analysis involves the selection of conservative initial conditions (see Section 7.3) and candidate rods (see Section 7.4) which would result in the most severe consequences for the CRDA. Following the selection of the candidate control rods, the rod drops are simulated with the AURORA-B code system. The results obtained from the simulation are then evaluated against the Interim Acceptance Criteria of Appendix B of SRP 4.2 (Summarized in Section 3.1 Regulatory Requirements of this document) to assess rod failure and subsequent fission product inventory release. The methodology includes generic components, plant specific components, and cycle specific components. The CRDA evaluation road map is presented in Figure 7.1. Detailed guidance for the analysis of the CRDA with the AURORA-B code system is provided in Reference 65.

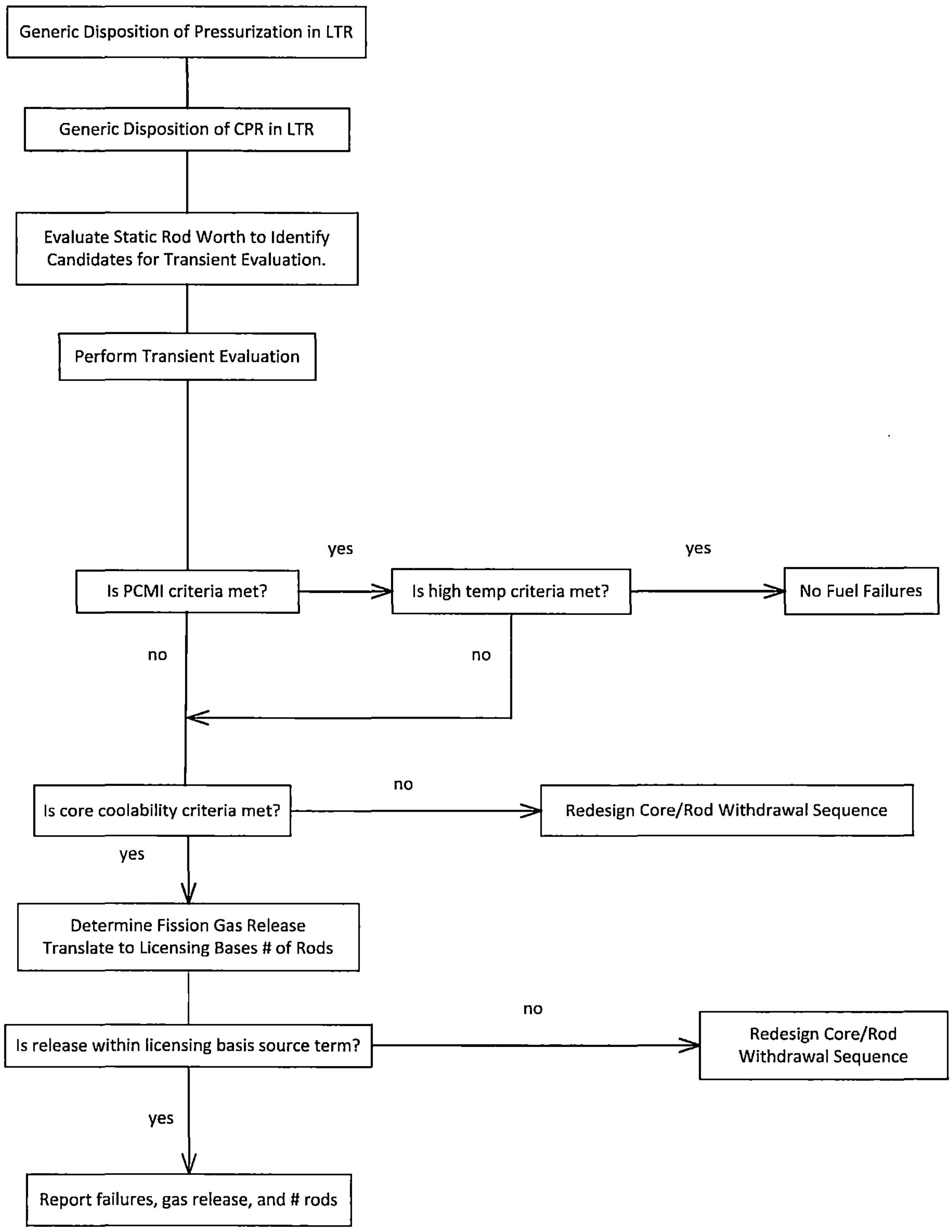

Figure 7.1 CRDA Evaluation Road Map

7.3 *Determination of Conservative Initial Conditions and Parameters*

The CRDA analysis methodology involves the selection of initial conditions that represent the most limiting conditions for the CRDA. The selection of the conservative initial conditions differs between the startup range and the power range. The primary initial conditions for a given point in the cycle include the core coolant conditions, power, flow, and control rod pattern. The in cycle evaluation times are also determined.

7.3.1 Startup Range

In the startup range the initial conditions that must be determined are the initial coolant properties, the initial core power, the initial core flow, and the inoperable control rods. The determination of the initial core flow and the inoperable control rods is determined on plant specific bases. Sensitivity studies for initial conditions are included within the methodology assessment results provided in Section 8.

7.3.1.1 Initial Coolant Conditions

The selection criteria has been developed using simulation of reactor startup sequences with various temperatures and pressures. Although the actual static rod worth varied little with moderator temperature, the lower temperature for a given rod worth results in higher fuel enthalpy. Sensitivity studies for initial coolant conditions are included within the methodology assessment results provided in Section 8.7.2.3.

The use of the cold temperature below that at which the reactor actually goes critical is conservative for the startup range CRDA for the following reasons:

- There can be a small positive reactivity increase with modern fuel designs as the moderator temperature increase up to 300 °F.
- The actual temperature at which BWRs go critical is higher than 70°F as indicated in Table 7.1.

Table 7.1 BWR Critical Temperatures

[

]

7.3.1.2 Initial Flow and Power

The initial flow rate and power are determined based on sensitivity studies as shown in Sections 8.7.2.4 and 8.7.2.5. The event is assumed to occur with a xenon free core. (See responses to RAI-5 and RAI-9 in ANP-10333Q1P Revision 0)

7.3.1.3 In-Cycle Evaluation Times

The analysis is performed at beginning of cycle, peak hot excess reactivity and end of full power conditions. These in-cycle conditions represent various core conditions and reactivity arrangements that may be encountered throughout the cycle.

7.3.1.4 Inoperable Control Rods

Inoperable rods locations are defined consistent with those allowed by plant technical specifications in such a manner to maximize the worth of the candidate rods. [

]

A typical A-sequence group assignment of control rods for a core with 764 assemblies is shown in Figure 7.2. The inoperable rods are located in South-East half of the core and are placed as close as allowed. Plant Technical Specification identify the maximum number of rods that can be out of service, the maximum number of rods per rod group, and the minimum separation distance between out of service rods. Although 3 inoperable rods may be allowed, two rods are assigned to each of the first 4 groups. The assignment requires two rods separation between inoperable rods.

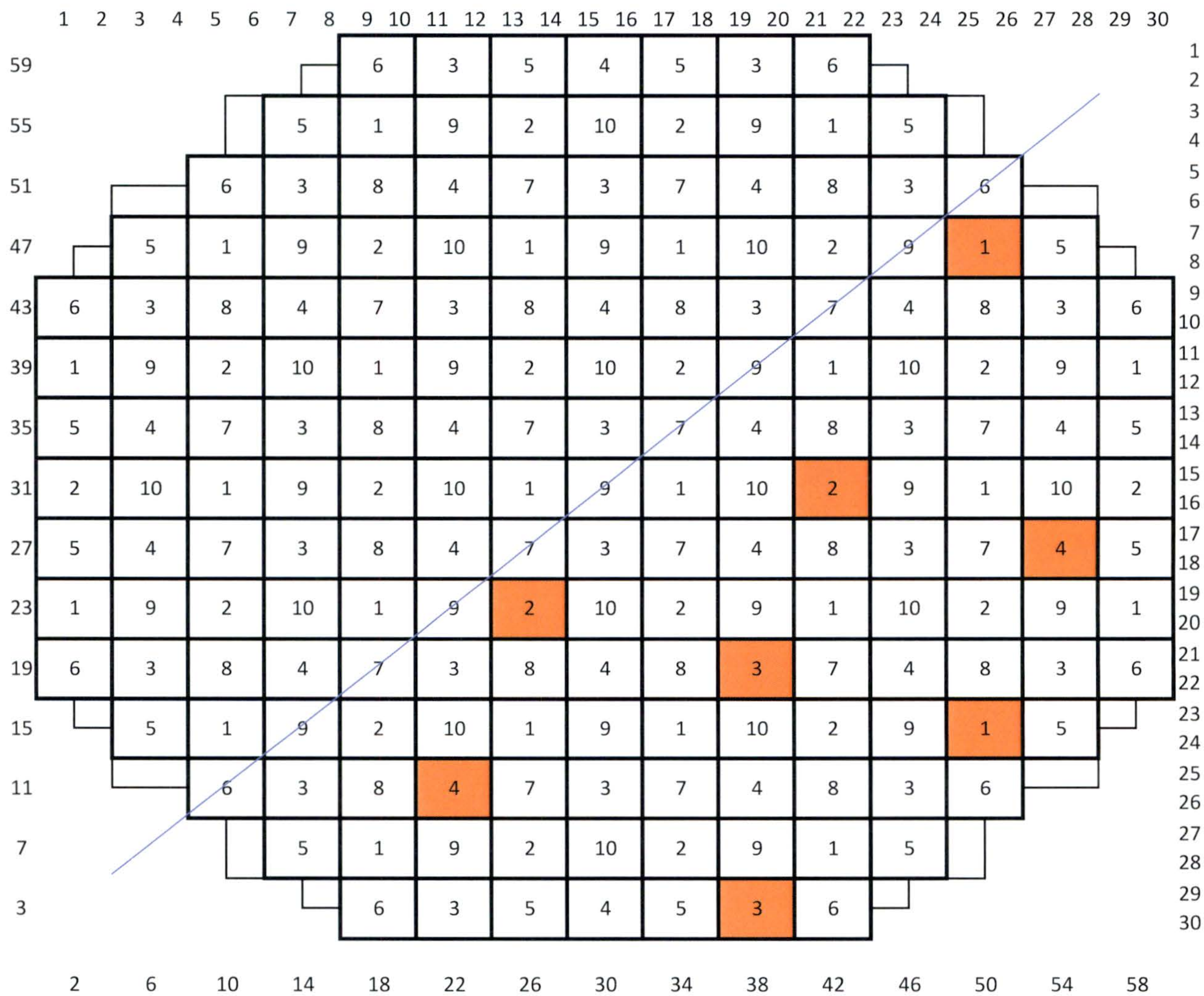

Figure 7.2 Example A-Sequence Groups with Inoperable Rod Locations

7.3.2 Power Range

For at power conditions, control rods are inserted and dropped to the current position of the control rod drive. This includes rods which are fully withdrawn. (The insertion and dropping of a rod would be similar to performing a sequence exchange and a rod of the prior sequence remains stuck in the core.) The rod drops are performed based on the licensing step through and utilizes the target rod patterns.

7.4 *Selection of Candidate Rods*

7.4.1 Startup Range

The rod selection process for the startup range CRDA is based on the static worth of the control rods and a broad range at which criticality is assumed to occur. The reactor is assumed to be critical if the calculated k-eff for the initial rod pattern is within the range;

[

] Table 7.2 provides a summary of cold critical eigenvalue variation with the MICROBURN-B2 code system. Since the target is established based on prior cycles, the consistency of the MICROBURN-B2 results demonstrate that the selection criteria will bound the actual reactor criticality.

Table 7.2 Cold Eigenvalue Variation for Recent Cycles

[

]

In order to determine the most limiting rod drop, each rod within a rod pull group is evaluated as if it is the first rod pulled in the group. This results in maximizing the worth of the control rod. As the control rod density in the core decreases and nuclear heating begins, the individual control rod worth also decreases. [ ]

To bound rods in higher order groups, the pull sequence can be analyzed with each higher order group being [ ]

Tabulation of the maximum rod worths for the three outer rings, as shown in Figure 7.3, is provided in Table 7.4. The rod worths in the first ring are very low as a result of the high exposed fuel and the neutron leakage from the core. As such, [ ] for the CRDA in cycle specific reload analysis.

Table 7.3 Upper Group Rods Bounded By Lower Group

[

]

Following the establishment of the static rod worth values, the rods are identified for which detailed transient analyses are to be performed.

The rod selection process is defined in the response to RAI-5 in ANP-10333Q1P Revision 0.

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					1	1	1	1	1	1	1				
55				1	2	2	2	2	2	2	2	1			
51			1	2	3	3	3	3	3	3	3	2	1		
47		1	2	3								3	2	1	
43	1	2	3										3	2	1
39	1	2	3										3	2	1
35	1	2	3										3	2	1
31	1	2	3										3	2	1
27	1	2	3										3	2	1
23	1	2	3										3	2	1
19	1	2	3										3	2	1
15		1	2	3								3	2	1	
11			1	2	3	3	3	3	3	3	3	2	1		
7				1	2	2	2	2	2	2	2	1			
3					1	1	1	1	1	1	1				

Figure 7.3 Example Control Rod Ring Locations

Table 7.4 Maximum Rod Worth (Δk) for Outer Rings

[

]

7.4.2 Power Range

The change in steady state CPR is used for the selection criteria in the power range. The power range criteria are not burnup dependent therefore, only the rod drop with the largest change in steady state CPR is evaluated. These selections are made using the approved core simulator (MICROBURN-B2 Reference 17) and an NRC approved critical power correlation (References 33, 34, or 35).

7.5 *Transient Evaluation Model*

The AURORA-B CRDA transient model maintains all capabilities of the AURORA-B base EM of Reference 9. The primary changes to the model include the expansion of processing cold library data, pin power reconstruction at cold conditions, and the inclusion of a peak rod heat structure. Due to the localized nature of the CRDA, a simplified plant system model is used.

The code structure, field equations, closure relations, and code numeric are the same as that described in Reference 9.

The thermal-hydraulic and thermal conduction field equations used within S-RELAP5 are the same as used in the RLBLOCA methodology described in Reference 18 and other USNRC

approved methodologies based on S-RELAP5. Extensive technical detail related to the field equations is provided in the S-RELAP5 theoretical description given in Reference 19.

The neutron kinetics equations contained within the MB2-K 3D kinetics model are described in detail within the MB2-K theoretical description (Reference 20). These field equations solve the transient three-dimensional and two-group equation set with six delayed neutron groups. The advanced nodal methods used in MICROBURN-B2 are also used in MB2-K to solve the spatial neutron flux distribution. A fully implicit numerical scheme is used in the temporal solution in which the neutron flux and the precursor density are factored into a fast varying exponential function and a smoothly varying component.

The additions to the AURORA-B code system (MB2-K and S-RELAP5) include:

- Pin power reconstruction at cold conditions.
- Inclusion of moderator temperature in cold cross section determination to support cold voided cross section feedback.
- Transfer of peak pin power from MB2-K to S-RELAP5.
- Transfer of moderator temperature from S-RELAP5 to MB2-K.
- Support a peak rod heat structure.

The peak rod heat structure and the cold pin power reconstruction allow the AURORA-B system to determine the peak pin enthalpy utilizing the realistic pin peaking determined throughout the transient. The pin power reconstruction at cold conditions is performed in the same manner as for hot operating conditions. This process takes into account the nodal surface fluxes and currents, average fluxes, and corner fluxes and flux gradients throughout the transient. The details of the pin power reconstruction are described in Reference 20 (MB2-K Theory Manual).
[

]

7.5.1 External Data Transfer

External data transfer is an important element of the procedures for treating the input information (particularly the code input arising from the assumed plant state at transient initiation). In addition, analysis of the target scenarios requires input from numerous data

sources and engineering disciplines because of the highly coupled neutronic / fuel thermal-mechanical / thermal-hydraulic nature of the events. As a result, the amount and content of data that will be transferred to the EM from external data sources is important in assuring accurate modeling of the BWR plant at the target initial conditions. A summary schematic highlighting important elements of the external data transfer is shown in Figure 7.4.

[

]

Figure 7.4 External Data Transfer to AURORA-B EM

7.5.2 Coupling of Component Calculational Devices

Coupling of the component calculational devices to create the code system is important to assure the relevant event characteristics are captured in the EM. Modeling the dynamic relationship between the neutron-kinetics, fuel rod models, and core thermal-hydraulics models is important aspect in modeling the CRDA. The coupling between component calculational devices for the system calculation is illustrated in Figure 7.5.

[

]

Figure 7.5 Coupling of Component Calculational Devices for CRDA

In the coupling, the 3D kinetics equations are simulating every fuel assembly in the core at the same level of detail used in the steady state core simulator. Modeling of the water channels/rods and the core bypass are also consistent with the modeling applied in the steady state core simulator.

Coupling from the thermal-hydraulic and thermal conduction equations to the 3D kinetics equations requires passing the active channel, core bypass, the water channel/rod moderator densities, moderator temperature, and the Doppler effective fuel temperature to the 3D kinetics [

]

[

]

7.5.3 Fuel Grouping

The CRDA scenario is a relatively localized event around the target rod. The individual fuel assemblies in either [ ] around the target rod are explicitly modeled as individual channels. Both an average fuel rod and peak power fuel rod are evaluated for the individual channels. [

] Within the buffer rings the fuel assemblies are assigned in groups with similar geometric, hydraulic, thermal power, and fuel thermal properties. The remainder of the fuel assemblies in the core are assigned to groups with similar geometric, hydraulic, thermal power, and fuel neutronic properties.

Sensitivity studies were completed with the code assessment relative to channel grouping for the CRDA and are provided in Section 8.7.1.

When developing the fuel grouping assignments, excluding the assemblies around the target rod, an algorithm is applied that ensures all assemblies in a fuel group have identical geometric design, orifice design, nuclear design (e.g. identical nuclear lattice designs), and are part of the

same reload batch. Additional characteristics such as exposure, power distribution, and/or the initial CPR predicted by MICROBURN-B2 may be used to define even smaller groups if desired.

7.5.4 Time Step Control and Advancement

The time step control and advancement are consistent with the AURORA-B base EM of Reference 9. However, due to the magnitude of the CRDA reactivity insertion at [

]

The time step sizes for the CRDA are provided in Table 7.5.

Table 7.5 Time Step Size for CRDA Evaluation

[

]

7.5.5 Plant Model Nodalization

Modeling of the plant hydraulic components and heat structures is a mathematical mapping from the physical system to the computational framework of the EM. For the CRDA analysis, a core only model is used in the AURORA-B system. The primary impact of the nodalization on the CRDA is the number of individual channels included in the core model.

In order to achieve the above, detailed technical guidance has been developed (Reference 24) to define the framework under which standardized input models are prepared for the AURORA-B EM. Details for the AURORA-B CRDA EM model are provided in Reference 65. Specifically, the purpose of the technical guidance is to establish a consistent approach for the following parameters;

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

- Material properties
- Initialization of the components and structures

The following sections summarize the key aspects of the technical guidance for nodalization of hydrodynamic components and heat structures, as well as the modeling practices for key components and processes.

7.5.5.1 Core Region

The nodalization of the fuel assemblies, flow channels, water channels/rods, core bypass region, and associated heat structures are developed in a manner consistent with the steady state core simulator and fuel design data, except for consideration of "fuel grouping" as described in Section 7.5.3 and peak rod heat structures. [

]

The core model components are identified in Table 7.6 and shown in Figure 7.6.

[

]

Figure 7.6 Core Modeling Components

Table 7.6 Description of Reactor Core Components

[

]

7.5.5.2 Plant Parameters Data

Plant parameters data are those parameters used in preparing input to the EM that describe the configuration of a given plant that could potentially change from cycle to cycle. A document defining the parameters is prepared jointly by AREVA and the utility customer for each plant and reviewed each fuel cycle. Every value is reviewed and accepted by the utility customer to ensure the most accurate data defining the plant configuration is used in AREVA analyses. This is particularly important to ensure AREVA is aware of values which have been superseded due to plant configuration changes, operation changes, or more recent information. Customer acceptance of the document validates the parameters within the document. Changes from prior cycles are typically highlighted to ensure analysts are aware of the changes.

The plant parameters data provides necessary information for the physical plant and any engineered features influencing plant performance as part of the "standard nodalization". The data are obtained from plant drawings, plant design data and specifications, component/equipment design data and specifications, plant system manuals, Technical Specifications, startup test data, plant operation data, and other similar sources.

The following list summarizes key plant parameters data used in developing input to the EM:

- Core thermal power, core flow, steam flow, dome pressure at rated conditions, and allowable power/flow map for off rated conditions.
- Safety system performance and set points including; scram set points for high neutron flux, high dome pressure, scram delays for the instrumentation, control rod insertion speeds.
- Allowed control rod sequences and inoperable control rod restrictions.

The approach for selecting specific values for plant parameters data that assure plant operations are bounded is described in Section 9.0 for the application methodology.

7.5.5.3 Individual Channel Model for CRDA

[

]

7.6 *Determination of Prompt and Total Enthalpy Response*

The specific acceptance criteria make a distinction between prompt and total enthalpy rise. If acceptable results can be obtained using total enthalpy, then total enthalpy rise may be used for the evaluation. However, if there is a significant difference between the prompt and total enthalpy, then both the prompt and total enthalpy may be used for event evaluations. The prompt enthalpy is the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse as defined on page 4.2-33 SRP4.2 Appendix B

(Reference 6). [

] of the transient is also illustrated in Figure 7.8 which contains the local peaking factors for all rods in a lattice. Each color represents a different time step in the transient. Prior to the rod drop, the lower rod numbers have higher peaking than the higher rods in the lattice. As the blade drops out the peaking flattens across the lattice. The peaking for all rods remains constant during the peak pulse at approximately 0.8 seconds. Therefore, the enthalpy rise during the power pulse is proportional to the peaking factors during the time of the peak power.

[

]

[

]

Figure 7.7 Pin Peaking Factor Data for Select Rods and Core Power

[

]

Figure 7.8 Pin Peaking Factor Distribution for All Rods with Transient Time

[

]

Figure 7.9 Enthalpy Determination

Additional information on the pin peaking factor distribution during the transient is provided in Section 8.7.3.12.

7.7 *Evaluation of Event MCPR Response*

The single channel model is applied for evaluating the fuel assembly MCPR for the [ ] assemblies around the dropped rod location. The MCPR response is obtained from the AURORA-B system during the transient. Evaluations of rod drops were performed in the power range. The evaluation consisted of determining the rod which when dropped would have the largest impact on CPR with the static calculation. All rods in one quadrant of the core were evaluated to determine the impact on CPR. If the rod was withdrawn, it was inserted and then dropped. The rods which had the largest impact on the steady state CPR were then evaluated with the AURORA-B system.

The power pulse in the power range is much broader than that in the startup range due to the presence of voids in the core as illustrated in Figure 7.10.

[

]

(Reference 11 page 3-2).

Table 7.7 Power Range Minimum CPR

[

]

Table 7.8 Rods in Boiling Transition versus CPR Safety Limit EOC

[

]

Table 7.9 Rods in Boiling Transition for SL=1.0 Near BOC

[

]

[

]

Figure 7.10 Power Pulse for CRDA in Power Range

[

]

Figure 7.11 Power Range CPR Response CRDA

7.8 *Evaluation of Event Pressure Response*

The most sever power pulse for the CRDA occurs at the cold conditions at atmospheric pressure. Therefore, a significant amount of energy is required to simply pressurize the core. Although local pressure increase occurs within a few assemblies, there is little change in the core system pressure in either the startup range or the power range.

The pressure response in the lower plenum for rod drops in the power range are provided in Figure 7.12 and the pressure response for a rod drop in the startup range is provided in Figure 7.13.

Therefore, the CRDA would not result in a reactor pressure which would cause increased stress to exceed the "Service Limit C" ASME Boiler and Pressure Vessel Code.

[

]

Figure 7.12 Power Range Lower Plenum Pressure Response

[

]

Figure 7.13 Startup Range Lower Plenum Pressure Response

7.9 *Evaluation of Radiological Consequences*

The direct determination of the radiological consequences is not addressed in this LTR. However, the method to determine the appropriate release fractions and fission product inventory is provided which can then be used in the evaluation against the source term used in the plants licensed radiological evaluation for the CRDA.

7.9.1 Nodal Rod Release Fraction

The release term for RIA includes an enthalpy dependent term. Therefore, the evaluation will be more intrusive/sophisticated than simply evaluating the number of failed rods. If a rod is determined to have failed as result of the CRDA, then the transient fission gas release term is determined for each failed rod.

Steady state release fractions will be utilized consistent with Regulatory Guidance and requirements. The steady state release terms are provided from Reference 12 and 13. For the AREVA CRDA methodology the steady state release terms are based on the appropriate steady state release term of Reference 12 and 13. The steady state gap release fractions are provided in Table 7.10. Transient enthalpy dependent release fractions from SRP 4.2 Appendix B

(Reference 6) are provided in Table 7.11. A revised set of release fractions for the CRDA event are provided in Reference 45. The total release fractions for the BWR CRDA are provided in Table 7.12 based on SRP 4.2 Appendix B and Reference 45 (PNNL-18212 Revision 1).

(It is recognized that the potential revision of the release fractions may occur as indicated in References 44 and 45. Upon adoption of revised release fractions, the AREVA methodology will incorporate release fractions consistent with the plant licensing basis. The inclusion of the values from Reference 45 are provided for example.)

The final release fraction for a failed rod is steady state release fraction and the average of the nodal transient release fractions.

$$\mathrm{RRF}_i = \mathrm{SSRF}_i + \frac{\sum TRF_i\left(\Delta \mathrm{H}_k\right)}{N} \tag{4}$$

Where;

RRF_i the rod release fraction for isotope $_i$ including steady state and transient release

$SSRF_i$ the steady state release fraction for isotope $_i$

$TRF_i\ (\Delta H)$ the enthalpy dependent transient release term for isotope $_i$

ΔH_k the enthalpy increase at node k

Table 7.10 Steady State BWR Fuel Rod Peak Gap Release Fractions

Isotope	Current RG 1.183 Table 3 (Reference 12)	Current RG 1.195 Table 2 (Reference 13)
I-131	0.08	0.08
I-132	0.05	0.05
Kr-85	0.10	0.10
Other Nobles	0.05	0.05
Other Halogens	0.05	0.05
Alkali Metals	0.12	-

NOTE: I-132 is set to the release fraction of other Halogens

Table 7.11 Transient BWR Fuel Rod Gap Release Fractions

Isotope	SRP 4.2 Appendix B (Reference 6)	
	%	Fraction
All	Max(0,0.2286*ΔH-7.1419)	Max(0,0.002286*ΔH-7.1419E-2)

Table 7.12 Local Gap Release Fractions for BWR CRDA

Isotope	SRP4-2 Appendix B	PNNL-18212 Revision 1
I-131	0.08+ max(0,0.00286*ΔH-7.1419E-2)	0.08+0.00073*ΔH
I-132	0.05+ max(0,0.00286*ΔH-7.1419E-2)	0.09+0.00073*ΔH
Kr-85	0.10+ max(0,0.00286*ΔH-7.1419E-2)	0.38+0.0022*ΔH
Other Nobles	0.05+ max(0,0.00286*ΔH-7.1419E-2)	0.08+0.00073*ΔH
Other Halogens	0.05+ max(0,0.00286*ΔH-7.1419E-2)	0.05+0.00073*ΔH
Alkali Metals RG 1.183	0.12+max(0,0.00286*ΔH-7.1419E-2)	0.50+0.0031*ΔH
Alkali Metals RG 1.195	0+max(0,0.00286*ΔH-7.1419E-2)	

7.9.2 Nodal Rod Fission Product Inventory

The nodal fission product inventory is assumed to be uniform over the length of the entire fuel rod and is determined in a manner consistent with Section 3.1 of Reg Guide 1.183 (Reference 12) or Section 3.1 of Reg Guide 1.195 (Reference 13). The fission product inventory is based on the maximum allowed power and the maximum exposure at the time of the event.

7.9.3 Fission Product Inventory Released

The total fission product inventory released is determined by the summation of the individual products of the release for each failed rod.

7.10 *Evaluation of Core Coolability*

There are four specific requirements for core coolability contained in SRP 4.2 Appendix B. This section describes how the coolability criteria are addressed and complied with in the AREVA CRDA methodology.

7.10.1 Peak Radial Average Fuel Enthalpy

The fuel enthalpy is calculated with the AURORA-B code system and the total enthalpy is compared with the 230 cal/g limit. The total enthalpy is combined with an enthalpy uncertainty for the AURORA-B code system to demonstrate the 230 cal/g limit is not exceeded.

7.10.2 Peak Fuel Temperature

The peak fuel temperature is calculated with the AURORA-B code system. The resulting temperature is then compared with the melting temperature for the UO2. The fuel melting temperature utilized for failure determination will be [ ] rods (Section 7.1.8 Figure 7.16 Reference 22).

7.10.3 Mechanical Energy

The typical power pulse width for the BWR CRDA is estimated between 45 and 75 ms for cold zero power conditions and 45 and 140 ms for hot zero power conditions (Table 1 of Reference 47). A review of the experimental data was provided in Reference 48 and 49. Table 7.13 provides a summary of test results presented in Appendix A of Reference 47. No UO2 fuel experienced dispersal for tests with pulse widths above 10 milliseconds and enthalpy deposits below 230 cal/gram. [

] See Table 3.1 Interim Criteria Summary, Page 3-2 of this document.

Table 7.13 RIA Test with Failures (Below 230 cal/g)

	Burnup	Pulse Width	Peak Enthalpy		Failure Enthalpy		Fuel Loss
Test	GWd/MT	ms	J/g	cal/g	J/g	cal/g	
PBF RIA 1-2	4.4	16	775	185	586	140	No
NSRR JM-4	21	5.5	743	177	743	177	No
NSRR JMH-5	30	6.2	910	217	790	189	Yes
NSRR JM-5	26	5.6	697	167	682	163	No
NSRR TK-2	48	4.4	448	107	251	60	Yes
NSRR HBO-1	50	4.4	306	73	251	60	Yes
NSRR HBO-5	44	4.4	335	80	322	77	Yes
NSRR TK-7	50	4.3	398	95	360	86	Yes
REP Na-8	60	75	410	98	184	44	No
NSRR FK-6	61	4.3	548	131	293	70	Yes
NSRR FK-7	61	4.3	540	129	260	62	Yes
NSRR FK-9	61	5.7	377	90	360	86	Yes
NSRR FK-10	61	5.1	430	103	335	80	Yes
NSRR FK-12	61	5.8	373	89	301	72	Yes
REP Na-10	63	31	410	98	338	81	No
REP Na-1	64	9.5	481	115	117	28	Yes
LS-1	69	4.4	469	112	222	53	Yes

7.10.4 Coolable Geometry

The design of the BWR fuel assembly and channel helps mitigate the loss of coolable geometry from the CRDA. Each BWR fuel assembly is channeled such that flow enters the bottom of the assembly and exits the top. The BWR fuel assembly includes axial spacers or grids which maintain the positions of the fuel rods within the assembly. Bypass coolant flows around the assembly channel and through the internal water rods/channels within the fuel assembly. Modern BWR assemblies utilize part length rods which provide an increased flow area in the upper portion of the fuel assembly. This increase in flow area reduces the two-phase pressure drop in the top of the assembly during power production.

The bypass flow and coolant would not be affected by the CRDA. Prior to the control rod dropping, the fuel adjacent to the control rod is either producing no power, or less than core average since it is controlled. [ ]

[

]

7.11 *Methodology Uncertainty*

The uncertainty for the AREVA CDRA methodology is determined in a manner similar to that used for the AURORA-B base EM. In that work, the treatment of biases and uncertainties was divided among three classes. The first were those related to the evaluation model structure relative to the interactions among the Component Calculations Devices (CCD's) and time step sensitivities. The second was the selection of plant parameters and initial conditions. The third and final were those involved in the prediction of highly ranked PIRT entries. The actual evaluation of the uncertainty is provided in Section 8.7.

7.11.1 Evaluation Model Structure

For the coupling between the computational tools and the time step specification, the time steps and CCD interactions are limited to those which do not vary, except in a trivial way. Thus, for example, the time step size for a particular transient is selected such that further reductions in that size results in virtually no changes (achieved convergence) in the results produced by the evaluation model. Table 7.14 identifies the model interactions that are evaluated. (The results of the model structure evaluation are provided in Section 8.7.1 of this document.)

7.11.2 Plant Parameters & Initial Conditions

Plant parameters are generally a mixture of realistic and conservative values. While the conservative values may generally be relied upon for use, the evaluation model may be exercised in sensitivity studies to determine appropriate values for parameters available as realistic values. Initial conditions are selected to be appropriate to the accident or transient, but are treated in the same fashion as realistically specified plant parameters to ensure that the results are appropriately conservative or insensitive to variations in the input values.

7.11.3 Highly Ranked PIRT Entries

For highly ranked PIRT entries sensitivity studies are performed and presented in Section 8.7.3 of this document. For these studies, the effect of the propagation of these perturbations and the resulting uncertainties are quantified. Based on these results, suitable treatments of these entries are formulated to ensure that net conservative FoM results are produced.

Table 7.14 Evaluation Model Structure Interactions

Model Element	Comment
Time step sensitivity	The time step sensitivity will be repeated for CRDA analysis.
Hydraulics to Kinetics Coupling Scheme	This represents the computational coupling between the two major Component Computational Devices for the methodology
Core Flow Distribution	The flow distribution will be evaluated for application at startup conditions.
Fuel Channel Grouping	This includes the number of fuel assemblies that are represented explicitly around the dropped rod and the grouping of the remainder of the assemblies within the core.

8.0 Assessment Results

8.1 *Regulatory Basis*

From SRP 15.0.2 II.1.A and B, a comprehensive description of the code assessment is required along with the determination of the code uncertainty for a sample plant accident calculation.

A summary of the code assessment data base is presented in Section 6.0. The results of the code assessment are provided within this Section. The code uncertainty basis are also determined within this Section while the actual determination of the uncertainty for a sample plant accident calculation is presented in Section 9.

8.2 *AURORA-B CRDA Assessment Results*

A comprehensive description of the AURORA-B base EM code system assessment is provided in ANP-10300P (Reference 9). The additional assessments of the AURORA-B system for the analysis of the CRDA are presented in this section along with the results of sensitivity studies.

The base assessment criteria were established for the AURORA-B base EM code system and are applied for the AURORA-B CRDA methodology. The primary scope of the assessment for the AURORA-B CRDA is to address modeling of the CRDA in the startup range.

8.3 *AURORA-B CRDA Model Fidelity or Accuracy Assessment*

This Model Fidelity Assessment for the AURORA-B Base EM is presented in Section 6.2 of Reference 9. Following are the assessment conclusions from Reference 9.

8.3.1 Rod Bundle Void Tests and Christensen Void Tests

The assessment data base includes test from different facilities and provides void information for a wide range of system pressures (400 to 1260 psia) and flow conditions that cover the typical operating BWR conditions. These assessments show excellent code-data agreement of the given tests for void distributions. The results are applicable for modeling phenomena associated with void generation for the CRDA in the power range. Void sensitivity studies are provided in Section 8.7.3.2 for the start-up range.

8.3.2 Summary of MICROBURN-B2 Qualification

A summary of the MICROBURN-B2 qualification is provided in Section 6.2.5 of Reference 9 for the AURORA-B system. This summary is based on the complete qualification of the NRC approved CASMO4/MICROBURN-B2 methodology (Reference 17). The MICROBURN-B2 qualification assessment summary as provided in Reference 9 is applicable for the CRDA in the power range.

In addition to the operating power conditions, the CASMO4/MICROBURN-B2 methodology includes extensive qualification against cold critical measurements for numerous reactor startups. From this it is inferred that the CASMO4/MICROBURN-B2 core simulator provides excellent prediction of [ ]

8.3.3 Summary of RODEX-4 Qualification

The summary of the RODEX-4 qualification as provided in Section 6.2.6 of Reference 9 is applicable for the CRDA in the power range. The fuel rod history effects are independent of the event initial conditions. Therefore, the assessment of the RODEX-4 codes and models remains applicable for the CRDA.

8.4 *AURORA-B CRDA Model Field Equation and Numeric Solutions Assessment*

The assessment of the field equations and numerical techniques presented in Reference 9 is directly applicable for the CRDA analysis. The numerical kinetic benchmark results from the 2-D TWIGL seed-blanket reactor problem (Reference 30), the LMW 3-D PWR delayed critical transient problem (Reference 31) and the LRA 2D and 3D BWR Control Blade Drop Transients (Reference 32) are presented in Section 6.4 of Reference 9. The results predicted by MB2-K showed excellent agreement with other published results for all of the problems.

8.5 *AURORA-B CRDA Model Applicability to Simulate System Components*

The assessment of rod bundle pressure drop and critical power test is presented in Section 6.5 of Reference 9 along with other components. The pressure drop and the critical power assessments from Reference 9 are applicable for the rod drop in the power range.

8.6 *AURORA-B CRDA Model Integral Tests*

8.6.1 Peach Bottom Turbine Trip Tests

The test results from the Peach Bottom Turbine Trip tests (Reference 37 and 39) are used to assess the global capability of the AURORA-B EM in predicting the neutron kinetics response to typical pressurization events. In addition, the Peach Bottom tests provide sufficient information for validating the capability of S-RELAP5 in predicting several phenomena in dynamic scenarios. This assessment is provided in Section 6.6.2 of Reference 9.

Direct measurement of the indicated PIRT phenomena cannot be made in the tests, so the validation is performed via measured and calculated pressure and LPRM response of the plant in three tests. The validation shows reasonable to excellent code-data comparisons of pressure and LPRM response. From this it is inferred that the EM makes excellent predictions of the indicated PIRT phenomena. In addition, the global capability of the EM to predict the events is shown to be excellent. The specific phenomena that are included in the Reference 9 assessment which are [

]

8.6.2 SPERT III RIA Tests

The Special Power Excursion Reactor Test III (SPERT III) reactor was a small pressurized-water, nuclear research reactor. The reactor was constructed as a facility for conducting reactor kinetic behavior and safety investigations. The facility description is provided in Reference 42 and the descriptions of the tests are provided in Reference 43. This section presents the results of the code assessment against the selected SPERT III experiments. The SPERT III initial experiment conditions represented cold startup, hot startup, hot standby, and hot operating conditions as listed in Table 8.1. It is noted that the average linear heat generation rate (LHGR) for the SPERT III core at operating power is close to that of commercial BWR reactors.

Table 8.1 SPERT III E-Core Tests Conditions

Range	°F	MWT	Avg Kw/ft
Cold Startup	70	5e-5	-
Hot Startup	260, 500	5e-5	-
Hot Standby	500	1	0.23
Hot Operating	500	20	4.7

The actual core configuration of the SPERT reactor is quite different than that of commercial BWRs. The SPERT core is relatively small and the core consists of only fresh fuel. The actual control rods are a combination of flux trap and fuel follower. A single transient rod is located at the center of the core which can be rapidly ejected out of the core for the reactivity insertion.

The modeling of the SPERT core required assumptions with respect to some of the components and initial conditions. The initial positions for the control rods and the transient rod are not identified for each experiment. However, critical positions were specified for static conditions along with the control rod worth and transient rod worth.

8.6.2.1 SPERT III Static Conditions

The model for the SPERT core was generated with the CASMO4/MICROBURN-B2 codes system and the static calculations were performed and compared with the static results provided in the reports (References 42, 43, and 66). Comparisons of the control rod and transient rod worth are provided in Figure 8.1 and Figure 8.2. These figures show good agreement between the MICROBURN-B2 results and the reported rod worth values.

[

]

Figure 8.1 SPERT III E Core Control Rod Worth

[

]

Figure 8.2 SPERT III E Core Transient Rod Worth

8.6.2.2 Transient Simulation

Transient simulations for SPERT test case were completed for each of the operating ranges. The initial conditions are set and the rod positions are iterated on to match the power pulse.

The nodalization for the transient cases were set to SPERT bundle size. To generate the [

] The results of the transient cases are provided in Figure 8.3 through Figure 8.9.

The results of the magnitude of the power peak and the time at which it occurred are provided in Table 8.2. All results obtained are within the published experimental uncertainties.

Table 8.2 SPERT III E-Core Tests Results

[

]

[

]

Figure 8.3 Hot Operating Case 86 ($1.17,19MW)

[

]

Figure 8.4 Hot Standby Case 82 ($1.29,1.2MW)

[

]

Figure 8.5 Hot Startup Case 56 ($1.04,50W)

[

]

Figure 8.6 Hot Startup Case 58 ($1.15,50W)

[

]

Figure 8.7 Hot Startup Case 29 ($1.10,50W)

[

]

Figure 8.8 Cold Startup Case 41 ($1.13,50W)

[

]

Figure 8.9 Cold Startup Case 43 ($1.21,50W)

Assessment Conclusions

The AURORA-B system can model the SPERT reactivity events within the uncertainty of the published experiment results. If detailed input conditions were known, the assessment would be considered an excellent agreement. Due to the lack of sufficient detail of the experiments, an explicit assessment of the model cannot be obtained. However, assessment conclusions are implied from the modeling of the SPERT experiments.

[

]

The range of temperatures and power levels evaluated are consistent with BWRs from cold shutdown to hot operating conditions. The AURORA-B codes system predicted the results of the selected SPERT-III test within the published uncertainties.

8.7 ***AURORA-B CRDA Model Biases and Uncertainties***

The determination of the AURORA-B CRDA Model Biases and Uncertainties is provided in this section following the process provided in Section 7.11. The summary of the model biases and uncertainty is provided in Section 8.7.4.

8.7.1 Evaluation Model Structure

The impact of the specific elements of the AURORA-B system for power conditions is provided in Section 6.8.2 of Reference 9. This section reviews the model impact with respect to the CRDA. The evaluation and conclusions are summarized below:

Time Step Sensitivity

[

]

Hydraulics to Kinetics Coupling Scheme

[

]

Core Flow Distribution

[

]

[

]

Fuel Channel Grouping

The reference analyses for the sample BWR-4 plant evaluated the impact of the channel grouping around the target control rod in the startup range. The grouping consisted of the explicit target array and the buffer region size. The results from the channel grouping are provided in Table 8.3 and shown graphically in Figure 8.10, Figure 8.11, and Figure 8.12. The nomenclature [

]

The core peak power decreases as the effective number of channels increases. The decrease nearly flattens at [

] channels and above.

When only [

] This generates lower fuel temperature (less Doppler feedback) and less moderator heating/voiding in these adjacent assemblies.

Based on these results [

] area is acceptable.

[

Table 8.3 Fuel Channel Grouping Results

]

[

]

Figure 8.10 Peak Core Power versus Channel Grouping

[

]

Figure 8.11 Prompt Enthalpy Rise versus Channel Grouping

[

]

Figure 8.12 Total Enthalpy Rise versus Channel Grouping

8.7.2 Plant Parameters & Initial Conditions

The impact of plant parameters and initial conditions are presented in this section.

8.7.2.1 Scram Speed

For the CRDA, the power pulse is turned around prior to the actual movement of the control blades. Therefore, use of conservative delay times and SCRAM speed is appropriate for the CRDA analysis.

8.7.2.2 Control Rod Drop Velocity

For BWRs 2 through 6, the control rod drop velocity is maintained with the control rod velocity limiter. Based on the results of velocity limiter tests, a conservative maximum rod velocity of 3.11 ft/s is used from the Reference 10 Velocity Limiter Tests. This was determined to be a bounding conservative velocity for BWRs 2 through 6. The use of the conservative drop velocity will be used for BWRs 2 through 6.

For BWRs without velocity limiters, an appropriate drop velocity must be determined for the application of this methodology.

8.7.2.3 Initial Coolant Temperature

Sensitivity studies were completed with respect to the impact of the initial coolant conditions on the CRDA figures of merit. In the startup range, the largest impact on the enthalpy FoM was observed using the cold startup conditions. Comparisons of the enthalpy response versus temperature were made for two rod drops one rod at peak reactivity and the other at end of full power conditions.

The end of cycle rod drop 12 was used in the sensitivity study consistent with the other sensitivity studies. The results for fuel group 3 of rod drop 12 are shown in Figure 8.13 and Table 8.4. The rod 127 drop is at peak reactivity and the response is similar to that of rod 12 as indicated in Figure 8.14.

For both rods, the largest enthalpy response occurred starting from the lower temperature conditions. Utilization of 68°F for the initial temperature provides bounding conditions for higher temperatures.

[

]

Figure 8.13 Peak Enthalpy Response Rod 12 at End of Full Power

Table 8.4 Core Initial Temperature Sensitivity Rod 12 EOFP

[

]

[

]

Figure 8.14 Peak Enthalpy Response Rod 127 at Peak Reactivity

8.7.2.4 Initial Core Power Level

Sensitivity studies were performed with respect to the power level. Relative to the reference case, the magnitude of the event decreased as the initial power was increased.

Table 8.5 Core Initial Power Sensitivity

[

]

8.7.2.5 Initial Core Flow

Sensitivity studies were performed with respect to the coolant flow rate and the results are provided in Table 8.6. As the core initial core flow increase the peak power and prompt enthalpy rise decrease. The total enthalpy rise increases as the initial core flow increases. The typical startup conditions for BWR plants are carried out with both recirculation pumps at minimum speed (Reference 60 page 6.2.6, Reference 61 Section 4.4.3.1.2 Page 4.4-11, Reference 62 Section 4.4.3.3.1 Page 4.4-10). With the recirculation pumps at minimum pump speed the core flow would be greater than 10%. Therefore, use of an initial core flow of 10% is appropriate for the determination of the prompt enthalpy rise for startup conditions.

Table 8.6 Core Initial Flow Sensitivity

[

]

8.7.2.6 Fuel Rod Power History

The AURORA-B methodology utilizes nodal average powers in constructing the rod power history effects. To assess the impact of this, rod power histories were interchanged for six fuel channels as indicated in Table 8.7. The core contained three reload batches. The sensitivity to the rod power history was evaluated by switching the power history profiles for different channels. A summary of modified gap properties are provided in Table 8.8.

The impact of the exchange of the power history files is provided in Table 8.9

Table 8.7 Power History Exchanges

Channel	Power History Modification
1	2nd cycle power history replaced with 3rd cycle power history
2	2nd cycle power history replaced with 1st cycle power history
3	1st cycle power history replaced with 3rd cycle power history
9	3rd cycle power history replaced with 2nd cycle power history
10	3rd cycle power history replaced with 1st cycle power history
11	1st cycle power history replaced with 2nd cycle power history

Table 8.8 Summary of Gap Properties

[

]

An increase in the prompt enthalpy rise can occur when the power history for lower exposed assembly is used for that of a higher exposed assembly. This indicates that the use of lower exposed and correspondingly lower powered rod history will produce a larger enthalpy rise. [

] for the evaluation of the CRDA. The impact on the enthalpy responses are provided in Figure 8.15, Figure 8.16, and Figure 8.17.

Table 8.9 Power History Exchange Results

[

]

[

]

Figure 8.15 Fuel Group 1 and 2 Enthalpy Sensitivity to Power History

[

]

Figure 8.16 Fuel Group 3 and 9 Enthalpy Sensitivity to Power History

[

]

Figure 8.17 Fuel Group 10 and 11 Enthalpy Sensitivity to Power History

8.7.2.7 Allowed Rod Withdrawal Sequence

The plant specific rod withdrawal sequence affects the maximum dropped rod worth. Each plant utilizes a specific control rod pattern sequence control system which minimizes the worth of the control rods for the CRDA. The Banked Position Withdrawal Sequence (Reference 15) is typically used for controlling the allowed startup sequences. The analyzed withdrawal sequence must be consistent with the plant limitations. To conservatively bound rods in higher order groups, the pull sequence can be analyzed with each higher order group being the second group pulled.

8.7.2.8 Inoperable Control Rods

Each plant's Technical Specifications allow for inoperable control rods. Since the actual location of the inoperable control rods would not be known at the time of core licensing process, a conservative assignment of the inoperable control rods is assigned for a plant. Assigning the inoperable control rod to one half of the core resulted in the largest increase in rod worth for the remaining control rods.

8.7.3 Analysis of Biases and Uncertainties from Highly Ranked PIRT Phenomena

The impact of propagating the biases and/or uncertainties of the highly ranked PIRT phenomena through the EM was investigated through a series of sensitivity cases.

8.7.3.1 Control Rod Worth

The actual worth of a control rod is dependent on the actual control blade design, the fuel cycle design, cycle lifetime, and initial conditions. The methodology assumes that the rod drop is always the first rod in a group. This is conservative in that the worth of the dropped rod is increased by assuming it is the first rod to drop in a group. The rod worths were tabulated and ranked for each potential withdrawal sequences through banking of the third group at notch 12. The results shown in Table 8.10 show that the highest worth rod for a given sequence is more [ ] the rod drops in the sequence. Eight different pull sequences were evaluated, four for the A sequence rods and four for the B sequence rods. The sequence identifier indicates if it is an A or B sequence and then the order in which the rod groups are pulled. (See Figure 8.18 for an illustration of all the rod pulls and their respective worths for one sequence.) Utilization of the highest worth rods ensures that the potential rod drops are bounded for criteria which are not burn up dependent.

Table 8.10 Rod Worth Ranking with Third Group Banked at 4, 8, and 12

[

]

[

]

Figure 8.18 Control Rod Drop Worth for Sequence A1234

The actual uncertainty of the cold rod worth is determined in the following manner to address the actual blade worth uncertainty, and the modeling of the blade worth uncertainty.

Control blades designed by both Westinghouse and GEH are designed to match original equipment blades within ±5% (Reference 51 page 7-3 Section 7.2.4, and Reference 52 page 10 Section 3.2.1). As such the uncertainty for the blade worth is 5%. As the control blade is irradiated, there is uncertainty associated with the depletion of the absorber material. For the CRDA, only beginning of life (maximum) blade worth is used. Therefore, no uncertainty is assigned to control blade depletion for the CRDA.

A modeling uncertainty for control blade worth was established in Reference 17 [

] The control blade worth uncertainty is summarized in Table 8.11.

Table 8.11 Control Blade Worth Uncertainty

[

]

The evaluation of the impact on the enthalpy response due to a change in rod worth was evaluated. Figure 8.19 shows the relation between fuel enthalpy and static rod worth to be fairly linear, consistent with Reference 46. [

]

Data were then evaluated over additional exposure points and temperatures and are shown in Figure 8.21 [

]

[

]

Figure 8.19 Peak Enthalpy versus Control Rod Worth EOFP

[

]

Figure 8.20 Relative Enthalpy versus Relative Rod Worth EOFP

[

]

Figure 8.21 Relative Enthalpy versus Relative Rod Worth

8.7.3.2 Moderator Feedback

For the CRDA, moderator feedback is the change in the reactivity feedback from moderator temperature and density changes in the active, by-pass and water channels/rods. This also includes the changes in void fractions in the coolant. These changes are a result of direct energy deposition to the coolant and heat transfer from the cladding. Both conditions are specifically modeled in the 3-d kinetics code with thermal hydraulics feedback.

The moderator feedback is a fundamental part of the cross sections. The sensitivity to the moderator density change during the transient was evaluated by increasing and decreasing the density feedback for both the active channel flow and the bypass flow. The results of the sensitivity studies (Table 8.12 and Table 8.13) show that the variation in the moderator feedback has insignificant impact on the prompt fuel enthalpy rise. The moderator feedback does have a minimal effect on the total enthalpy rise. Based on the results of the sensitivity studies an uncertainty of [ ] on the enthalpy is used for the moderator feedback.

Table 8.12 Active Channel Moderator Density Feedback

[

]

Table 8.13 Bypass Channel Moderator Density Feedback

[

]

8.7.3.3 Fuel Temperature Feedback

Fuel temperature feedback is the reactivity feedback from fuel temperature changes. The coupling of the kinetics and the thermal hydraulics codes results in the use of fuel temperature determined on the average rod rather than the peak rod for the given lattice. A [ ] perturbation is utilized for the Doppler feedback based on Reference 56 and 63. The [ ] perturbation on Doppler reactivity feedback is chosen [ ]. The impact of the CRDA is approximately inversely proportional to the Doppler feedback as indicated in Table 8.14. The sensitivity study shows that Doppler feedback is very important for the CRDA event and that the change in enthalpy is consistent or slightly more than the change in the Doppler feedback. An uncertainty of [ ] will be used for the enthalpy rise based on the Doppler uncertainty.

Table 8.14 Fuel Temperature Feedback

[

]

8.7.3.4 Effective Delayed-Neutron Fraction

The total delayed neutron fraction (β_{eff}) determines when prompt criticality is reached. The effective delayed neutron fraction varies depending on the heavy nuclide inventory or the nodal exposure. The neutronics data libraries generated, for each lattice represented in the AURORA-B system contain the delayed neutron fraction data for six groups tabulated as a function of exposure and spectral history. A node-specific delayed neutron fraction is obtained for each node by interpolating on the table with available nodal exposure and spectral history.

A sensitivity study was performed by modifying the cross section data libraries to increase and decrease the delayed neutron fraction. The effective delayed neutron fraction was increased and decreased by [ ] A reference transit calculation was run and the impact on the peak enthalpy is inversely proportional to the change in the total delayed neutron fraction as shown in Table 8.15. An uncertainty of [ ] is utilized for the total delayed neutron fraction consistent with Reference 50 (page 7-2 Section 7.1.4). Based on the sensitivity study, an uncertainty of [ ] is determined for the enthalpy rise.

Table 8.15 Delayed Neutron Fraction Sensitivity

[

]

8.7.3.5 Reactor Trip Reactivity

Reactor trip reactivity is the negative reactivity associated with insertion of control rods after receipt of a reactor trip signal and the rate at which it is inserted. Since the actual power pulse is turned around by Doppler prior to the start of motion of the rods being inserted, the rate of SCRAM and delay time are based on minimum Technical Specifications requirements.

With the exception of the dropped rod, all other rods are inserted by the scram and the core is in a sub-critical state which terminates the event.

By utilizing the minimum technical specification scram times, no uncertainty is assigned to the reactor trip reactivity.

8.7.3.6 Fuel Cycle Design

The fuel cycle design includes those important design elements that determine the neutronic properties of the core at event initiation, such as the loading pattern, control history (control rod pattern, spectral shift), burnup, and exposure. The impact of the fuel cycle design is the key parameter in determining the control rod worth. Therefore, [

]

8.7.3.7 Heat Resistances in High Burnup Fuel, Gap, and Cladding

The resistances offered by the fuel, gap, and cladding to the flow of thermal energy from regions of high temperature to regions of lower temperature. The resistance is dependent upon the path length and thermal conductivity, which change with burnup.

Sensitivity calculations were performed for fuel thermal mechanical properties. The calculations were performed with adjusting the gap alone, and adjusting the gap coefficient. The gap width was increased and decreased to show the impact on heat transfer as indicated in Table 8.16. Sensitivity studies on the fuel conductivity were also performed by enhancing and reducing the conductivity by 20% as shown in Table 8.17. The fuel thermal conductivity uncertainty from the MATPRO model (Reference 55) is less than 10%. The actual impact on the fuel enthalpy is minimal and an uncertainty of [ ] is assigned to the prompt fuel enthalpy rise for heat resistances based on the results of the sensitivity studies.

Table 8.16 Gap Width Adjustments

[

]

Table 8.17 Fuel Heat Transfer Coefficient Sensitivity

[

]

8.7.3.8 Transient Cladding-to-Coolant Heat Transfer Coefficient

The correlation that determines transport of energy at the interface by one or more of the following modes: forced convection-liquid, nucleate boiling, transition boiling, film boiling, or forced convection-vapor.

In the startup range, the rapid nature of the power burst, there is limited change in heat transfer during the power burst. In this range the failure criterion is enthalpy based. The heat transfer has little impact on the fuel enthalpy since the event is turned around by Doppler prior to any significant heat transfer from the rod.

Use of steady state heat transfer correlations have been shown to be conservative or appropriate for transient conditions (References 33, 34, 35, 53, and 54). Therefore, no uncertainty is applied for the transient heat transfer coefficient.

Reference	Location within Reference
33	Last part of Section 5.0 of SER top of Page 6
34	Page 60 Response to 19, ANP-10249Q1P
35	Page 8 of SER Section 3.3.4 and additional information which discuses References 53 and 54 is contained on Page 13 of ANP-10298Q3P Item 8 Round 2 RAI question 7.
53	Chapter IIA addresses use of steady-state correlations. Page 11: "Comparison with flow and power transient experiments indicated that the prediction was quite conservative."
54	Section 9.6.2, Page 405, addresses critical heat flux under transient conditions.

8.7.3.9 Heat capacities of Fuel and Cladding

The heat capacity of UO2 and cladding are established based on the RODEX4 fuel properties established for the fuel rods. The temperature change during the prompt pulse is based on the deposited energy and the heat capacity. The variation of heat capacity of the UO2 is only a function of temperature. The [ ] which is considered a standard for defining heat capacity for UO2. No error estimate or special treatment is used for the UO2 heat capacity.

The actual heat capacity of the cladding does not impact the initial prompt power pulse since the event is turned around by Doppler prior to any significant heat transfer from the rod. Therefore, no error estimate or special treatment is used for the cladding heat capacity.

8.7.3.10 Direct Energy Deposition to Moderator

Direct energy deposition into the active channel fluid occurs as the fluid attenuates γ-rays and neutrons that escape the fuel rods. The amount of energy deposited in the fluid is a function of the local fluid density and locally generated fission power of the fuel. The partitioning of the energy deposition which is deposited directly in the fuel pellet and that deposited in the moderator and structural components may affect the reactivity feedback mechanism.

Sensitivity studies were performed with respect to the energy partitioning between fuel and moderator. The fraction of power deposited in the coolant and structural material was increased and decreased by modifying the energy deposition fractions in the cross section libraries. The energy deposition fractions to the coolant/moderator were increased and decreased by 20% in the cross section libraries. The CRDA evaluations were repeated with the revised cross section libraries. The results are provided in Table 8.18 The change in energy deposition to the moderator has little impact on the actual fuel enthalpy. An uncertainty value of [ ] is assigned for the impact on the peak enthalpy for this parameter.

Table 8.18 Heat Deposition Sensitivity

[

]

8.7.3.11 Pellet Radial Power Distribution

The radial distribution of the power produced in the fuel rod is determined with the RODEX4 (Reference 21) modules in the S-RELAP5 code system. Sixteen radial regions are utilized to predict the power and temperature profile for the pellet.

Sensitivity studies were performed varying the coefficients for determining the Doppler effective temperature. Variations in the temperature weighting coefficients were performed to assess the impact on the effective Doppler temperature and the results are provided in Table 8.19. The default weighting for the Doppler effective temperature, [ ] are used as the reference in the sensitivity study.

Based on the results of the sensitivity study a [ ] uncertainty is assigned to the enthalpy rise.

Table 8.19 Doppler Effective Temperature Coefficient Sensitivity

[

]

A Pellet average temperature weighting factor

B Pellet surface temperature weighting factor

C Pellet centerline temperature weighting factor

8.7.3.12 Rod-Peaking Factors

Rod-peaking factors represent the nodal rod or pellet power distribution within an assembly during the transient. The MB2-K code utilizes pin power reconstruction to update the local rod peaking factors for each time step of the transient. As the rod is withdrawn there will be a significant change in the pin peaking factors from the initial controlled state. However, the actual peaking factors are fairly constant during the power pulse as illustrated in Figure 7.7 and Figure 7.8. The inclusion of the pin power reconstruction based on the flux reconstruction model provides an accurate estimate of local peaking during the transient.

The assessment of the methodology pin power reconstruction and pin peaking factor determination is included within Reference 17 (CASMO/MICROBURN-B2 Methodology). The same methodology for pin power reconstruction is employed within the MB2-K system. The local peaking uncertainty was determined based on [

]

[ ] (from Reference 17 page 2-6) is applicable for the control rod drop evaluation since the same methodology is used to construct the pin power distribution.

8.7.3.13 Cladding Hydrogen Content

The content of hydrogen in the cladding at the start of the transient is used in the determination of the failure criteria. This does not have an impact on the actual kinetic response, but impacts the fuel cladding failure threshold for PCMI. The hydrogen model is a best-estimate nodal hydrogen model and no uncertainty is applied. From page 12 of Reference 49, "a best-estimate nodal hydrogen concentration is judged sufficient to address the local cladding properties."

8.7.3.14 Power Distribution

The total core power and core power distribution at the start of the event. The power distribution during the transient is driven by the local rod worth and fuel characteristics. The nodal power uncertainty [ ] The nodal power distribution is reflected in the nodal isotopic concentration which is used in the determination of the cross sections for the transient calculation. Therefore, the CASMO4/MICROBURN-B2 power distribution uncertainty is applied for the rod drop.

The initial power level uncertainty is addressed in the startup range by selecting conservative initial conditions (Section 8.7.2.4). As such there is no specific uncertainty on total core power.

8.7.3.15 Coolant Initial Conditions

The coolant initial conditions includes pressure, temperature and the flow distribution, active channel, bypass, and water rod flows at the start of the transient. Conservative parameters are selected for the initial coolant conditions as described in Section 8.7.2. Therefore, there is no uncertainty term attributed to the coolant initial conditions.

8.7.3.16 Fuel Rod Internal Pressure

The fuel rod internal pressure relative to the system pressure is used to determine the high temperature failure criteria. This parameter is of primary importance if the less restrictive high temperature criteria are applied.

In the power range, the system pressure exceeds that of the rod internal pressure for low burnup fuel. The approved RODEX-4 is used for determining the rod internal pressure if the less restrictive high temperature criteria is applied. The RODEX-4 is an approved rod mechanical model which predicts the internal rod gas pressurization.

8.7.4 Uncertainty Summary

The uncertainties determined in the prior Sections are combined to provide the enthalpy rise uncertainty. The uncertainties are identified in Table 8.20. [

]

Table 8.20 Uncertainty Summary for Enthalpy

[

]

9.0 Evaluation Model Implementation

9.1 *Regulatory Basis*

From SRP 15.0.2 II.1.A, D, and F; the documentation requirements include an overview of the evaluation model, the determination of code uncertainty for a sample plant, and instructions and guidance for the code usage and limitations.

This section presents an example plant analysis of the CRDA with the AURORA-B methodology.

9.2 *Steady State Evaluations Candidate Rod Selection*

The AREVA CRDA methodology is demonstrated on a BWR4 reactor with 764 assemblies, 2 year cycles and EPU.

The sample plant adheres to BPWS for rod pattern control during startup. The A and B sequence groups defined for the sample plant are given in Figure 9.1 and Figure 9.2. For this sample plant, all BPWS allowed pull orders are supported.

9.2.1 Initial Conditons

The determination of the conservative initial conditions is performed for the first application of the methodology for a given plant class, reload size, and fuel product line. The evaluation of initial conditions from Section 8.7.2 is used for this sample application. The initial conditions as determined in Section 8.7.2 are shown in Table 9.2.

9.2.2 Inoperable Rod Positions

A maximum of 8 inoperable rods are allowed for this plant. [

] The inoperable rod locations are indicated in Figure 9.1 and Figure 9.2.

9.2.3 Group Critical Position

The first step is to determine the end of group or bank position k-effective values to determine where criticality is anticipated to occur for the given rod withdrawal sequence. Figure 9.3, Figure 9.4, and Figure 9.5 show the calculated k-effective values at the end of groups 1 through 4 for an A and B sequence withdrawal sequence.

[

]

9.2.4 Determination of Static Rod Worth

Based on the results of the group worth, static rod worths were then determined. The highest worth rods for each cycle exposure are selected for further evaluation.

For EOFP more rods are selected due to the increased cladding hydrogen content, the increased rod worth, and the top peaked power shape. [

]

Table 9.1 Candidate Rod for Transient Evaluation

[

]

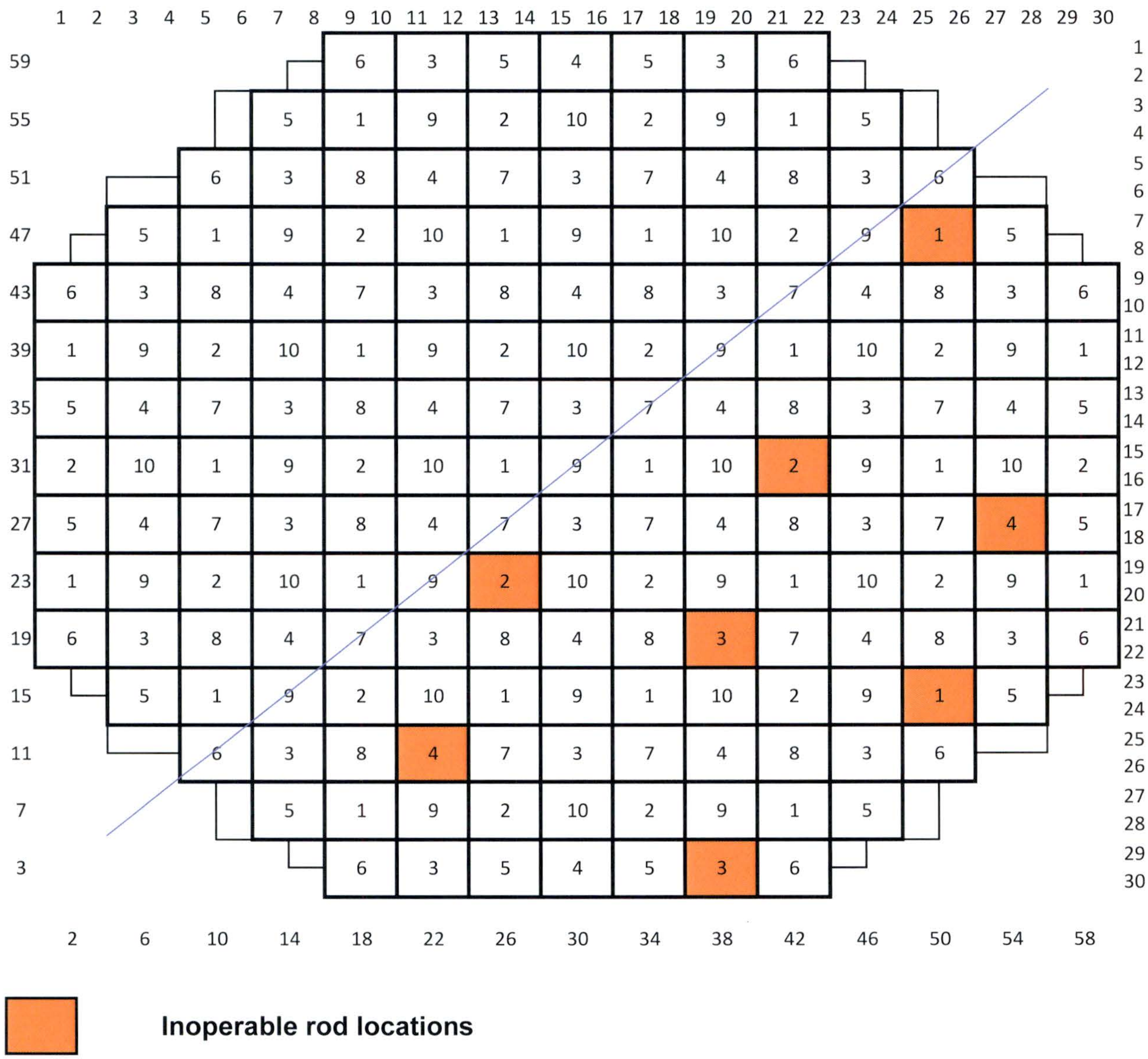

Figure 9.1 Sample Plant A-Sequence Rod Groups

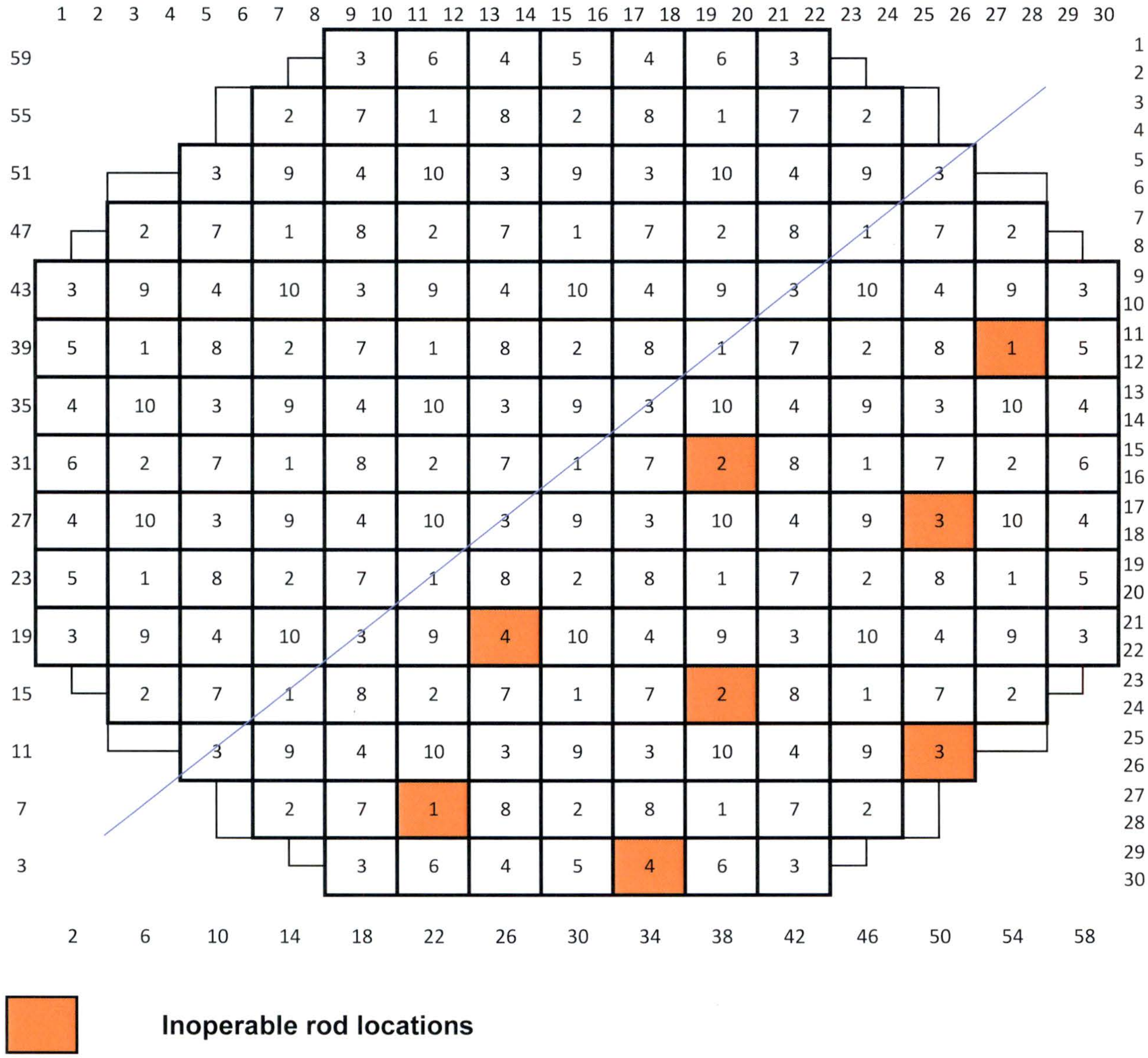

Figure 9.2 Sample Plant B-Sequence Rod Groups

[

]

Figure 9.3 BOC K-Effective for A and B Sequences Groups 1 Through 4

[

]

Figure 9.4 PEAK K-Effective for A and B Sequences Groups 1 Through 4

[

]

Figure 9.5 EOFP K-Effective for A and B Sequences Groups 1 Through 4

9.3 *Transient Evaluation*

The evaluation of each rod drop is performed with the AURORA-B system. The initial pre-rod drop state point is established with the MICROBURN-B2 core simulator. The initial conditions used for the transient calculation are identified in Table 9.2.

Table 9.2 Initial Conditions

Parameter	Value
Core Power	0.001 MW
Core Flow	10%
Core Temperature, coolant and fuel	68°F
Core Pressure	15 psia

The channel grouping with a [ ] is used for this sample plant. Once the channel grouping is defined, the power history information is processed to obtain the fuel rod characteristics for use in the RODEX-4 rod mechanical models.

The rod drops were performed and the maximum prompt and total enthalpy rise determined for the drops are provided in Table 9.3. The reported enthalpy includes the maximum average rod

enthalpy increase (both prompt and total) as well as the peak rod enthalpy increase (both prompt and total). The maximum enthalpy increase with an [

] for this sample evaluation.)

Table 9.3 Maximum Enthalpy Rise for Sample Rod Drops

[

]

Table 9.4 Maximum Enthalpy Rise for Sample Rod Drops with Uncertainty Multiplier

[

]

The first letter of the sequence identifier indicates "A" or "B" sequence and the last four digits indicate the group pull order.

Since the PCMI failure criterion is dependent on the hydrogen content, a detailed evaluation of all nodal enthalpy increases and rod nodal hydrogen content is required. The result of this evaluation is provided in Figure 9.6 and is based only on the peak rod nodal enthalpy increase [ ]. The same data provided is also provided in Figure 9.7 segregated based on reload.

For this sample evaluation there are no rod failures identified. Therefore, no radiological consequence would be required.

[

]

Figure 9.6 Prompt Enthalpy versus Cladding Hydrogen Content

[

]

Figure 9.7 EOFP Prompt Enthalpy Rise by Reload

9.4 *Evaluation Against Failure Criteria*

9.4.1 Fuel Cladding Failure

9.4.1.1 High Temperature Criteria

The total enthalpy of the peak rod is compared against the high temperature criteria as indicated in Figure 9.8. The high temperature enthalpy criterion is met.

[

]

Figure 9.8 Total Enthalpy versus High Temperature Failure Threshold

9.4.1.2 PCMI Cladding Failure

The evaluation of PCMI cladding failure is evaluated in Section 9.3 and comparison to the failure threshold is provided in Figure 9.6. The prompt enthalpy did not exceed the PCMI failure criteria.

9.4.2 Core Coolabilty

Since no fuel failures are identified in this sample calculation the four elements of the core coolability criteria are met:

- Peak radial average enthalpy <230cal/g
- Below incipient fuel Melting conditions

- Reactor pressure boundary, reactor internals, and fuel assembly structural integrity is maintained
- No loss of coolable geometry

However, the following discussion is provided.

9.4.2.1 Peak Radial Enthalpy

Evaluation of the high temperature criteria, Figure 9.8, shows that the peak radial enthalpy remains below 230 cal/gram.

9.4.2.2 Peak Fuel Temperature

The fuel temperature was evaluated for the event based on the node with the highest enthalpy. The mesh point temperatures of the peak rod of the node with the highest enthalpy are provided in Figure 9.9, Figure 9.10, and Figure 9.11. The maximum temperature is [

] This is well below incipient fuel melting.

[

]

Figure 9.9 Mesh Point Temperatures Across Peak Rod

[

]

Figure 9.10 Central Mesh Point Temperatures for Peak Rod

[

]

Figure 9.11 Outer Mesh Point Temperatures for Peak Rod

9.4.2.3 Mechanical Energy

The power pulse width for the rod drop was evaluated. [

] that reactor pressure boundary, reactor internals and fuel assembly structural integrity are maintained.

[

]

Figure 9.12 Rod Drop 118 Power Pulse

9.4.2.4 Cool-able Geometry

The cladding temperature for the peak fuel rod is shown in Figure 9.13 and it remains below 2200°F. [ ] coolable geometry is maintained.

[ ]

Figure 9.13 Cladding Temperature for Peak Rod

9.5 *Example Radiological Evaluation*

The PCMI failure criterion was reduced for the evaluation provided in Section 9.3 to demonstrate the radiological evaluation when failed rods are identified. The reduced failure threshold for demonstration of the Radiological Evaluation is provided in Figure 9.14. With the reduced failure threshold, two assemblies indicated fuel failures as identified in Table 9.5.

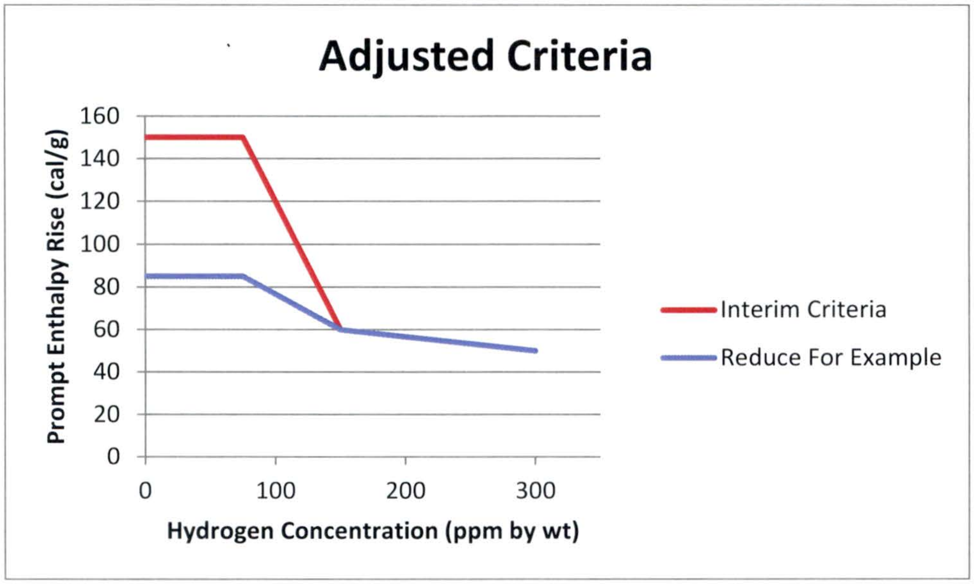

Figure 9.14 Adjusted Failure Criteria for Example Evaluation

Table 9.5 Assemblies with Failed Rods

Exposure	Sequence	Rod	Assemblies with Failures
EOFP	AG1234	118	V17118, V17134

In this example evaluation, a simplified assumption is made that all rods are at the maximum rod nodal hydrogen content rather than utilizing the individual pin hydrogen concentration values.

A fuel pin is determined to fail if it exceeds its failure threshold at any axial level.

$$PINF_i = \begin{cases} 1 & \text{If } \left(\Delta h_{i,k} > \Delta hf_{i,k}\right) \text{for all k} \\ 0 & \end{cases} \quad (5)$$

Where $PINF_i$ is a fuel rod failure flag

$\Delta h_{i,k}$ is the enthalpy rise in fuel rod i at axial level k (with uncertainty multiplier)

$\Delta hf_{i,k}$ is the enthalpy failure threshold for fuel rod i at axial level k

The total number of fuel rods failed (NFAIL) in an assembly is then the summation of the individual fuel rod failure flags.

$$NFAIL = \sum_{L=1}^{npin} PIN_i$$

Table 9.6 Determination of Rod Failures

[

]

The enthalpy dependent release fractions are determined based on the deposited enthalpy as determined directly from Equation 3.

The example determination of release fractions is based on both SRP 4.2 and PNNL-18212 Revision 1.

Table 9.7 Fission Gas Release Fraction (SRP 4.2)

[

]

Table 9.8 Fission Gas Release Fraction (PNNL-18212 Rev. 1)

[

]

For each assembly, the enthalpy dependent release terms are determined along with the ratio of the release fractions to the Reg Guide 1.183 release fractions for the rod drop.

Table 9.9 Fission Gas Release Fraction Ratios

[

]

The number of failed rods are tabulated for those assemblies which have failed rods indicated. This number of failed rods is then multiplied by the ratio of the transient release fraction to the reference release fraction in order to establish an equivalent number of failed rods. The maximum ratio of all groups (I-131, I-132, Kr85, Nobles, Halogens, Alkali Metals) is used to determine the equivalent number of rods.

For assembly V17118, 19 rods are identified as exceeding the reduced failure threshold. Utilizing the ratios of the release fractions, the actual number of failed rods is determined using the two different transient release fractions. The maximum ratio of release fraction to the reference release fraction is [ ] based on other Halogens using the steady state release fractions of Table 3 of RG1.183 and the transient release term from SRP4.2 Appendix B. The maximum ratio of release fraction to the reference release fraction is [ ] for Alkali Metals using the transient release term from PNNL-18212 Rev1.

Utilizing the release fraction ratios, an equivalent number of rod failures is determined for the two different transient release terms. The same process is repeated for each of the assemblies for which rod failures are identified for the given rod drop. Table 9.10 contains the total equivalent rod failures for all bundles for the example drop.

The approach of using the maximum ratio implies that release isotope groups are increased by this ratio. No additional radiological evaluation is required past demonstrating that the release term is bounded by that used within the radiological evaluation. For this sample plant 2000 rod failures are utilized in the source term for the CRDA. Therefore, in this sample evaluation, the dose would not exceed the licensing basis.

Table 9.10 Equivalent Rod Failures

[

]

10.0 Quality Assurance Program

10.1 *Regulatory Basis*

From SRP 15.0.2 II.1.H, a description of the Quality Assurance program under which the evaluation model was developed and assessed, and the corrective action program that will be used to address error that might be discovered is required.

10.2 *AREVA QA Program*

The quality assurance plan is not discussed within the topical report. The AREVA quality assurance plan is contained in Reference 64.

11.0 References

1. NUREG-0800, Section 15.4.9, Revision 3. "SPECTRUM OF ROD DROP ACCIDENTS (BWR)." *Standard Review Plan: LWR Edition,* US NRC: Washington, DC. March 2007.

2. 10 CFR 50. "General Design Criteria for Nuclear Power Plants." *United States Code of Federal Regulations,* Part 10 (Energy), Section 50.

3. 10 CFR 100, "Reactor Site Criteria," *United States Code of Federal Regulations,* Part 10 (Energy), Section 100.

4. NUREG-0800, Section 15.0, Revision 3, "INTRODUCTION—TRANSIENT AND ACCIDENT ANALYSES," *Standard Review Plan: LWR Edition,* US NRC: Washington, DC, March 2007.

5. ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Power Plant Components,* American Society of Mechanical Engineers.

6. NUREG-0800, Section 4.2, Revision 3. "FUEL SYSTEM DESIGN." *Standard Review Plan: LWR Edition.* US NRC: Washington, DC. March 2007.

7. NUREG-0800, Section 15.0.2, Revision 0. "REVIEW OF TRANSIENT AND ACCIDENT ANALYSIS METHODS." *Standard Review Plan: LWR Edition.* US NRC: Washington, DC. March 2007.

8. Regulatory Guide 1.203. *Transient and Accident Analysis Methods.* US NRC: Washington, DC. December 2005.

9. ANP-10300P Revision 0. *AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,* AREVA NP, December 2009.

10. NEDO-10527. *ROD DROP ACCIDENT ANALYSIS FOR LARGE BOILING WATER REACTORS.* Class I. General Electric: San Jose, CA.

11. NEDO-10527, Supplement I. *ROD DROP ACCIDENT ANALYSIS FOR LARGE BOILING WATER REACTORS.* Class I. General Electric: San Jose, CA.

12. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.* US NRC: Washington, DC. July 2000.

13. Regulatory Guide 1.195, *Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-water Nuclear Power Reactors, US NRC: Washington DC. May 2003.*

14. ANP-2829P Revision 0, *General BWR Design and Event Descriptions,* AREVA NP December 2009.

15. NEDO-21231. *BANKED POSITION WITHDRAWAL SEQUENCE.* Class I. General Electric: San Jose, CA. January 1977.

16. NUREG/CR-6742, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," September 2001.

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

18. EMF-2103(P)(A) Revision 0, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, Framatome ANP, April 2003.

19. FS1-0009406 Revision 3.0, *FSQA-08 S-RELAP5 Models and Correlations Code Manual (Theory),* AREVA Inc., March 2014.

20. FS1-0008073 Revision 2.0, FSQA-08-MB2-K Theory Manual: *A Code for Advanced Neutron Kinetics Method for BWR Transient Analysis*, AREVA Inc., March 2014.

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

22. EMF-2994(P) Revision 6, *RODEX4: Thermal-Mechanical Fuel Rod Performance Code Theory Manual*, AREVA NP, February 2012.

23. NUREG/CR-5249, *Quantifying Reactor Safety Margins*, Nuclear Regulatory Commission, December 1989.

24. 51-9079012-000, "Guidelines for Input Development and Problem Execution for Analysis of BWR Events," AREVA NP, November 20, 2009.

25. ANP-2831P, *Phenomenon Identification and Ranking for BWR Events*, AREVA NP, December 2009.

26. XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.

27. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors,* Advanced Nuclear Fuels Corporation, November 1990.

28. ANP-10307PA Revision 0, A*REVA MCPR Safety Limit Methodology for Boiling Water Reactors*, AREVA NP, June 2011.

29. EMF-2102(P), S-RELAP5: Code Verification and Validation, Framatome ANP, August 2001.

30. L. A. Hageman and J. B. Yasinsky, *Comparison of Alternating-Direction Time-Differencing Methods with Other Implicit Methods for the Solution of the Neutron Group-Diffusion Equations*, Nuclear Science and Engineering, 38, 8-32 (1969).

31. S. Langenbuch, W. Maurer, and W. Werner (LMW), *Coarse-Mesh Flux-Expansion Method for the Analysis of Space-Time Effects in Large Light Water Reactor Cores*, Nuclear Science and Engineering, 63, 437-456 (1977).

32. *Argonne Code Center: Benchmark Problem Book*, ANL-7416, Supplement 2 (1977).

33. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

34. ANP-10249PA Revision 1, *ACE/ATRIUM-10 Critical Power Correlation*, AREVA NP, September 2009.

35. ANP-10298(P)(A) Revision 0, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, March 2010.

36. Nuclear Energy Agency, *Boiling Water Reactor Turbine Trip (TT) Benchmark,* Volume I: Final Specifications. Technical Report NEA/NSC/DOC(2001)1, Nuclear Energy Agency, October 2001. Revision 1.

37. Electric Power Research Institute, *Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2.* Technical Report NP-564, EPRI, June 1978.

38. NUREG/CR-4128, *BWR Full Integral Simulation Test: Phase 2 Test Results and TRAC-BWR Model Qualification*, U.S. Nuclear Regulatory Commission, June 1985. (Also issued as EPRI NP-3988 and GEAP-30876).

39. Electric Power Research Institute, *Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2*, Technical Report EPRI NP-563, EPRI, June 1978.

40. NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," General Electric, March 1972 (NRC ADAMS Accession Number ML010870249).

41. XN-NF-80-19(P)(A) Volume 1 and Supplements 1&2, *Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, Exxon Nucle*ar Company, March 1983.

42. IDO-17036, "SPERT III Reactor Facility: E-Core Revision," AEC Research and Development Report, November 1965, (NRC ADAMS Accession Number ML080320408).

43. IDO-17281, "Reactivity Accident Test Results and Analyses for the SPERT III E-Core – A Small, Oxide-Fueled, Pressurized-Water Reactor", AEC Research and Development Report, March 1969 (NRC ADAMS Accession Number ML080320431).

44. DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Proposed Revision 1 of Regulatory Guide 1.183), US NRC, October 2009.

45. PNNL-18212, Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard, Pacific Northwest National Laboratory, Revision 1, June 2011. (NRC ADAMS Accession Number ML112070118)

46. Diamond, D.J., "Analyzing the Rod Drop Accident in a BWR with High Burnup Fuel," BNL-NUREG-63663, Rev. 2/97, CONF-970315—1-REV., OSTI ID: 465206.

47. ISBN 978-92-64-99113-2, NEA/CSNI/R(2010)1, Nuclear Fuel Behavior Under Reactivity-initiated Accident (RIA) Conditions, 2010.

48. Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria, TR-1002865, EPRI, June 2002, (Withdrawn 2007), (NRC ADAMS Accession ML021720080).

49. NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance", January 19, 2007 (NRC ADAMS Accession Number ML070220400).

50. ANP-10286P-A Revision 0, *U.S. EPR Rod Ejection Accident Methodology,* AREVA NP, November 2011.

51. WCAP-16182-NP-A, Revision 1, Westinghouse BWR Control Rod CR 99 Licensing Report – Update to Mechanical Design Limits, October 2009 (NRC ADAMS Accession Number ML093240366).

52. Final Safety Evaluation of Topical Report NEDE-33284P, Supplement 1, Revision 0, "MARATHON_ULTRA CONTROL ROD ASSEMBLY" (NRC ADAMS Accession Number ML120380180).

53. NUREG/CR-0056 "Critical Heat Flux Under Transient Conditions: A Literature Survey," J. C. M. Leung (ANL-78-39), June 1978.

54. Collier,J. G., Thome, J. R, "Convective Boiling and Condensation," 3rd edtion, Oxford University Press, New York, 1996 (ISBN 978-0198562962).

55. NUREG/CR-6150, Vol. 4, Rev.2 SCDAP/RELAP5/MOD 3.3 Code Manual, MATPRO-A Library of Materials Properties for Light-Water-Reactor Accident Analysis, US-NRC, January 2001. (NRC ADAMS Accession Number ML010330363,ML010330400)

56. Mosteller, R. D.; Eisenhart, L. D.; Little, R. C.; Eich, W. J.; Chao, J; "Benchmark Calculations for the Doppler Coefficient of Reactivity", Nuclear Science and Engineering, v 107, n 3, p 265-271, Mar 1991.

57. FS1-0010616, *FSQA-07-RODEX4-UAPR13-0_UsrMan-000,* AREVA-NP, May 2013.

58. BAW-10247PA Revision 0, *Supplement 1Q2P Revision 0,* "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding, Responses to NRC Request for Additional Information, AREVA NP, November 2013.

59. BAW-10247PAQ4P Revision 0, *Second Round of RAIs on RODEX4 Documents* contained in BAW-10247PA, AREVA NP, February 2008.

60. Boiling Water Reactor GE BWR/4 Technology Advanced Manual, Chapter 6, (NRC ADAMS Accession Number ML023010606)

61. Quad Cities UFSAR Revision 11.

62. Susquehanna Steam Electric Station Final Safety Analysis Report, Revision 65.

63. E. Hellstrand, et al., "The Temperature Coefficient of the Resonance Integral for Uranium Metal and Oxide," Nuclear Science and Engineering, Vol. 8, 497-506 (1960).

64. Fuel Management Manual, Revision 4, "AREVA Front End Business Group Fuel Business Unit Management Manual," AREVA NP Inc., July 2013.

65. 51-9216365-000, "Guideline for Analyses of the BWR CRDA with the AURORA-B Methodology," AREVA NP, December 2013.

66. IDO – 17206, "Quarterly Technical Report SPERT Project – January, February, March, 1966," September 1966, Phillips Petroleum (Available from US Department of Commerce, National Technical Information Service).

67. NUREG-0933 Resolution of Generic Safety Issues: Issue 53: Consequences of a Postulated Flow Blockage Incident in a BWR (Rev. 1)

68. FS1-0009130 Revision 2, *FSQA-07 S-RELAP5 Input Data Requirements (User's Manual),* AREVA NP, July 2013.

69. FS1-0008072 Revision 1, *FSQA-07 MB2-K User's Manual,* AREVA NP, May 2013.

Appendix A ANP-10333Q1NP-000

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

AREVA

AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)

ANP-10333Q1NP
Revision 0

Responses to NRC Request for Additional Information

April 2017

AREVA Inc.

Nature of Changes

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

Contents

List of Tables

List of Figures

INTRODUCTION

The AURORA-B CRDA LTR (Reference 1) was written to address interim criteria identified in SRP Section 4.2 Appendix B. Subsequent to the submittal of the CRDA LTR, Draft Regulatory Guide 1327 (Reference 3) has been proposed which has modified the criteria. Therefore, in addition to addressing the specific question posed, additional information has been included within the response indicating how the methodology could address criteria similar to that presented in DG-1327.

The original topical report was developed using the UDEC12 version of S-RELAP5. The information provided in this document has been developed using a new version of AURORA-B which contains updates to both S-RELAP5 and the MB2-K modules. Appendix A summarizes the justification for the code updates and describes the software change control process utilized by AREVA. As a result of the necessary code changes and other documentation issues found in the original topical report, AREVA will update select sections of ANP-10333P to ensure the final methodology is sufficiently captured by the approved version of the LTR. The updated topical report will be provided to the NRC in the released document ANP-10333PA, Revision 1.

Potential Deviations

Two of the most important parameters used to determine if the interim acceptance criteria are met (fuel rod enthalpy and fuel temperature) are directly dependent on the magnitude of the power burst resulting from the CRDA. As a result, any potential deviation between the physical reality and the CRDA evaluation that may lead to a reduction in the limiting reactivity response in the CRDA analysis needs to be addressed. The NRC staff has identified some such potential deviations and determined that the following information is required in order to complete the evaluation.

RAI-1:

The primary mitigation mechanism for the CRDA event is the Doppler reactivity effect as the fuel temperature increases. The TR explains that this effect **[**

], *which were set to maintain consistency with the steady state core simulator code (MICROBURN-B2). The CRDA event is a very fast transient that primarily consists of a fuel power/temperature response and its mitigation by negative reactivity due to Doppler feedback. As such, it would be expected to be more sensitive to the accuracy of the MB2-K treatment of the variation in the strength of the Doppler effect in the fuel rod due to variation in the fuel temperature. The TR provides results from a sensitivity study in section 8.7.3.11, but it is not clear how the study supports the selected weighting factors or the uncertainty associated with the pellet radial power distribution. Please provide a discussion of the technical basis supporting the weighting factors and the uncertainty selected for use with the CRDA analysis methodology, including the following:*

a. *changes in the radial fuel temperature distribution during the power excursion associated with the CRDA, and*
b. *the impact of fuel geometry changes due to irradiation (in particular, pellet growth and cracking) on the radial fuel temperature distribution.*

AREVA Response RAI-1.:

The weighting factors were set consistent to those of the AURORA-B TR (ANP-10300) for consistency. The Doppler temperature is adequately reflected with the volume average [

] to the calculated enthalpy rise. [

] The Doppler effective temperature coefficient results from ANP-10333P (Table 8.19) have been re-calculated with the latest code version and updated in Table B.11 of Appendix B. The data from Table B.11 has been recast in Table 1.1 to better show the impact of the surface weighting on the peak power, the prompt enthalpy, and total enthalpy.

Table 1.1 Summary Impact of Surface Temperature weighting

[

]

To better understand the basis for this behavior the following discussions are provided

- Changes in Radial Fuel Temperature Distribution
- Irradiation and Fuel Geometrical Changes
- Justification of Temperature Weighting and Uncertainty

Changes in Radial Fuel Temperature Distribution

Changes in the radial fuel temperature distribution during the power excursion are illustrated in Figures 1.1 and 1.2 for rods with exposures of 22 MWd/MTU and 44 MWd/MTU respectively. During the initial power excursion the fuel pellet temperature increases rapidly at all radial nodes. (The center of the pellet is identified as node 1 and the surface is [

] The temperature in the outer regions of the fuel pellet begins to decrease shortly after the core power peaks as heat is transferred out of the pellet and to the coolant. However the inner portion of the fuel continues to rise in temperature. The pellet radial power profile varies with burnup due to changes in the radial isotopic distribution of fissile material with irradiation (because of Pu buildup at the pellet outer rim). Fresh UO2 fuel pellets have a uniform radial isotopic distribution. Therefore, at beginning of life the power profile in the pellet in the radial direction only reflects the neutron thermal flux depression inside the pellet. As the fuel is irradiated there is a buildup of fissile plutonium isotopes from the resonance capture of neutrons in the U238 and subsequent decay to fissile isotopes of plutonium. The majority of the neutron capture takes place at or near the pellet surface (self-shielding effect) resulting in burnup dependent power distribution as indicated in Figure 1.3. This radial power distribution is the combined effect of neutron thermal flux and fissile isotope distributions within the fuel pellet. These processes are accounted for in RODEX4 methodology (Reference 7).

The nodal ring [

]

[ ]

Figure 1.1 Mid Range Burnup Temperature Rise

[ ]

Figure 1.2 Higher Range Burnup Temperature Rise

Figure 1.3 Pellet Radial Power Distribution at Various Burnups

Irradiation and Fuel Geometrical Changes

During normal operation, the fuel pellet develops a number of cracks in the radial direction because of the parabolic temperature distribution, which creates tensile hoop stresses in the outer pellet region. These radial cracks are [

] The enhanced porosity occurs at the pellet outer rim and decreases the fuel thermal conductivity and is accounted for in RODEX4. Therefore, the impact of fuel geometry changes due to irradiation (such as pellet growth and cracking) are accounted for in the RODEX4 methodology and the mechanical model was accepted by the NRC (Reference 7, SER

Section 3.6). As such the fuel properties are defined with an approved methodology prior to the power excursion.

Justification of Temperature Weighting and Uncertainty

The increase in the prompt enthalpy with an increase in surface weighting is an [

] The prompt enthalpy is defined at the time of FWHM (full width half maximum) past the time of the peak power. An increase of the temperature surface weighting during the power ascension results in a higher effective temperature and more negative Doppler feedback. This reduces the peak power and [

] results in slight increase in the prompt enthalpy.

Following the rapid temperature rise, the surface temperature of the pellet begins to decrease as heat is conducted to the coolant. In this phase, the use of surface temperature weighting will have a [

] a conservative bias to the total enthalpy in that a cooler surface temperature is applied.

The use of the [ ] surface weighting provides a conservative bias for both prompt and total enthalpy rise. No credit is assumed for this conservative bias. The uncertainty reported was [

] profile in the fuel.

RAI-2:

Page 8-24 of the TR discusses the technical basis for control rod modeling and its uncertainty. This discussion does not appear to address the fact that some control rods are designed with axial variations in neutron-absorbing material. Such variations may affect the reactivity insertion curve during a CRDA. Please provide a discussion of how axial variations in neutron-absorbing material used in the control rod blade will affect the reactivity insertion curve, including any limitations to the applicability of the recommended modeling approach and uncertainties.

AREVA Response RAI-2:

Modern control blade designs may have varying axial cross section compositions (Paragraph 1 Section 2.2 Reference 4). The actual drop is performed with a single axial zoned control blade. More recent control blades with axially vary designs tend to have a slightly lower worth in the top node when hafnium is used. Also, newer blades may have a lower worth near the bottom with the removal of absorber material. The typical design criteria for new control blades demonstrate that the blade designs "meets all scram insertion criteria, reactivity control criteria, and CRDA" (Reference 5 Section 8). Therefore utilization of a uniform axial blade composition with a worth [

] is consistent with blade design criteria. Even though some modern blades have a lower worth in the blade tip, the use of a higher worth blade tip increases the impact of the CRDA. In other words, more positive reactivity would be inserted when the dropped rod is modeled as a uniform axial blade as opposed to a blade which may contain some lower worth axial zones.

The uncertainty of the blade worth is accounted for within the methodology by utilizing a [

] modeling uncertainty are included in the evaluation of the

methodology uncertainty shown in Table 8.20 of the CRDA LTR report. The methodology uncertainty is discussed further in the response to RAI-10.

The application of this methodology with respect to control blade designs is reflected in the methodology uncertainty. The control blade uncertainty used in the methodology has been selected consistent with current control rod designs, i.e. those designed to match worth with existing blade types. If new blade types are introduced in which the change in rod worth exceeds the blade uncertainty for the methodology, then adjustments are required by either choosing a blade type with a higher worth or by adjusting the blade uncertainty to assure that the range of blade worth's in the core remain bounded by the analysis.

RAI-3:

Page 8-20 of the TR discusses the power history studies to justify the use of nodal average powers in constructing the rod power history effects. The approach used appears to result in lower exposures for lower powered histories and higher exposures for higher powered histories. It is not clear if the observed changes in the enthalpy rise are due to the change in average power or the change in exposure. The latter effect is not relevant because the CRDA analysis is performed at the cycle exposures which are considered to be most likely to be limiting. If the intent of this study is to demonstrate that use of nodal average powers in constructing the rod power histories is acceptable, [

]

used to arrive at the same fuel exposure.

AREVA Response RAI-3:

Additional sensitivity cases were run to illustrate the impact of a change in power level only on the rod power histories. The depletion power was decreased and increased by both [

] in which the last three digits represent the percent of nominal depletion power. The numerical results of the impact on both prompt and total enthalpy rise are given in Table 3.1 and the enthalpy rise with time is provided in Figure 3.1. The prompt enthalpy rise and core power rise have the same trend which increases with a lower depletion power and decreases with higher depletion power.

The change in depletion power primarily affects the amount of fission gas released from the pellet. An increase in depletion power will enhance the migration of fission gas from the pellet to the free gas volume of the rod. This increase in fission gas in the gap would decrease the gap thermal conductivity which would then hold the energy in the fuel longer and delay or decrease the moderator temperature feedback. This can result

in an increase in the total enthalpy rise. The impact of fission gas will only affect the [ ] occurs is essentially adiabatic.

There is no significant increase in gap conductivity with a decrease in the depletion power below nominal. This is to be expected since the release of fission gas during depletion at nominal conditions is already small, and a further reduction of this release due to reduced depletion power would have little effect. Therefore the total enthalpy rise increases for cases using a lower power level for the rod depletion which is consistent with the prompt enthalpy rise.

[ ]

Table 3.1 Impact on Enthalpy with Variation of Depletion Power

[ ]

Figure 3.1 Peak Node Enthalpy Rise for Power Depletion Sensitivity

The steady state power distribution uncertainty and rod peaking factor were originally incorporated directly into the enthalpy uncertainty. However, the steady state power uncertainty is more appropriately reflected [

]

However, the steady state power uncertainty and the rod peaking factor will be conservatively retained in Table 8.20. (See RAI-10 for additional discussion of the uncertainty.)

Supplement to RAI-3 with respect to DG-1327

Subsequent to the submittal of the CRDA LTR, draft guidance (Reference 3) was published which changed the high temperature failure criteria. The proposed high temperature failure is decreased below that in SRP 4.2 and varies with the differential cladding pressure. Additional sensitivity cases were run to assess the adequacy of using the average rod power histories for the high temperature evaluation.

These additional sensitivity cases better illustrate the impact of the combined power level and burnup on the rod power histories. In the combined sensitivity evaluation, the rod power level was increased and decreased by both [

] also increased or decreased. These results are provided in Table 3.2.

Table 3.2 Impact on Enthalpy with Variation of Depletion Power and Burnup

[

]

Both the combined results shown in Table 3.2 and the depletion power alone shown in Table 3.1 indicate that a large variation in the burnup history and depletion power has a very small impact on the enthalpy rise. However, the depletion power and burnup have a large impact on the internal rod pressure as a result of the amount of fission gas generated and the amount released from the pellet.

For evaluating the high temperature failure threshold, the rod internal pressure [ ] if the total enthalpy is greater than the minimum failure threshold. For illustration, the results from a cold case are used to demonstrate the application [ ] increase in rod depletion power and corresponding burnup. The enthalpy values determined from the nominal depletion results are tabulated against the differential pressure determined for both the nominal depletion and the high power/high burnup depletion histories. The transient fission gas release for rod pressurization is determined in accordance with Figure 6-1 of Reference 3, and [ ] These results are shown in Figure 3.2 along with the proposed high temperature failure threshold from DG-1327. Fuel rods with a total enthalpy less than the minimum high temperature failure threshold (100 Cal/g in Figure 3.2) would not require application of the increased pressure. (The results in Figure 3.2 also indicate the small impact on fuel enthalpy with the variation in the fuel rod history.)

[ ]

Figure 3.2 Example High Temperature Evaluation with Modified Power History

RAI-4:

The sensitivity study documented in Section 8.7.2.3 for the core initial coolant temperature consists of a series of perturbations on the core initial coolant temperature. An increase in the core initial coolant temperature would result in a reduction in reactivity due to the corresponding increase in fuel temperature and the Doppler effect. If this was not compensated for in some other way (e.g., rod pattern adjustment), then the sensitivity studies may incorporate a less critical core as the starting point, which could reduce the severity of the prompt power pulse. Please provide a discussion of the effect of changes to the core initial coolant temperature on the initial reactivity of the core, and how they are captured by the sensitivity study.

AREVA Response RAI-4:

The sensitivity study documented in Section 8.7.2.3 is not a true sensitivity with respect to coolant temperature alone. It is actually an evaluation of the temperature dependent reactivity compensation. This temperature dependent reactivity compensation decreases the event severity as the initial temperature is increased. Although using the lower temperature simulates a more reactive core, the AURORA-B system normalizes the eigenvalue to be critical. The fuel temperature and moderator conditions decrease the system reactivity as the temperature is increased. Therefore the use of the lower initial temperature as the initial condition for the drop will produce bounding results relative to the same rod drop configuration at a higher initial temperature. Although this lower temperature is used for the evaluation of the drop, [

] The sensitivity results have been completed with the new code version and the results are included in Appendix B. The original conclusion remains supported.

Scenarios Selection

Some of the details provided in the TR for the approach used to select CRDA scenarios for analysis do not include a justification that the approach is appropriate for its intended purpose. If the CRDA analysis is performed for scenarios that do not bound the worst case scenario, then a non-conservative result will be used to demonstrate that the acceptance criteria for the CRDA event are met. Therefore, the NRC has determined that additional information is needed to clarify how the selection process outlined in the TR to select the appropriate scenarios for analysis will bound all possible scenarios.

RAI-5:

Page 7-8 states that the rod drop with the highest static rod worths at three exposures for the cycle are used in the CRDA evaluation, along with other candidate rods identified to evaluate the impact of the CRDA on fuel rods with high exposure and cladding content. The PCMI failure threshold is dependent on the hydrogen content in the cladding, so fuel with higher exposure may fail the acceptance criteria even if the prompt enthalpy rise is smaller than lower exposure fuel. In order to address this possibility, selection criteria are provided to guide the selection of additional rods as necessary. It is not clear how the proposed selection criteria will ensure that any potentially limiting rods will be identified for a broad range of possible cladding hydrogen contents, fuel types, and plant configurations. Please describe how the selection criteria will be effective in identifying suitable candidate rods for analysis that will ensure that the acceptance criteria are met, especially for fuel with cladding hydrogen content in the range where the failure threshold rapidly decreases (75 to 150 ppm).

AREVA Response RAI-5:

The [

] Table 5.1 shows actual end of cycle core average delayed neutron fractions for several recent AREVA fuel reloads. These results include both 18 and 24 month cycle lengths and a range of core sizes from 408 to 800 assemblies.

Table 5.1 End of Cycle Delayed Neutron Fraction

[ ]

The criteria and evaluation process are revised to account for broad range of potential failure thresholds and core loading strategies. This revised selection criteria is now independent of the actual form of the failure threshold criteria and core loading strategy.

The selection will be based on [ ] provides a method to assure that an intermediate failure threshold would not be exceeded. EOC evaluations will utilize the minimum failure threshold at that condition [ ]

During the initial application for a given plant group or fuel loading type, rod drops for a range of rod worths greater than [

] The results of this initial evaluation are

[

] The establishment of the evaluation boundary is illustrated in Figure 5.1. Candidate control rod drops and the minimum failure threshold are tabulated and compared with the evaluation boundary. To apply the evaluation boundary, the y-axis is changed from [ ] as shown in Figure 5.2. Candidate rods for which the minimum failure threshold [

] since the drop will not challenge the failure threshold. However, a drop with a minimum failure threshold [

] requires additional evaluation.

Figure 5.1 Establishing Evaluation Boundary

Figure 5.2 Example Evaluation Boundary

Following is an example application to demonstrate this process.

As discussed previously, the evaluation [ ] application. The rod drops with worth values greater than [ ] at BOC, PHE (peak hot excess) reactivity, and EOC. The first tabulation at BOC is given in Table 5.2. In this example the failure thresholds around the rods are [ ] only the highest worth drop is evaluated. Since there are no failures the dose limits are not exceeded for the BOC rod drops using failure criteria determined for the [ ]

Table 5.2 BOC Candidate Rods

[ ]

The tabulation at PHE is given in Table 5.3. Only a few drops were found with worth values greater than [ ] and no candidate rods were to the [ ] Therefore, only the highest worth rod is evaluated. Since the maximum enthalpy rise is less than the [ ] no additional evaluation of the PHE rods is required [ ]

Table 5.3 PHE Candidate Rods

[ ]

The tabulation at EOC candidate rods is given in Table 5.4. Several rod worth values are greater than [ ] This is primarily due to the gadolinium burnout in this cycle. The minimum failure threshold is similar for several rods. Evaluation of the highest worth rod is completed for [ ] Rods 12, 87, 174, 14, and 102 are one row in from the edge. Rods 83, 155, and 37 have similar fuel loading and failure thresholds. Although the maximum fuel enthalpy exceeds the [ ] the final evaluation is made by comparing the individual enthalpy rise for each fuel rod to its specific failure threshold. If fuel rod failures are determined the dose consequences are then evaluated.

Table 5.4 EOC Candidate Rods

[

]

In the above example, the fuel enthalpy rise for BOC and PHE were less than the failure criteria [ ] However, if the assembly fuel enthalpy rise results in fuel failures based on the [ ] but are exceeded at the [ ]

[

]

It is anticipated that with the application of the methodology additional data results can be tabulated and that a generic evaluation boundary curve may be established.

The actual selection process is presented in the following steps:

[

]

(See the response for RAI-9 for discussion of step 5.)

RAI-6:

Section 7.6 discusses the approach used to determine rod enthalpy increases for individual fuel rods, which is then used to determine how many rods will experience PCMI failure for fission gas inventory release purposes. The text is not clear regarding how the [

] *Please describe why this assumption* [ ] *would be expected to yield bounding results of fission gas inventory releases for all possible fuel lattices, including those with strong poisons that have not yet fully burned out or those that have experienced strongly asymmetric operating conditions (e.g., adjacent control rod insertion).*

AREVA Response RAI-6:

This approach has been revised to address the entire history of the fuel assembly. The [

] individual pin enthalpies using the peak and average enthalpy rise at each axial level of each assembly. The average is increased [

] discussed in Section 8.7.3.12 of the CRDA LTR. The following equation will replace Equation 3 within Section 7.6 of the CRDA LTR.

[

]

[ ]

A fuel pin is determined to fail if it exceeds its failure threshold at any axial level.

$$\mathrm{PINF}_i = \begin{cases} 1 & \text{If } \left(\Delta h_{i,k} > \Delta hf_{i,k}\right) \text{for all k} \\ 0 & \end{cases}$$

Where PINF_i is a fuel rod failure flag

$\Delta h_{i,k}$ is the enthalpy rise in fuel rod i at axial level k (with uncertainty multiplier)

$\Delta hf_{i,k}$ is the enthalpy failure threshold for fuel rod i at axial level k

The total number of fuel rods failed (NFAIL) in an assembly is then the summation of the individual fuel rod failure flags.

$$\mathrm{NFAIL} = \sum_{L=1}^{npin} \mathrm{PIN}_i$$

The enthalpy that is used for the determination of the transient fission gas release (TFGR) for evaluation of dose consequence is determined for each axial node as the [ ]

Sections 7.6 and 9.5 of the CRDA LTR will be revised to reflect this change. The example provided in Section 9.5 was reevaluated using [ ] The tabulation of fuel rod failures for two assemblies are provided in Table 6.1 and 6.2. Tables 6.3 through 6.6 contain the average release

fractions for two assemblies using SRP-4.2 release fractions and PNNL-18212 R1 release fractions.

Table 6.7 and 6.8 contain the [

] as shown in Table 6.9.

Changes in the final fission gas release formulation for evaluation of dose consequences, such as those proposed in DG-1327, would be handled in a manner consistent with the process demonstrated in the example problem.

Table 6.1 Determination of Rod Failures for Assembly V17118

Table 6.2 Determination of Rod Failures for Assembly V17134

Table 6.3 V17118 Fission Gas Release Fraction Determination Reg. Guide 1.183

[ ]

Table 6.4 V17134 Fission Gas Release Fraction Determination Reg. Guide 1.183

Table 6.5 V17118 Fission Gas Release Fraction Determination PNNL-18212 Rev 1

[

]

Table 6.6 V17134 Fission Gas Release Fraction Determination PNNL-18212 Rev 1

[ ]

Table 6.7 Fission Gas Release Fractions For Assembly V17118

[ ]

Table 6.8 Fission Gas Release Fractions For Assembly V17134

[ ]

Table 6.9 Equivalent Rod Failures

[ ]

RAI-7:

Section 7.7 describes the evaluation of the CRDA event for at-power conditions using the critical power ratio (CPR). The proposed approach seems sufficient for typical analyses, but it is not clear how broad the generic applicability of the analysis is. Based on the discussion in the TR, it appears that the [

]. *If that is true, please describe how unusual operating conditions may affect this determination, such as insertion of suppression rods which result in a radial asymmetry in the core power distribution.*

AREVA Response RAI-7:

The evaluation of MCPR response in Section 7.7 of the LTR is [

] The negative reactivity from the generation of significant or increased voiding dampens the impact of the BWR CRDA when the event occurs "at-power" conditions as opposed to the unvoided startup conditions. The power pulse in the power range (Figure 7.10 of the CRDA LTR) shows a very broad pulse with limited power increase. The results provided on Page 7-26 of the CRDA LTR as well as the results provided later in this response demonstrate that the CPR [

] These results and conclusions are consistent with the reduced impact for power range rod drop as discussed in Reference 9 page 3-2. The characteristic behavior of the BWR with respect to void reactivity is independent of plant type.

Impact of Asymmetric Operation

An evaluation has been performed to address the question of the impact of asymmetric operation, such as may be encountered with the insertion of suppression rods. In the process of evaluating the impact of the asymmetric power distribution it was discovered

that some of the initial cases reported in Table 7.7 and Figure 7.11 of the CRDA LTR were incorrect due to an input error. This error has been documented in AREVAs corrective action program and the corrected results are provided later in this response. The asymmetric operation evaluation results are compared to these corrected results.

The possibility of unusual operating conditions such as stuck control rods or insertion of suppression rods could result in a radial power asymmetry in the core. It is anticipated that in such circumstances that the plant would be operated in conditions as close to the nominal cycle licensing basis as possible. To address asymmetric operation, an additional evaluation was performed by inserting suppression rods and then simulating a rod drop. The initial pattern and the modified rod pattern with asymmetrical operation are given in the Figures 7.1 and 7.2, respectively. It is noted that to maintain the eigenvalue additional modifications were made to the control rod pattern.

[ ]

Figure 7.1 Starting Rod Pattern for PHE drop at 3875MW Rod 30-19

[

]

Figure 7.2 Adjusted Asymmetric Rod Pattern for PHE drop at 3875MW Rod 30-19

Rod 30-19 was dropped and the resulting MCPR was evaluated. A comparison of the MCPR between the two rod drops is provided in Figure 7.3. [

] The change in void reactivity feedback for the BWR is small since the core average void would be similar between symmetric and asymmetric power distributions. This void reactivity feedback limits the severity of the rod drop when operating in the power range. Therefore there is only a small impact on the CPR change due to the suppression rods.

[

]

Figure 7.3 Comparison of CPR Response with Asymmetric Initial Configuration

Supplemental Information Regarding Identified Calculation Error

[

] As such the conclusion drawn from the original CRDA LTR results is not challenged by the corrected results and remains valid.

In the corrected evaluation, the rod selected for BOC was changed to a more conservative rod drop which resulted in a decrease in the BOC CPR at 88% power. The selection of the more conservative rod for BOC conditions and [

] in the CRDA LTR.

Table 7-7 and Figure 7.11 of the CRDA LTR will be replaced with the corrected results provided in Table 7.1 and Figure 7.4 of this response.

Table 7.1 Power Range Minimum CPR

[ ]

[ ]

Figure 7.4 Power Range CPR Response CRDA

RAI-8:

The TR does not appear to discuss the applicability of this methodology to mixed cores. Please describe any limitations or changes necessary to account for cores with non-AREVA fuel, including evaluation parameters not directly involved in the CRDA calculation such as the cladding hydrogen content.

AREVA Response RAI-8:

The neutronic and transient modeling for the CRDA will be completed with the same methodology regardless of fuel type. The modeling will explicitly consider geometric differences in fuel designs present in mixed core applications. The fuel mechanical properties of UO2 pellets and zircaloy cladding are modeled with the RODEX-4 code for each fuel type according to the respective fuel mechanical properties. The sensitivity studies show that there is little impact on the actual event based on the variation of the rod thermal mechanical properties. Therefore, the establishment of the rod thermal mechanical properties with the RODEX4 code is appropriate. The database used to develop the AREVA hydrogen uptake model includes recrystallized (RXA) cladding typically used in non-AREVA fuel. It is anticipated that the AREVA hydrogen uptake model will be used for all fuel types. However, hydrogen uptake models designated as acceptable by the NRC such as those in Reference 6 may be used for other vendor cladding.

With respect to the evaluation of MCPR for mixed core configurations, AREVA utilizes an NRC-approved process (Reference 8) for developing additive constants (correlation coefficients) to apply the AREVA CPR correlations to the non-AREVA fuel types.

Sensitivity studies were performed on various input parameters for the CRDA calculation. In some cases, these studies were used to support use of a bounding value for the CRDA analysis. In other cases, the study results were used to support a value for the uncertainty in the enthalpy rise. These studies and their results are used to provide reasonable assurance that the results of the CRDA analysis will bound real-world conditions. The NRC identified some cases where the sensitivity study approach or the conclusions did not clearly support the intended purpose, so further information is necessary.

RAI-9:

A number of the conclusions derived from the sensitivity studies do not appear to be supported by the actual results from the calculations performed for the studies. For example, section 8.7.2.5 discusses the sensitivity of the CRDA analysis results to the initial core flow. The text states that the prompt enthalpy rise decreases as the initial core flow increases, while the total enthalpy increases as the initial core flow increases. This is used to support the use of a minimum core flow as a bounding value for determining the prompt enthalpy rise. However, no clear recommendation is given for evaluation of the total enthalpy, and the sensitivity studies show that the limiting value for the prompt enthalpy rise was calculated for an initial core flow just above the recommended minimum value. Please provide further clarification for the behavior of the prompt enthalpy rise as a result of variations in the initial core flow, and provide guidance on the appropriate initial core flow to use when evaluating the total enthalpy.

AREVA Response RAI-9:

The sensitivity studies in Section 8.7.2 and 8.7.3 of the CRDA LTR have been evaluated with the enhanced convergence methodology and are included in Appendix B. The revised sensitivity results for the initial core flow (Table B.3) indicate between [

] Although the total enthalpy increases with the initial core flow, to achieve a higher core flow would require the addition of heat from recirculation pumps, decay heat, or actual power generation. The results of the sensitivity with respect to the initial temperature show that an increase in initial coolant temperature significantly

reduces the enthalpy rise. Therefore use of the lower flow temperature conditions should bound the results for the high temperature evaluation based on total enthalpy. To ensure this, an additional step has been added to the evaluation process (see RAI-6 response) to confirm the results from the low flow evaluation bound that of the high flow condition.

The additional step assures that the total enthalpy is evaluated at the initial flow conditions which results in the highest total enthalpy.

RAI-10:

Section 8.7.4 states [

] *Please provide a justification for the appropriateness of the statistical approach used.*

AREVA Response RAI-10:

[

]

Table 10.1 [

]

[ ]

[

]

Table 10.2 Uncertainty Parameters for CRDA Sample Problem Non-parametric Uncertainty Analyses

[ ]

With the exception of parameter rod worth, the sampling ranges used for all of the parameters in the above table are the same as those used for the AOO methodology, as shown Section RAI-49b of the RAI responses for the AOO methodology (ANP-10300Q2P).

Parameter rworth is a new sampling parameter. Since rod worth cannot be sampled directly through [ ]

Lattice calculations were performed to generate modified cross sections which

represented an [

]

An example of ordered prompt and total ΔH results for test case ROD121 is seen in Table 10.3. A sample ensemble [

]

[

]

Table 10.3 Example of Ordered ΔH Results from Non-Parametric Statistical Process

[

]

[

]

[

]

Two of the parameters originally included in Table 8.20 of the CRDA LTR were not included in this evaluation; the power distribution and rod peaking factors. [

] Therefore they are omitted from this evaluation.

Table 10.4 CRDA Test Case Results from Non-Parametric Statistical Process – ROD 118 Drop Location

Table 10.5 CRDA Test Case Results from Non-Parametric Statistical Process – ROD 121 Drop Location

Table 10.6 Maximum CRDA Enthalpy Rise Uncertainties

[ ]

RAI-11:

The TR recommends a different time step size for some CRDA evaluations due to the nature of the transient. Sensitivity studies were performed to determine the impact of any further changes in time step size. However, it is not clear how the specific value recommended in the TR was determined. The documentation of the sensitivity studies only discusses calculations performed at a different time step size. Please clarify if the discussion on the time step sensitivity in Section 8.7.1 was intended to characterize the sensitivity study as showing that a further change in time step size beyond the recommendation in Table 7.5 would not yield a significant change in calculated enthalpy. If the NRC staff interpretation is in error, please provide sufficient information to enable an understanding of how the sensitivity study relates to the final recommendation on time step sizes.

AREVA Response RAI-11:

Sensitivity studies were performed with respect to the maximum time step size. The actual numerical results of the study, provided in Table 11.1 do not lead to a straight forward conclusion. For this case, some of the larger maximum time steps actually lead to higher deposited enthalpies and the prompt and total enthalpy rise did not always trend in the same direction.

The time trace plots (Figure 11.1) show that the results converge as the time step size is decreased. Based on the comparison traces the time steps sizes of Table 7.5 of the CRDA LTR are specified.

Table 11.1 Response to Change in Maximum Time Step Size

[ ]

[ ]

Figure 11.1 Trace of Enthalpy Rise with Different Maximum Time Steps

REFERENCES

1 ANP-10333P Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," AREVA, Inc., March 2014.

2 ANP-10300P Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," AREVA NP, Inc., December 2009.

3 US NRC Draft Regulatory Guide DG-1327, *Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents*, November 2016. (NRC ADAMS ML16124A200)

4. WCAP-16182-NP Revision 3, *Westinghouse BWR Control Rod CR 99 Licensing Report – Update to Mechanical Design Limits*, August 2016 (NRC ADAMS ML16235A108)

5. NEDO-33284-A Revision 2, *Marathon-5S Control Rod Assembly*, October 2009 (NRC ADAMS ML092950284)

6. Memorandum from Paul M. Clifford (NRC) to Timothy J. McGinty (NRC) "Acceptable Fuel Cladding Hydrogen Uptake Models," May 13, 2015. (NRC ADAMS ML15133A306)

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

8. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

9. NEDO-10527, *ROD DROP ACCIDENT ANALYSIS FOR LARGE BOILING WATER REACTORS.* Class I. General Electric: San Jose, CA. (NRC ADAMS ML010870249)

APPENDIX A

CODE ISSUES AND METHODOLOGY REVISIONS

[

]

With respect to the change control process resulting in the certification of UMAR16, AREVA's Software Quality Assurance Procedures for NRC approved methods require both the verification and validation (V&V) of code modifications and the assessment of the changes on the approved methodology results. These are two separate steps in the code release process.

1. The V&V of a code modification ensures the modification functions as intended. It includes new validation against data or higher order methods when the particular update modifies physical models and the existing test suite is inadequate to assess the change in performance.

2. In addition to the modification specific V&V described in item 1, AREVA evaluates the impact of the code modifications on the EM performance

through a series of tests referred to as the Continuity of Assessment (CoA) process. The CoA process recalculates a sufficient cross section of the analyses included in the methodology LTR to assess the impact of modifications relative to both the previous code version and to the final results (i.e. results updated as a result of RAIs) presented in the approved LTR.

[

]

Table A.1 Identification of Modifications to ANP-10333P

[ ]

APPENDIX B

The sensitivity studies in Section 8.7.2 and 8.7.3 of the CRDA LTR have been evaluated with the enhanced convergence methodology which has been verified for the AURORA-B CRDA methodology. This enhanced convergence within the MB2-K module of S-RELAP5 has been verified for the AURORA-B AOO methodology. The sensitivity studies were repeated for Rod 118 and the results are provided in this appendix. As discussed in the RAI-4 and RAI-9 responses, the original conclusions drawn with the Rod 12 studies in the LTR remain appropriate.

Initial coolant temperature 8.7.2.3:

The actual magnitude varies but the results show the similar trend as provided in Table 8.4 of the CRDA LTR.

Table B.1 Core Initial Temperature Sensitivity Rod 118 EOFP

[ ]

Initial Core Power Level Section 8.7.2.4:

The actual magnitude varies but the results show the similar trend as provided in Table 8.5 of the CRDA LTR. There are minor variations in the tail of the power pulse which results in slightly different values for total enthalpy when starting from a power level an order of magnitude greater.

Table B.2 Core Initial Power Sensitivity Rod 118 EOFP

[ ]

Initial Core Flow Sensitivity Section 8.7.2.5:

In general the trend is similar to of Table 8.6 of the CRDA LTR, however the variation in the prompt enthalpy rise is small across the range of initial core flows with the latest code version. The total enthalpy trend is consistent and increases with core flow. Examination of the tail of the power pulse shows that it [ ] of rated flow for cold startup conditions is appropriate.

Table B.3 Core Initial Flow Sensitivity Rod 118 EOFP

[ ]

Fuel Rod Power History Section 8.7.2.6:

The fuel rod power history is addressed in RAI-3.

Moderator Feedback 8.7.3.2:

The actual magnitude of the impact varies for the moderator feedback however the results show the similar trend as provided in Table 8.12 and Table 8.13 of the CRDA LTR.

Table B.4 Active Moderator Density Feedback

[ ]

Table B.5 Bypass Channel Moderator Density Feedback

[ ]

Fuel Temperature Feedback 8.7.3.2:

The results with the enhanced convergence are consistent with those given in Table 8.14 of the CRDA LTR.

Table B.6 Fuel Temperature Feedback

[ ]

Effective Delayed-Neutron Fraction 8.7.3.2:

The results with the enhanced convergence are consistent with those given in Table 8.15 of the CRDA LTR. The conclusions in the CRDA LTR remain unchanged.

Table B.7 Delayed Neutron Fraction Sensitivity

[ ]

Heat Resistances in High Burnup Fuel, Gap, and Cladding 8.7.3.7:

The results with the enhanced convergence are consistent with those given in Tables 8.16 and 8.17 of the CRDA LTR. The conclusions in the CRDA LTR remain unchanged.

Table B.8 Gap Width Adjustments

[ ]

Table B.9 Fuel Heat Transfer Coefficient Sensitivity

[ ]

Direct Energy Deposition to Moderator 8.7.3.10:

The results with the enhanced convergence are consistent with those given in Tables 8.18 of the CRDA LTR. The conclusions in the CRDA LTR remain unchanged.

Table B.10 Heat Deposition Sensitivity

[ ]

Pellet Radial Power Distribution 8.7.3.11:

The results with the enhanced convergence are consistent for the prompt enthalpy rise with those given in Table 8.19 of the CRDA LTR. However, the total enthalpy rise values are larger. The selection of the Doppler weighting is discussed in detail in the response to RAI-1.

Table B.11 Doppler Effective Temperature Coefficient Sensitivity

[ ]

Changed Pages in ANP-10333NP-000

Identification of Modifications to ANP-10333NP-000

Item	Section(s) or Page(s)	Description of Change
1	7-4	Added reference to the response to RAI-5 and RAI-9 at the end of Section 7.3.1.2.
2	7-8	Text at the bottom of page 7-8 regarding the rod selection process has been superseded with information from the response to RAI-5. Additional guidance for determination of the initial flow (RAI-9) is provided by including an additional step in the selection process provided in the response to RAI-5. Figure 7.3 is not specifically used in the rod selection process but is retained for identification of outer rings of rods.
3	7-23	The determination of pin enthalpy based on **[** **]** in Section 7.6 is superseded with information in the response to RAI-6.
4	vi & 8-15	Figure 8.12 - The figure label has been corrected to state total enthalpy rather than prompt enthalpy. Updated table of figures title for Figure 8.12. Correction of typo not associated with an RAI.
5	9-16 through 9-19	Equation 5 and Tables 9.6 - 9.10 and associated text have been updated to reflect the response for RAI-6. Equation 6 has been removed as it is no longer required.
6	v & 9-21 and 9-22	Table 9.8 – The table label has been corrected from PNNL-1812 to PNNL-18212. Updated list of tables title for Table 9.8 on page v. Correction of typo not associated with an RAI. Correction also made in the third paragraph on page 9-22.

NOTE: The above listed changes have been highlighted within the document.

Figures

Table 7.1 BWR Critical Temperatures

[

]

7.3.1.2 Initial Flow and Power

The initial flow rate and power are determined based on sensitivity studies as shown in Sections 8.7.2.4 and 8.7.2.5. The event is assumed to occur with a xenon free core.

7.3.1.3 In-Cycle Evaluation Times

The analysis is performed at beginning of cycle, peak hot excess reactivity and end of full power conditions. These in-cycle conditions represent various core conditions and reactivity arrangements that may be encountered throughout the cycle.

7.3.1.4 Inoperable Control Rods

Inoperable rods locations are defined consistent with those allowed by plant technical specifications in such a manner to maximize the worth of the candidate rods. [

]

Table 7.3 Upper Group Rods Bounded By Lower Group

[

]

Following the establishment of the static rod worth values, the rods are identified for which detailed transient analyses are to be performed. The rod drop with the highest worth from BOC, PEAK, and EOFP are the first three rods selected. Selection of additional rods occurs after the transient evaluation of these three rods. The transient results from these first three cases are then used along with the static rod worth to define the most limiting in cycle condition. This is typically EOFP with top peaked reactivity distribution. [

] The outer ring evaluation is used to evaluate the impact of the CRDA on fuel rods with high exposure and cladding hydrogen content.

(Reference 6). [

] of the transient is also illustrated in Figure 7.8 which contains the local peaking factors for all rods in a lattice. Each color represents a different time step in the transient. Prior to the rod drop, the lower rod numbers have higher peaking than the higher rods in the lattice. As the blade drops out the peaking flattens across the lattice. The peaking for all rods remains constant during the peak pulse at approximately 0.8 seconds. Therefore, the enthalpy rise during the power pulse is proportional to the peaking factors during the time of the peak power. A linear adjusted pin peaking factor distribution is obtained by increasing the average enthalpy by [ ] and leaving the peak constant.

[

]

The application of this process is illustrated in Figure 7.9. The nominal values identified in Figure 7.9 assumes linear interpolation and extrapolation based simply on the peak and average enthalpy increases whereas the enhanced values are based on the increased average value.

[

]

Figure 8.12 Prompt Enthalpy Rise versus Channel Grouping

8.7.2 Plant Parameters & Initial Conditions

The impact of plant parameters and initial conditions are presented in this section.

8.7.2.1 Scram Speed

For the CRDA, the power pulse is turned around prior to the actual movement of the control blades. Therefore, use of conservative delay times and SCRAM speed is appropriate for the CRDA analysis.

8.7.2.2 Control Rod Drop Velocity

For BWRs 2 through 6, the control rod drop velocity is maintained with the control rod velocity limiter. Based on the results of velocity limiter tests, a conservative maximum rod velocity of 3.11 ft/s is used from the Reference 10 Velocity Limiter Tests. This was determined to be a bounding conservative velocity for BWRs 2 through 6. The use of the conservative drop velocity will be used for BWRs 2 through 6.

For BWRs without velocity limiters, an appropriate drop velocity must be determined for the application of this methodology.

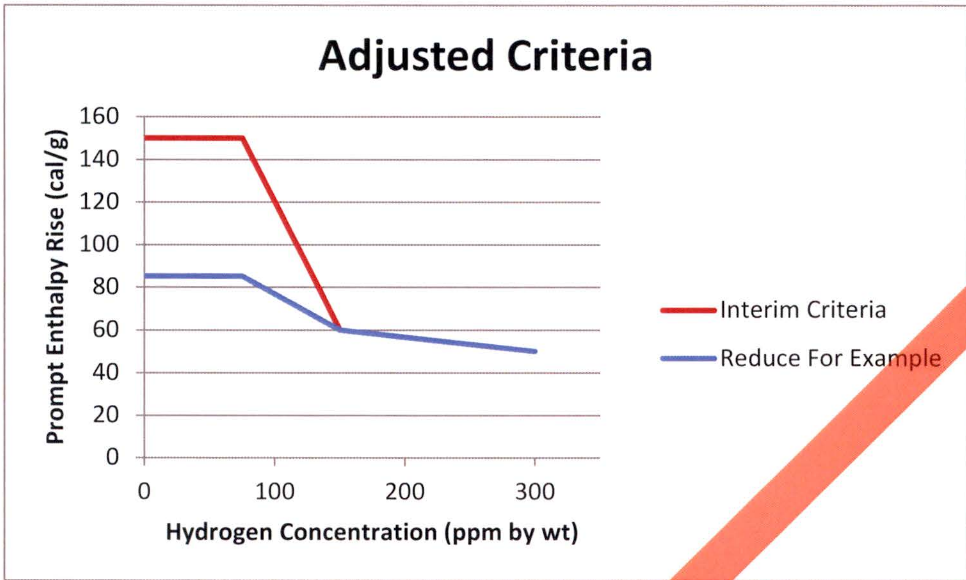

Figure 9.14 Adjusted Failure Criteria for Example Evaluation

Table 9.5 Assemblies with Failed Rods

Exposure	Sequence	Rod	Assemblies with Failures
EOFP	AG1234	118	V17118, V17134

In this example evaluation, a simplified assumption is made that all rods are at the maximum rod nodal hydrogen content rather than utilizing the individual pin hydrogen concentration values.

The number of failed rods were determined for each node by rearranging (Equation 3) from Section 7.6 and substituting the failure threshold Δh_f for Δh_i and nfail for prank. This allows determination of number of rods that fail.

[

]

Where:

Δh_f	Enthalpy failure threshold
Δhmax	Peak rod enthalpy increase
nfail	Number of rods that fail
Δhavg	Average rod enthalpy increase
nrods	The number of rods in the lattice

The equation used to determine the enthalpy for each rod is based on the peaking factor rank with [ ] conservatism applied to the average enthalpy and assumed that all rods have the maximum hydrogen concentration for the given node for this example.

Table 9.6 Determination of Rod Failures

[

]

* Failed rods are determined based on prompt enthalpy rise (ΔH_p).

The enthalpy dependent release fractions are determined based on the deposited enthalpy as determined from rearranging Equation 5 and determining the average enthalpy increase of the failed rods.

[

] (6)

The average pin rank for the failed rods is equal to one half of the total number of rods failed.

The example determination of release fractions is based on both SRP 4.2 and PNNL-18212 Revision 1.

Table 9.8 Fission Gas Release Fraction (PNNL-1812 Rev. 1)

[

]

For each assembly, the enthalpy dependent release terms are determined along with the ratio of the release fractions to the Reg Guide 1.183 release fractions for the rod drop.

Table 9.9 Fission Gas Release Fraction Ratios

[

]

The number of failed rods are tabulated for those assemblies which have failed rods indicated. This number of failed rods is then multiplied by the ratio of the transient release fraction to the reference release fraction in order to establish an equivalent number of failed rods. The maximum ratio of all groups (I-131, I-132, Kr85, Nobles, Halogens, Alkali Metals) is used to determine the equivalent number of rods.

For assembly V17118, 19 rods are identified as exceeding the reduced failure threshold. Utilizing the ratios of the release fractions, the actual number of failed rods is determined using the two different transient release fractions. The maximum ratio of release fraction to the reference release fraction is [ ] based on other Halogens using the steady state release fractions of Table 3 of RG1.183 and the transient release term from SRP4.2 Appendix B. The maximum ratio of release fraction to the reference release fraction is [ ] for Alkali Metals using the transient release term from PNNL-1812 Rev1.

Utilizing the release fraction ratios, an equivalent number of rod failures is determined for the two different transient release terms. The same process is repeated for each of the assemblies for which rod failures are identified for the given rod drop. Table 9.10 contains the total equivalent rod failures for all bundles for the example drop.