FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2103, REVISION 3,

"REALISTIC LARGE BREAK LOCA METHODOLOGY

FOR PRESSURIZED WATER REACTORS"

AREVA NP, INC.

PROJECT NO. 728

Enclosure

EXECUTIVE SUMMARY

Topical Report (TR) EMF-2103, Revision 3, "Realistic Large Break LOCA [loss-of-coolant accident] Methodology for Pressurized Water Reactors [(PWRs)]," was submitted to the U.S. Nuclear Regulatory Commission (NRC) by AREVA NP, Inc. (AREVA), for review by letter dated September 13, 2013. The principal purpose of the revision was to address numerous long-standing issues that had been identified with EMF-2103(P)(A), Revision 0, in the 10 years between the 2003 approval of Revision 0 and the submittal of Revision 3. The NRC staff review and basis for approving EMF-2103, Revision 3, is provided in this safety evaluation (SE). This SE is also intended to provide guidance to the NRC staff for reviewing plant-specific requests for licensing actions that are based on, or supported by, EMF-2103, Revision 3.

Chapter 1 of this SE provides a brief revision history of EMF-2103, and its NRC staff review evolution. This introductory chapter also summarizes the model changes that were made in Revision 3, relative to Revision 0. The chapter further serves to identify the focus of the NRC staff review. The focus was on changes that have been made to EMF-2103 since the NRC staff approval of Revision 0, which have not received any sort of prior review in plant-specific licensing applications (refer to Section 1.2 of the SE for additional detail regarding the NRC staff review scope).

Chapter 2 summarizes the regulatory basis for the NRC staff review, including the applicable NRC requirements, and the associated review guidance documents.

The technical evaluation performed by the NRC staff is provided in Chapter 3. Since this evaluation was performed on a revision to a method that has already received NRC approval for use, the evaluation is structured based on specific evaluation model changes rather than a complete review of the entire method. Each sub-section within Chapter 3 evaluates a specific evaluation model change, and provides a basis for determining the acceptability of the change. In some cases, the NRC staff approval requires constraints in the form of review limitations. Where applicable, the justifications for such constraints are discussed in the technical evaluation.

Chapter 4 presents a summary of the review limitations that are applicable to Revision 3 of EMF-2103. This summary is intended to: 1) clarify the constraints on the NRC staff approval, 2) to review the justification for and, in some cases, expected result of the review constraint, and 3) to identify what type of information or justification would be required in plant-specific instances where the constraint cannot be satisfied or does not apply. Chapter 4 is intended to serve as guidance for preparing plant-specific submittals and for reviewing such submittals.

Chapter 5 presents the overall conclusion of the NRC staff review. It notes that EMF-2103, Revision 3, is acceptable for referencing in licensing applications to the extent delineated in this SE, subject to the limitations specified in Chapter 4.

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1.0 INTRODUCTION

By letter dated September 13, 2013, AREVA, submitted TR EMF-2103, Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors".[1] The TR describes a revision to an emergency core cooling system (ECCS) evaluation model (EM) that AREVA developed to analyze PWR LOCAs. For the sake of brevity and specificity, this safety evaluation (SE) will hereafter refer to the TR as EMF-2103, Revision 3, and to the EM as realistic large break LOCA (RLBLOCA).

During the NRC staff review, the TR was supplemented by four letters. AREVA submitted two letters dated January 16, 2015.[2, 3] These first two letters provided a response to a request for additional information (RAI), errata sheets, and revised sample problems for the TR. AREVA submitted a third letter, dated February 16, 2016, which provided an updated RAI response, and superseded the first RAI response.[4] The fourth letter, dated February 19, 2016, provided updated errata sheets.[5]

1.1. <u>REVISION HISTORY</u>

The original revision (Revision 0) of EMF-2103 was submitted to the NRC for review by letter dated August 20, 2001.[6] Following the NRC staff review, the approved version of EMF-2103(P)(A), Revision 0, was published in April 2003.[7] The content in EMF-2103(P)(A), Revision 0, was supported by two additional documents. These included EMF-2100(P), Revision 4, "S-RELAP5 Models and Correlations Manual," and EMF-2102(P), Revision 0, "S-RELAP5: Code Verification and Validation."[8, 9]

In order to clarify generically several issues that arose during plant-specific reviews, AREVA subsequently submitted EMF-2103, Revision 1, to the NRC for review by letter dated August 9, 2004.[10] EMF-2103, Revision 1, constituted a brief appendix which would have been added to the original TR. AREVA opted to resolve the issues documented in EMF-2103, Revision 1, along with several others identified during the NRC staff review by use of a Transition Package. The Transition Package constituted a standard set of plant-specific information that would be provided to support plant-specific review of EMF-2103 implementation requests. AREVA subsequently withdrew EMF-2103, Revision 1, from NRC staff review.[11]

AREVA submitted EMF-2103, Revision 2, to the NRC for review by letter dated November 15, 2010.[12] The update associated with EMF-2103, Revision 2, would have incorporated the improvements associated with Revision 1, along with additional issues identified during the NRC staff review of Revision 1, and several new enhancements to the methodology for addressing severe results. Particularly, analyzing the behavior of ballooned and ruptured fuel cladding segments was addressed. AREVA submitted a supplement to this revision by letter dated December 20, 2011.[13] The supplement addressed the effects of nuclear fuel thermal conductivity degradation (TCD). This TR, as supplemented, was in a similar, modular format to EMF-2103(P)(A), Revision 0, in that it referenced EMF-2100(P), Revision 14, "S-RELAP5 Models and Correlations Code Manual," and EMF-2102(P), Revision 1, "S-RELAP5: Code Verification and Validation."[14, 15]

By letter dated September 13, 2013, AREVA submitted EMF-2103, Revision 3. The intent of Revision 3 was to supersede the Revision 2 submittals and to provide a comprehensive description of the EM in a stand-alone document. The stand-alone format of Revision 3 greatly simplified the NRC staff review of the updated EM.

1.2. SUMMARY OF MODEL CHANGES

Since Revisions 1 and 2 of EMF-2103 were withdrawn or superseded prior to completion of the NRC review, the NRC-approved version of RLBLOCA remains that documented in EMF-2103(P)(A), Revision 0. Thus, this review addresses those elements of the EM that have been updated since the NRC approved Revision 0. A distinction is made between changes associated with the Transition Package for EMF-2103(P)(A), Revision 0, and changes that are newly introduced in EMF-2103, Revision 3. Transition Package changes have already been reviewed and accepted by the NRC staff in prior, plant-specific licensing actions, whereas newly introduced changes have not been previously reviewed.

1.2.1. Transition Package Changes

Many of the changes have already been implemented and accepted by the NRC staff in plant-specific analyses. These changes are considered those associated with the Transition Package for EMF-2103 and are shown in Table 1.

Prior Change	Purpose
Cold Leg Condensation Model	Cold leg condensation modeling has been modified to increase susceptibility to downcomer boiling.
Second Cycle Fuel	The evaluation model includes simulation of second cycle fuel to determine whether second-cycle fuel exhibits limiting attributes.
Break Modeling	The distinction between split and double-ended guillotine breaks has been changed to introduce more independence between break geometry and break area.
Decay Heat Simulation[A]	A sampling approach to address decay heat uncertainty has been replaced with a deterministic decay heat curve that incorporates analytic margin for uncertainty associated with the curve.
GDC 35 Compliance	Disposition for limiting single failure is unchanged; the sample is analyzed **[** **]**.
Operating Power Level	Rather than sample the power level within the applicable uncertainty range, the power level is set to a deterministic value that includes plant design basis calorimetric uncertainty.

Table 1. Evaluation Model Upgrades
Associated with the Transition Package for EMF-2103(P)(A), Revision 0.

Given their prior implementation and approval in plant-specific licensing actions, the NRC staff did not review the changes contained in Table 1 in detail as a part of the present review effort.

[A] Additional information regarding the decay heat simulation is provided in the response to RAI 10. This approach has been accepted in recent applications of EMF-2103(P)(A), Revision 0. Refer, for example, to Section 3.5.3, “Decay Heat,” of the Safety Evaluation for Amendment 138 to Facility Operating License NPF-63, dated May 30, 2012; Agencywide Documents Access and Management System Accession No. ML12076A103.

Some discussion is provided regarding General Design Criterion (GDC) 35 compliance (see Section 3.4.3).

1.2.2. Novel Changes Associated With EMF-2103P, Revision 3

In addition to incorporating the changes listed above in Table 1 into the EM documentation, AREVA also proposes to incorporate eleven additional model upgrades to RLBLOCA. Table 2 shows these EM updates.

New Feature	Purpose
Wong-Hochreiter Heat Transfer Correlation	Dispersed flow heat transfer ahead of quench front; replaces the Forslund-Rohsenow and Sleicher-Rouse heat transfer correlations
Rod-to-Rod Radiation Model	Upgrades improve prediction of reflood heat transfer; separate radiation enclosures are provided for each rod
Statistical Analysis	Replaces Wilks's theorem-based tolerance interval for a univariate peak cladding temperature (PCT) distribution with a simultaneous confidence interval estimated for tri-variate distributions of PCT, maximum local oxidation (MLO), and core-wide oxidation (CWO)
Upgraded fuel performance model	Replaces legacy models for fuel pellet and cladding initial conditions and modeling
Interfacial Drag Package	Improves logic for transition between flow regimes to cover a wider range of experimental data
Local Cladding Oxidation	Limiting oxidation results accounts for operational (pre-transient) and interior cladding oxidation for ruptured fuel rods
Fuel Rod Swelling, Rupture, and Relocation	Models for fuel rod swelling and rupture, and pellet relocation, have been added to the EM
Steam Absorptivity	A pressure limit was set on computing the vapor absorption coefficient; correction to the steam absorptivity implemented
Core Nodalization Shift	Node boundaries aligned with bottom of grid spacers, rather than grid centerline
Kinetic Droplet Model	Model adds credit for improved interfacial heat transfer based on droplets shattering on grid structures and deformed fuel rods
Mist Flow Interphase Heat Transfer	Accuracy improved for interphase heat transfer modeling for mist flow, relative to separate effects reflood test data

Table 2. List of EMF-2103, Revision 3, Model Improvements Relative to EMF-2103(P)(A), Revision 0.

This SE is limited in scope to the changes listed in Table 2. Notwithstanding the Transition Package changes listed in Table 1, all remaining portions of the EM as described in EMF-2103(P)(A), Revision 0, remain applicable and acceptable to the extent delineated in the NRC staff SE approving the TR. Prior limitations applicable to Revision 0 were either incorporated into the limitations presented in Section 4.0 of the present SE, or addressed by content in the body of Revision 3 of the TR, or by modeling guidelines in Appendix A of Revision 3 of the TR. Thus, the limitations described in Section 4.0 of the present SE are the only limitations that apply to the current method.

1.3. SUMMARY OF NRC STAFF REVIEW APPROACH

The NRC staff drew on the extensive experience accumulated with reviewing requests for licensing actions supported by RLBLOCA in the approximately 12 years between the NRC staff approval of EMF-2103(P)(A), Revision 0, and the completion of the SE for EMF-2103, Revision 3, as summarized in Section 1 of this SE. The regulatory basis for the NRC staff review is described in Section 2 of this SE. The technical evaluation is provided in Section 3. Section 4 of this SE provides a new set of limitations applicable to future implementations of RLBLOCA. Section 5 presents the overall conclusion for the NRC staff review.

2.0 REGULATORY BASIS

The RLBLOCA EM was developed in accordance with the regulatory requirements established in Title 10, "Energy," of the *U.S. Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" (10 CFR 50.46). In developing RLBLOCA, AREVA considered guidance contained in two NRC regulatory guides (RGs). These RGs include: (1) RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance" and (2), RG 1.203, "Transient and Accident Analysis Methods."[16, 17]

The NRC staff reviewed EMF-2103, Revision 3, to determine whether RLBLOCA is an acceptable EM as set forth in 10 CFR 50.46. In its review, the NRC staff relied on the regulatory guidance described above, as well as applicable chapters contained in NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants, Light Water Reactor Edition." These chapters included Chapter 6.3, "Emergency Core Cooling System," Chapter 15.0.2, "Review of Transient and Accident Analysis Methods," and Chapter 15.6.5, "Loss-of-Coolant-Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."[18, 19, 20]

As described in Section 2.1 below, the RLBLOCA EM is required to provide an estimated uncertainty associated with its results and comparisons must be made to experimental data to show that its results realistically describe reactor behavior under hypothetical LOCA conditions. The NRC provides an acceptable approach to determining uncertainty associated with safety analysis methods in NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident."[21] In addition, the NRC provides a compendium of experimental data pertinent to ECCS EMs in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis."[22] These documents provide additional, supporting guidance for the documentation contained in EMF-2103, Revision 3, and the related NRC review.

2.1. APPLICABLE REGULATORY REQUIREMENTS

Holders of operating licenses under 10 CFR Part 50 are required, pursuant to 10 CFR 50.34, to submit final safety analysis reports (FSARs) to the NRC. In part, 10 CFR 50.34(b)(4) states:

> …Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46…

AREVA developed RLBLOCA as a realistic, best-estimate, or best-estimate plus uncertainty (BEPU) EM [B] to be used to evaluate ECCS performance at PWRs. The enabling regulatory framework is established in Paragraph (a)(1)(i) of 10 CFR 50.46, which states, in part:

> Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated…. [T]he evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model….

Paragraph (b) of 10 CFR 50.46, as referenced in the above excerpt, provides the acceptance criteria for ECCS evaluation. The RLBLOCA EM is intended to demonstrate compliance with 10 CFR 50.46(b)(1-3), as follows:

> (b)(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.
>
> (2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation….

[B] Although this SE favors the term, "realistic," the terms, "best-estimate," and, "BEPU," are also frequently used in a similar context. It should be understood that, in order for the analytic results to be considered acceptable for the purposes of demonstrating compliance with 10 CFR 50.46 requirements, the realistic, or so-called best-estimate, results must be expressed *at some upper level that includes an allowance for estimated uncertainty*. The distinction among these terms is discussed in further detail in Enclosure C, "ACRS Comments on Code Scaling, Applicability and Uncertainty Associated with the use of Realistic ECCS Evaluation Models," to SECY-88-162, "Revision of the ECCS Rule Contained in Appendix K and Section 50.46 of 10 CFR Part 50."

> (3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Chapter 3 of the EMF-2103, Revision 3, states that the other two criteria contained in 10 CFR 50.46(b), regarding coolable geometry and long-term core cooling, are treated separately during plant-specific evaluations. This regulatory basis is reflected as Limitation 1, "Acceptance Criteria Satisfied by this Evaluation Model," in Chapter 4 of this SE.

Additional requirements, which govern assumptions that must be employed in the ECCS evaluation, are contained in 10 CFR Part 50 Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 35, "Emergency core cooling," which states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available)the system safety function can be accomplished, assuming a single failure.

These requirements – 10 CFR 50.34, 10 CFR 50.46, and GDC 35 – form the regulatory basis for the NRC staff review.

2.2. STANDARD REVIEW PLAN GUIDANCE

As discussed in Section 2.0, the NRC staff performed its review using the SRP. Applicable chapters included Chapter 6.3, "Emergency Core Cooling System," Chapter 15.0.2, "Review of Transient and Accident Analysis Methods," and Chapter 15.6.5, "Loss-of-Coolant-Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary." While Chapter 15.0.2 provides the overarching guidance for the EM review, additional guidance as it pertains to plant-specific considerations of ECCS evaluation is contained in Chapters 6.3 and 15.6.5.

The NRC staff relied principally on the overarching guidance in SRP Chapter 15.0.2. This SRP chapter documents the areas of an EM to be reviewed, including: (1) Documentation; (2) Evaluation Model; (3) Accident Scenario Identification Process; (4) Code Assessment; (5) Uncertainty Analysis; and (6) Quality Assurance Plan. The SRP chapter provides guidance intended to focus review resources, primarily, on ensuring: (1) that the scenario to be analyzed is appropriately specified, (2) that previously un-reviewed aspects of the EM that are pertinent to the scenario under consideration are appropriately reviewed, and (3) that the application of the model evaluates uncertainties appropriately. Accomplishing these review objectives provides assurance that the results obtained from RLBLOCA, with uncertainty accounted for, demonstrate with high probability that the criteria of 10 CFR 50.46(b)(1-3) are not exceeded, consistent with 10 CFR 50.46(a)(1)(i) requirements. This SE is formatted in a manner largely consistent with SRP 15.0.2 guidance.

Chapter 6.2 of the SRP provides guidance for performing the system review of the ECCS. The guidance is largely focused on the hardware itself, rather than the ECCS evaluation. However, the NRC staff considered this guidance when reviewing the EM, code assessment, and demonstration analyses to ensure that the relevant hardware requirements are accurately reflected in the RLBLOCA EM.

Chapter 15.6.5 of the SRP provides guidance for performing reviews of LOCA analyses for the spectrum of postulated pipe breaks within the reactor coolant pressure boundary. This SRP chapter supplements the guidance contained in SRP 15.0.2, by addressing plant-specific aspects that must be considered when reviewing ECCS evaluations. The NRC staff considered this guidance when performing its review of the demonstration analyses, the model application, and licensing considerations discussed in Chapters 3 and 4 of this SE, but like Chapter 6.2, the guidance was considered more generally.

These SRP chapters provide guidance to the NRC staff in performing the safety review of EMF-2103, Revision 3. They describe methods or approaches that the NRC staff has found acceptable for meeting NRC requirements. For the purposes of reviewing an ECCS EM, the SRP is not considered a complete, standalone reference to provide all the required review guidance. Additional documents discussed in Sections 2.3 and 2.4 of this SE also apply.

2.3. NRC REGULATORY GUIDES

Regulatory Guides provide guidance to licensees, applicants, and vendors on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. The particular guides applicable to ECCS evaluation model development include RG 1.203 and RG 1.157.

Regulatory Guide 1.203 is the analog to SRP 15.0.2. It describes a process that the NRC staff considers acceptable for use in developing and assessing EMs that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

Regulatory Guide 1.157 was developed in concert with the revision to 10 CFR 50.46 that permitted the use of realistic ECCS EMs. This RG describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a LOCA and for estimating the uncertainty of that calculation.

2.4. ADDITIONAL LITERATURE

2.4.1. Code Scaling, Applicability, and Uncertainty

Prior revisions of EMF-2103 had been structured consistently with NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation to a Large-Break, Loss-of-Coolant Accident." This NUREG/CR report describes an uncertainty evaluation methodology called code scaling, applicability, and uncertainty (CSAU).

Although EMF-2103P, Revision 3, departs slightly from the CSAU structure in favor of that delineated in RG 1.203, many of the steps of the CSAU methodology remain applicable. The NRC staff considered the information contained in NUREG/CR-5249 in its review.

2.4.2. Compendium of ECCS Research

The requirements contained in 10 CFR 50.46(a)(1)(i) state, in part, that "comparisons to applicable experimental data must be made…." Accordingly, the guidance in RG 1.203 and RG 1.157 frequently indicate that models, correlations, formulas, etc., will be considered acceptable, provided they are checked against or compared to relevant data sets. The NRC provides a set of such relevant data sets in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis."

2.4.3. Additional Sources

The NRC-published regulatory guidance and technical reference materials specifically pertinent to ECCS EMs were all published prior to 1990. The state of the art for LOCA research and EM development has evolved since then. Hence, additional sources of relevant research, which are not necessarily included in the literature reviewed above, are considered as appropriate.

3.0 TECHNICAL EVALUATION

3.1. HEAT TRANSFER MODELS

3.1.1. Wong-Hochreiter Heat Transfer Correlation

AREVA has replaced the Forslund-Rohsenow and Sleicher-Rouse correlations for dispersed flow film boiling for core bundle heat structures with a modified version of the Wong-Hochreiter heat transfer model. Wong-Hochreiter is a set of four correlations, each applicable in a different Reynolds number turbulence regime. These correlations were published in NUREG/CR-1533, "Analysis of the FLECHT SEASET [Full-Length Emergency Core Heat Transfer Separate Effects and System Effects Test] Unblocked Bundle Steam Cooling and Boiloff Tests."[23] The first is intended for laminar flow. The second is a linear interpolation intended to provide a smooth transition between the laminar flow region and an intermediate region. The correlation for the intermediate region is similar to Dittus-Boelter, but produces better agreement with FLECHT-SEASET data. The final correlation, at the highest Reynolds number values, is the Dittus-Boelter correlation, whose performance in a rod bundle geometry improves with increasing Reynolds number.

In implementing the Wong-Hochreiter model, AREVA made several modifications to the correlations presented in NUREG/CR-1533. AREVA removed the transition (**[** **]**) correlation and instead uses the Holman correlation for natural convection to vapor. Further, AREVA eliminated the Reynolds number-based selection logic and instead selects the highest value of the four coefficients. The substitution of the Holman natural convection correlation for the linear interpolation function results in the greatest value approach being reasonably equivalent to the Reynolds number based approach. This is addressed in the following paragraph. The intermediate and high-Reynolds number correlations also employ a correction factor for high wall-to-vapor temperature differences. Otherwise, the correlations intersect at the limiting Reynolds number appropriate for the particular correlation in use.

The NRC staff reviewed the substitution of the Holman correlation in greater detail, including information provided in response to RAI 11. The FLECHT-SEASET Steam Cooling tests included two runs, 36766 and 36867, for which the steady-state Reynolds number was in the range where the Holman correlation applies. The data from these tests **[**

] on Figure 2.11-2 of the RAI response. If the selection logic were inappropriately implemented (i.e., selecting the highest heat transfer coefficient results in overestimation of the heat transfer), excessive spread, particularly below the line of perfect agreement, would be apparent. However, the figure shows that this is not the case, indicating that the logic performs acceptably. As further confirmation, Figures 8.2-118 and 8.2-119 of EMF-2103, Revision 3, compare S-RELAP5 heat transfer coefficients to FLECHT-SEASET Tests 31805 and 31504, respectively, showing that the coefficients trend well with the data early in the transient at the PCT elevation. More summarily, the updated assessment provided in response to RAI 19 shows that the code generally predicts PCTs consistent with or greater than the measured PCTs in the upper elevations of the low reflood rate FLECHT-SEASET tests. For example, refer to Figures 2.19-36 thru 2.19-38 and 2.19-64 thru 2.19-66 of the response to RAI 19 for elevation-dependent PCT comparisons from runs 31504 and 31805, respectively.

The NRC staff reviewed the theoretical basis for the correlation, which is presented in NUREG/CR-1533, also considering additional information provided in response to RAI 11. In response to RAI 11, AREVA provided additional data comparing the FLECHT-SEASET boiloff tests, which were used to define the Wong-Hochreiter correlation, to simulations of the tests performed using S-RELAP5. The code predictions included those modeled using the correlation as modified in S-RELAP5 and those modeled using the Sleicher-Rouse heat transfer correlation, which was used in previous applications of EMF-2103. The use of modified Wong-Hochreiter indicated improved agreement with data compared to Sleicher-Rouse. The response to RAI 11 provides a quantitative assessment of the agreement, and it should be noted that both models tend to under-estimate the rod bundle heat transfer slightly. This tendency is an indication of slightly conservative modeling.

Despite the noted level of agreement between S-RELAP5 analysis and the FLECHT-SEASET boiloff test data, some limitations to the comparison should be noted. First, the comparison is based on planar average thermocouple readings. The comparison does not account for sub-channel flow variations or for the uncertainty associated with individual thermocouple readings. Second, the comparison was made at five different heights for eight separate tests (i.e., 40 data points). As noted in NUREG/CR-1533, the two lower elevations never exceeded dryout conditions, suggesting that the heat transfer mode was either subcooled or nucleate boiling. These heat transfer modes are calculated using different correlations within S-RELAP5 that are also modifications to the Dittus-Boelter convective heat transfer correlation. That is, they perform similarly but modified Wong-Hochreiter should not be in use at these elevations, and as such, their agreement to data should not be considered in the evaluation.

Despite the limitations noted above, the Wong-Hochreiter correlation is intended to calculate rod bundle heat transfer characteristics, meaning that the comparison to average thermocouple data at each elevation is appropriate. In addition, the comparison is intended to gauge the level of agreement, but is a separate exercise from the disperse flow and film boiling heat transfer uncertainty evaluation. Thus, based on the improved level of agreement observed in comparing the S-RELAP5 analysis using Wong-Hochreiter to FLECHT-SEASET boiloff test data, relative to that obtained using Sleicher-Rouse, the NRC staff determined that the use of Wong-Hochreiter is acceptable in S-RELAP5. The NRC staff also notes that the comparison is based on a data set specifically included in Regulatory Position 3.9.2, “Experimental Data for Post-CHF Heat Transfer,” of RG 1.157.

In its review of EMF-2103, Revision 3, the NRC staff also observed that code is assessed against Oak Ridge National Laboratory (ORNL) Thermal Hydraulic Test Facility (THTF) level swell tests.[24] As noted in the comparison provided in response to RAI 24, a revision to the pressure limit for vapor absorptivity has also improved the code’s predictive capability with respect to the predicted fuel rod temperature response. The revised model shows general agreement with the data. Further, the confirmation of the FILMBL and DFFBHTC[C] uncertainty evaluation was based on THTF transient boiloff and reflood tests. Comparisons to these tests provided in EMF-2103, Revision 3, Section 8.2.1, showed that the code produced conservative estimates of the THTF fuel rod temperature response, even when lower bound uncertainty multipliers (i.e., those that provide a lower bound of the fuel rod temperature response) are used.

Based on the review described above, the NRC staff concluded that the implementation of Wong-Hochreiter within S-RELAP5 is acceptable in that its theoretical basis is valid. The updated code assessment, discussed in Section 3.5 of this SE, confirms that the correlation performs acceptably against integral effects tests. This additional confirmation provides assurance that the heat transfer correlation does not cause non-conservative predictive results, such as early rod quench or over-estimation of heat transfer coefficients.

3.1.2. Rod-to-Rod Radiation Model

As originally approved, EMF-2103, Revision 0, did not account for rod-to-rod radiation. Issues with this approach were identified in plant-specific reviews. As a result, AREVA incorporated a simplistic rod-to-rod radiation model in the EMF-2103 Transition Package. In EMF-2103, Revision 3, AREVA has implemented the rod-to-rod radiation model contained in RELAP5-Mod3.3 in the S-RELAP5 code. The model provides a separate radiation enclosure for each surface.

As stated in EMF-2103P, Revision 3, Chapter 2:

> A rod-to-rod radiation model has been incorporated into the methodology and the reflood heat transfer benchmarking has been redone. This upgrade was incorporated to more accurately assess reflood heat transfer by recognizing the individual components of the process. The alteration is presented in Section 7.6.8.2 and assessed in Sections 8.2.5, 8.5.2.4, and 8.6.2.1. The model was subsequently revised in Revision 3 to implement separate radiation enclosures for each burned rod, rather than

[C] The identifiers “DFFBHTC” and “FILMBL” refer to multipliers that are used to de-bias the S-RELAP5-calculated heat transfer coefficients in the disperse flow and film boiling heat transfer regimes. In the statistical combination of uncertainty, the exact values of the multipliers are sampled from a distribution defined by the code assessment. The nominal values and distributions for the multipliers are determined from simulation of integral effects tests, and confirmed through additional simulation exercises, including benchmarks of the statistical methodology and demonstration plant analyses. Section 3.4.1 of this SE reviews the updated assessment that determined the central values and probability density functions for these multipliers.

one enclosure for all burned rods. The model change is presented in Section 9.0 and its impact qualitatively assessed in Section 8.1.5.

The incorporation of a rod-to-rod radiation model enables S-RELAP5 to assess convective heat transfer in the dispersed flow regime separately from surface-to-surface radiation. In EMF-2103(P)(A), Revision 0, the effects of surface-to-surface radiation were not explicitly calculated, but the uncertainty assessment for dispersed flow convective heat transfer included the radiation effects in the FILMBL and FRHTC (which was replaced by DFFBHTC) multipliers. The present inclusion of rod-to-rod radiation modeling enables the use of lower heat transfer coefficient multipliers.

Treating the two heat transfer modes separately is viewed as an improvement to the predictive capability of S-RELAP5. The convective heat transfer rate varies linearly with temperature difference, whereas the radiative heat transfer rate varies with temperature difference to the fourth power. Noting the difference in behavior of the two coefficients, it is difficult to characterize radiative heat transfer accurately by including it in the convective heat transfer formulation. Using explicit models for both is thus considered more realistic.

AREVA included the updated rod-to-rod radiation model in its updated code assessment. Of particular note, the response to RAI 6 includes the results of the S-RELAP5 simulation of FLECHT-SEASET Test 31504, which is an appropriate benchmark because of its low reflood rate. The low reflood rate causes the PCT location to remain uncovered for a longer amount of time as compared to the faster reflood rate tests. The response to RAI 6 indicates that the contribution of radiation heat transfer calculated by S-RELAP5 is in line with the analyses associated with FLECHT-SEASET.[25] In addition, in response to RAI 19, AREVA provided a comparative analysis using S-RELAP5 both with and without the rod-to-rod radiation model in Figure 2.19-5. The figure shows that, for Test 31504, the influence of rod-to-rod radiation on the overall predicted PCT is quite minor.

The specific modeling of rod-to-rod radiation is described in several sections of EMF-2103, Revision 3, that were not mentioned by AREVA in the paragraph quoted above. Specifically, Section 3.1.3.3.7 of EMF-2103, Revision 3, describes the modeling approach. Modeling guidelines are also provided on Pages A-67 through A-71 of Appendix A to EMF-2103, Revision 3. The approach relies on several auxiliary rods, one for each radiation enclosure, for the hot assembly model. This passage of EMF-2103, Revision 3, notes that for the FLECHT-SEASET assessment described in the preceding paragraph, **[**

]

(EMF-2103, Revision 3, Page 3-25). **[**

], reduces the overall ability of the hot rod to radiate to the auxiliary rod. When the DFFBHTC and FILMBL multipliers for the hot assembly are otherwise held constant, this results in a conservative estimation of the effect of radiation heat transfer on the overall transient performance. It also constrains the heat transfer coefficient multipliers, DFFBHTC and FILMBL, so that there is not a compensating increase in heat transfer to the coolant. Since this **[** **]** value is in line with the uncertainty associated with monitoring power peaking, the NRC staff determined that the value is reasonable. Since the **[**

], the NRC also agrees that the auxiliary rod modeling approach is generally conservative relative to the code assessment, and as such, has determined that it is acceptable.

The inclusion of the rod-to-rod radiation model in the updated code assessment demonstrates that the model treats the phenomenon appropriately. The additional results, which show the model performance is consistent with other analyses, confirm the acceptable performance. Based on these considerations, the NRC staff determined that the rod-to-rod radiation model is acceptable for inclusion in RLBLOCA.

3.1.3. Steam Absorptivity

EMF-2103, Revision 3, states the following regarding changes to the steam absorptivity model:

> A correction to the steam absorptivity was made. In computing the vapor absorption coefficient, the pressure is conservatively truncated at 150 psi [pounds per square inch]. This alteration is presented in Section 7.6.8.1 and its impact qualitatively assessed in Section 8.1.5.

AREVA provided additional correction and clarification to this modification in response to RAIs 24 and 33. In response to RAI 24, AREVA clarified that the pressure is truncated to **[ ]**, instead of 150 psi. In response to RAI 33, AREVA also clarified that EMF-2103, Revision 3, contained a leading constant to Equation 7.540 that was different from the value previously used in earlier revisions of EMF-2103 and supporting documentation. AREVA further clarified that the previous value of the constant contained a unit conversion error, which has been corrected in EMF-2103, Revision 3. The response to RAI 33 also noted that both of these changes have been dispositioned as changes in accordance with 10 CFR 50.46(a)(3) reporting requirements and their estimated effects in plant-specific applications have been reported to the NRC.[D]

AREVA re-ran THTF level swell benchmark tests to illustrate the effects of this change, as documented in the response to RAI 24. The results contained in Figures 2.24-5 and 2.24-6 of the RAI response showed that, in the vapor region of THTF tests 3.09.10j and 3.09.10.m, the implementation of the pressure limit had the effect of elevating the rod surface temperature and bringing it into closer alignment with the test data, when compared to the 150 psi limit documented in EMF-2103, Revision 3.

The NRC staff reviewed the information in response to RAIs 24 and 33 and concluded that the modification to the treatment of steam absorptivity was acceptable based on the following three considerations. First, the change in the leading constant to Equation 7.540 was implemented to correct a unit conversion error. Second, the re-benchmarking exercise showed that the lower pressure limit brought the THTF analysis into better agreement with the data by predicting higher cladding surface temperatures in the vapor region of the tests. Third, these changes appear to correct code errors and have already been dispositioned as such in plant-specific applications of EMF-2103(P)(A), Revision 0.

3.1.4. Mist Flow Interphase Heat Transfer

The vapor side heat transfer coefficient, described in Section 7.5.4.3 of EMF-2103, Revision 3, is given by Equation 7.383, on Page 7-133. The equation is repeated below:

[D] This statement indicates that these changes have already been incorporated into EMF-2013(P)(A), Revision 0, despite not necessarily having received, nor required, prior NRC staff review and approval.

[ ]

where [

][E] It is expressed more simply if an additional parameter, B, is introduced:[F]

[ ]

The equation, along with much of the discussion surrounding it, is unchanged between EMF-2100(P), Revision 4, and EMF-2103, Revision 3. However, AREVA has changed the value of a in the equation and has introduced an alternative formulation for the mass transfer scaling factor, F.

In the prior revision, the values of a and F were based on a paper by **[ ]**. The value of a was **[ ]** and the value of F was given by the following relation:

[ ]

Both documents (EMF-2100(P), Revision 4, and EMF-2103, Revision 3) note, "**[ ]**." The value has been reduced from **[ ]**.

Regarding F, AREVA opted to apply a **[ ]**:

[ ]

where **[ ]**.

The response to RAI 3 investigates the overall sensitivity of the vapor-side heat transfer coefficient to the value of B. Using the **[ ]** of F, AREVA applied a multiplicative factor to B, so that the effect of gradually reducing the value of B could be studied. The response to RAI 3 shows that as B **[ ]**, the term

[ ]

[E] The response to RAI 3 clarified **[ ]**.

[F] AREVA's response to RAI 4 notes that **[ ]**. This detail is omitted from the submitted copy of EMF-2103, Revision 3. The NRC staff also notes that the **[ ]** was actually dependent on the state properties of the specific tests, a detail also omitted from EMF-2103, Revision 3.

[ ]. This effect is shown in Figure 2.3-1 of the RAI response. Thus, for any given void fraction, the effect of **[ ]**. This trend, in turn, causes an increase in the vapor-side heat transfer.

AREVA further implemented the multiplicative factor in S-RELAP5 and simulated FLECHT-SEASET Test 31504. The results showed that, when applied in analysis, minimizing **[ ]**. This effect can be observed by inspection of Figures 2.3-2 and 2.3-3 of the RAI response.

In the response to RAI 3, AREVA demonstrated that the **[ ]**. However, AREVA's analyses also indicated that the **[ ]**.

In the response to RAI 4, AREVA performed a broader comparison of heat transfer characteristics using the previous **[ ]** formulation for the value of B, to those obtained using the **[ ]** for B. Consistent with the results depicted in response to RAI 3, the comparison showed that the **[ ]** in FLECHT-SEASET Test 31504.

The response to RAI 4 also provides the results of two additional sensitivity studies investigating the use of the **[ ]**. The first examined the **[ ]**. The second compared the **[ ]**.

Finally, the NRC staff compared both values of **[ ]**, for thermal hydraulic conditions associated with FLECHT-SEASET Test 31504. Based on the NRC staff evaluation, the **[ ]** factor, further suggesting that the **[ ]**.

Based on the review described above, the NRC staff determined that the modification to the vapor-side mist flow interphase heat transfer coefficient is acceptable. The following three considerations apply. First, AREVA provided information showing that the specific coefficients used in the new formulation generally serve to scale the heat transfer coefficient by some value less than one and that the scaling produces a net PCT increase in the FLECHT-SEASET benchmark to Test 31504. Second, the NRC staff compared the two F values for conditions associated with the FLECHT-SEASET test and determined that the proposed formulation produces a lower heat transfer coefficient. This is conservative relative to the previous formulation. Third, both formulations are supported with experimental data, consistent with the recommendations in Regulatory Positions 3.9.2 and 3.9.3 of RG 1.157.

3.1.5. Kinetic Droplet Model

AREVA calls this modification the "grid spacer droplet breakup heat transfer enhancement." The modeling change allows an adjustment to the vapor side interphase heat transfer coefficient downstream of a channel blockage (i.e., a spacer grid or a ruptured fuel element) by crediting the breakup of incident droplets, and the increase in vapor-liquid heat transfer surface area downstream of the obstruction. This modeling change is discussed in Section 7.5.4.10.1 and is qualified in Sections 8.2.3 and 8.4.1.

Empirical Basis

AREVA cites the BEACH Topical Report for the development of the droplet shattering model, and the theory closely aligns with the published work of Sugimoto and Murao. [26] The droplet shattering effects are quantified by calculating an atomization factor, or a ratio of droplet diameter downstream of the spacer to the upstream droplet diameter. This ratio is correlated to the average number of droplets produced by collision and the assembly blockage factor, i.e., the fraction of the channel obstructed by the spacer grid.

The relations describing the droplet breakup behavior were derived from tests using a rig consisting of a 6x6 heater rod bundle with capability to measure temperature and pressure at various axial locations. The test rig also had two viewing windows, one at the top and one at assembly mid-height. Droplet behavior was recorded at the mid-height window using motion picture; the central spacer at this location could be moved either above or below the window. Thus, the test enabled comparison of both heat transfer and droplet interactions above and below the central grid spacer. The grid spacer droplet shattering model was implemented in a reflood analysis computer code, and the test was simulated. Sugimoto and Murao showed that the implementation of the spacer grid model improved agreement with cladding temperature data just above the spacer grid. The published results also indicated that the modeling improvement did not result in an under-prediction of heater rod cladding temperature.

Chiou *et al.* implemented a model to account for similar phenomena in the BART code, and compared the results to an unmodified version of BART by simulating FLECHT-SEASET Tests 32333 and 42606. [27] Run 32333 was performed in the 161-rod bundle test rig with a stepped flow rate. The initial flow rate was 6 inches per second for 5 seconds, and it was followed by a 0.8 inches per second reflood rate. In comparison, Run 42606 was performed in the 21-rod flow blockage test array with a constant flooding rate of 0.9 inches per second. Thus, both of these tests include slower reflood rates that extend the amount of time portions of the test bundle remain in disperse flow conditions. Comparisons to both tests showed that the implementation of a grid spacer heat transfer model improved agreement to thermocouple data without under-prediction of the cladding temperature results.

The grid spacer droplet shattering model was derived from data that considered the spacer grid effect on droplet size distribution. The model, and models similar to it, have been checked against relevant data, including FLECHT-SEASET. Therefore, the NRC staff concluded that the model is adherent to the guidance in RG 1.157 and hence acceptable for implementation within S-RELAP5. Specifically, Regulatory Positions 3.9.2, "Experimental Data for Post-CHF Heat Transfer," and 3.9.3, "Post-CHF Heat Transfer from Uncovered Bundles," recommend that correlations and models developed for such heat transfer calculations be based on and checked against relevant data sets.

Implementation in Code

The implementation of the droplet shattering heat transfer enhancement is summarized on Pages 7-151 and 7-152 of EMF-2103P, Revision 3. **[**

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The implementation of the droplet shattering model is validated within the S-RELAP5 model assessment. In particular, the modified code was used to simulate low reflood rate FLECHT-SEASET tests with acceptable agreement to data. The code assessment is reviewed in further detail in Section 3.4.1 of this SE.

3.2. THERMAL-HYDRAULIC MODELS

3.2.1. Interfacial Drag Package

The changes to the S-RELAP5 interfacial drag package are summarized as follows:

> The interfacial drag package has been modified with improved logic for transition between flow regimes to cover a wider range of experimental data. This serves to update the state-of-the-art of S-RELAP5. The details of this alteration are presented in Section 7.5.2.

While Section 7.5.2 of EMF-2103P, Revision 3, provides a description of the S-RELAP5 interfacial drag package, it does not succinctly identify specific changes to the package. AREVA provided additional detail summarizing these changes in the response to RAI 34.

The response to RAI 34 indicated that AREVA modified the interfacial drag package based on AREVA's experience simulating broader sets of plant transients and benchmarks. The increased experience base indicated that "minor adjustments to the interphase drag and heat transfer models" had become necessary. The RAI response indicates that some of the changes are expected to have negligible impact on large break LOCA simulation, but are necessary to maintain and improve the broad applicability of S-RELAP5. However, AREVA also stated, in the RAI response, that some of the interfacial drag package changes were implemented to facilitate the more accurate calculation of disperse flow and film boiling heat transfer coefficients, thus enabling the use of DFFBHTC and FILMBL multipliers that are distributed about a value **[ ]** than in previous revisions of EMF-2103.

In the RAI response, AREVA goes on to describe a modification to junction interphase drag and three changes to the dispersed flow drag models. The dispersed flow drag model changes include: **[**

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Two additional changes – [ ] – were implemented to obtain more accurate results when simulating BWR steady-state operation and transient behavior, and to improve code continuity when simulating plant transients such as non-LOCA events. Since these changes affect modeling in S-RELAP5 applications other than for RLBLOCA, the NRC staff determined that these changes are outside the scope of the RLBLOCA review.

For each of the three in-scope changes, AREVA provided information describing the motivation for the change and a description of the methods used to ensure each change did not adversely affect the predictive capabilities of S-RELAP5. To summarize, each change was implemented to address previously identified issues with or limitations to the code's predictive capabilities, or to improve continuity. More specific details are provided in Table 3, below.

Change	Description	Motivation
[		
		]

Table 3. Interfacial Drag Package Changes.

The response to RAI 19, in particular Figure 2.19-1, illustrates the effect of the implementation of the new drag package in concert with the implementation of the Wong-Hochreiter heat transfer model. The illustration is a comparison of PCT versus test elevation for FLECHT-SEASET Test 31504, and the S-RELAP5 results are compared to Revision 0 of the model and the thermocouple data. The results show that, while these two modifications together effect a very slight reduction in PCT, the temperature still trends at or above the highest-temperature thermocouples for each elevation. Based on this comparison, the NRC staff determined that these changes to the interfacial drag package are acceptable.

3.3. FUEL MODELS

3.3.1. COPERNIC-Based Fuel Performance Model

The previously approved revision of EMF-2103(P)(A) obtained fuel steady-state fuel rod initial conditions from the RODEX3A fuel performance code.[28] AREVA has modified EMF-2103, Revision 3, to obtain its fuel performance inputs from the COPERNIC2 fuel rod design code, instead.[29] COPERNIC2 is a more modern fuel rod design code, which incorporates models that account for more recently identified phenomena, such as nuclear fuel TCD as a function of burnup. The code is also NRC-approved, and as such, the specific models and relations used by the code were not revisited in the present review.

The NRC staff reviewed the modeling guidelines discussed in Appendix A to EMF-2103, to verify that the guidelines will ensure that analysts develop S-RELAP5 fuel rod models that are consistent with the COPERNIC2 code with respect to fuel rod nodalization, gap performance, etc. The NRC staff determined that the implementation of COPERNIC2 within the RLBLOCA EM is acceptable since AREVA is obtaining fuel rod initial conditions from an NRC-approved fuel performance code and S-RELAP5 models will be developed consistently with the COPERNIC2 modeling,.

3.3.2. Fuel Rod Swelling, Rupture, and Relocation

Revision 3 to EMF-2103 incorporates models to calculate fuel rod cladding swelling and rupture behavior and to calculate the attendant effects of within-stack pellet relocation, **[**

]. While the swelling and rupture models are new to RLBLOCA, they are based on previously approved M5® cladding models,[30] which have been incorporated in both COPERNIC2 and Babcock and Wilcox Nuclear Technologies (BWNT) -LOCA.[31] The present review focuses mainly on adaptations to these models and the fuel relocation effects as implemented within RLBLOCA.

Modeling Overview

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Downstream Applications

The cladding strain and rupture, and fuel pellet relocation models are used by the EM in two noteworthy downstream calculations. **[**

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Fuel Clad Rupture Temperature

[ ]. The correlation is based on the work documented in NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis,"[32] and its development is described in detail in Appendices C and K of BAW-10227P-A, the M5® cladding properties TR. The correlation has been reviewed and approved by the NRC staff. That review is not repeated herein.

For application within RLBLOCA, once the rupture temperature is calculated, AREVA originally [ ]. In EMF-2103, Revision 3, AREVA stated [ ]. In response to RAI 27, AREVA further clarified that the variability is attributable to [ ].

In its review, the NRC staff considered the data supporting AREVA's correlation, which is tabulated in an erratum to BAW-10227P-A.[33] In figures K-5.11 and K-5.12 of BAW-10227P-A, AREVA [ ]. These figures indicate that there is significantly more than [ ] variation in the data set, particularly at [ ]. However, these data are plotted by [ ].

To review the data, the NRC staff separated the M5® data by ramp rate and evaluated whether the data fit within the [ ]. The NRC staff determined that, based on its study, the data for 10 °C/s and 15 °C/s ramp rates tended to fall significantly below the lower bound of rupture temperatures for [[G]].

An example of this study is illustrated in Figure 1, which shows the fast ramp rate (i.e., greater than or equal to (≥) 28 °C/s) data plotted with the upper and lower bounds of the rupture temperature curve; similar curves were generated to evaluate the correlation at all intermediate ramp rates for which EDGAR data were available.

[G] The NUREG-0630 authors note that "ramp-rate effects on the rupture temperature saturate at 28 °C/s" (NUREG-0630, Page 10). This was noted as an assumption in the Chapman correlation, but Figure 1 shows that the assumption appears to hold for the M5® EDGAR test data, [ ].

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Figure 1. [ ]

In addition to the bounds of a **[ ]**, Figure 1 also shows the bounds of a **[ ]**.

Similar trends are observable for the intermediate ramp rates. Although the application of such a **[ ]**. Therefore, the NRC staff determined that the proposed **[ ]** will be acceptable, provided a **[ ]** range is used. AREVA agreed, in response to RAI 27, to **[ ]**.

The approach is consistent with Regulatory Position 3.3.1 of RG 1.157, which indicates that the best-estimate methods to calculate the degree of fuel cladding rupture will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. Based on this consideration, the NRC staff determined that the approach to calculating the rupture temperature was acceptable.

Fuel Cladding Strain

Similar to the approach used for calculating the fuel cladding rupture temperature, AREVA uses a family of fuel cladding strain curves that are based on those found in NUREG-0630, but re-fit to data specific for M5®. The development of these curves is discussed in Appendices C and K of BAW-10227P-A. Since that TR is NRC-approved, the curve development review is not repeated in this SE.

To adapt the fuel cladding strain model for use in RLBLOCA, AREVA calculates cladding strain in one of three different ways depending on the **[** **]**.

The first two are adopted within BAW-10166P-A, "BEACH: Best-Estimate Analysis Core Heat Transfer, a Computer Program for Reflood Heat Transfer During LOCA."[34] **[** **]**. These models have been implemented acceptably in BEACH, which is a component of the BWNT-LOCA EM. Since they have been previously approved for use by the NRC staff, they are considered acceptable in this application and the previous review was not repeated; however, additional detail regarding the model is provided in response to RAIs 29 and 30.

A new adaptation for cladding burst strain is introduced within EMF-2103, Revision 3. Similar to **[** **]**. However, the actual burst strain is **[** **]**. The burst strain correlations are similar to those presented in NUREG-0630 and are intended for use in Appendix K EMs. Thus, **[** **]**. AREVA provided additional information regarding the validity of this approach in response to RAI 28. The information indicated that the **[** **]**. Additionally, the NRC staff reviewed the M5® database contained in Appendices C and K of BAW-10227P-A and confirmed that the cladding strain data are well represented by **[** **]**. Furthermore, AREVA pointed out in the RAI 28 response that the **[** **]** is supported by CSAU (NUREG/CR-5249). Because the approach for calculating burst strain is supported by available data, consistent with Regulatory Position 3.3.1 of RG 1.157, the NRC staff determined that the approach is acceptable for implementation within RLBLOCA.

Fuel Pellet Relocation

As nuclear fuel burns, the fuel pellet becomes fragmented. If higher-burnup fuel resides within a rod that experiences swelling and rupture, the fragmented fuel pellet within the stack can relocate to the ruptured region. Given a high enough packing fraction of relocated fuel, this behavior can result in an increased heat load to the cladding in the ruptured region. AREVA has developed an **[** **]**.

[**]**.

AREVA's response to RAI 15 provides further detail regarding the method used to determine the **[** **]**. The response to RAI 15A provides additional detail regarding the determination of the **[** **]**. The response to RAI 15B justifies AREVA's selected data set with regard to burnup. It is the subject of further evaluation described in the following paragraph. The responses to RAI 15C and RAI 15D provide additional justification regarding the distinction between **[** **]**.

The response to RAI 15B justifies AREVA's selected data set with regard to burnup. The NRC staff required this information because available data show that at higher burnups fuel pellets degrade into fine fragments, which would lead to relocation with a higher packing fraction. The response to RAI 15B clarifies that, in the currently available pellet relocation database, fragmentation fine enough to support packing factors beyond the limits of AREVA's proposed correlation only occurs at fuel burnup values beyond current licensing limits. While the NRC staff agrees with AREVA's assessment, the response indicates that two limitations are necessary. These are: (1) in view of the fact that licensing limits for fuel burnup are applied on a fuel-specific basis and subject to change with NRC staff approval, the fuel pellet packing factor model is valid only to fuel rod average burnup values of **[** **]**, as noted in the response to RAI 15B, and (2) should new data become available to suggest fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data. These are Limitations 5, "Burnup Limitation," and 6, "Pellet Relocation Packing Factor Data Set," in Chapter 4 of this SE.

In EMF-2103, Revision 3, as supplemented by response to RAI 15, AREVA has shown that the **[** **]** performs consistent with available data regarding fuel pellet relocation and has adopted a **[** **]**. The NRC staff determined that the proposed approach regarding fuel pellet relocation is acceptable, subject to the limitations discussed above and in Chapter 4 of this SE.

3.3.3. <u>Local Cladding Oxidation</u>

EMF-2103, Revision 3, includes a modified approach to calculate local cladding oxidation for comparison to the acceptance criterion set forth in 10 CFR 50.46(b)(2). The **[**

].[H] [

] calculated using C-P for the transient. The C-P correlation is considered acceptable best-estimate calculations of metal-water reaction, as documented in Regulatory Position 3.2.5.1 of RG 1.157. In the sub-sections below, two additional aspects are considered: (1) an appropriate acceptance criterion for cladding oxidation calculated using C-P and (2) the analytic treatment for pre-transient cladding oxidation.

<u>Use and Implementation of Cathcart-Pawel Correlation Relative to 10 CFR 50.46(b)(2)</u>

The requirements in 10 CFR 50.46(b) impose a limit on cladding oxidation of 0.17 times the total cladding thickness before oxidation. The oxidation limit is usually considered as a percentage. The oxidation limit can also be expressed as equivalent cladding reacted (ECR) (i.e., 17 percent ECR). The Atomic Energy Commission (AEC) deliberation over the 17 percent ECR acceptance criterion is discussed in detail in the 1973 Opinion of the Commission regarding acceptance criteria for ECCS for light-water-cooled nuclear power reactors (6 AEC 1085).

In its proceedings, the AEC noted that the "limits specified in these criteria will assure that some ductility would remain in the zircaloy cladding as it goes through the quenching process". The values were selected because experimental data indicated that cladding ductility is influenced not only by oxidation alone, but also by the temperature at which the oxidation occurs. The AEC received recommendations from fuel vendors, the AEC staff, and the public regarding the selection of an appropriate oxidation limit. The AEC's consideration included not only the total oxidation but also the thickness of brittle oxidation and zirconium layers in the cladding and the ratio of the thickness of the brittle layers to the remaining ductile layers. Noting wide agreement on the value of 17 percent ECR as a threshold above which cladding generally exhibited brittle behavior, the AEC settled on this value as the cladding oxidation limit.

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

Upon revision to 10 CFR 50.46 in 1988 to allow more realistic emergency core cooling performance calculations, the state of the art for cladding oxidation calculations had evolved. In addition to Baker-Just, Chapter 6.13 of NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," reviews C-P alongside two additional oxidation rate equations. The NUREG-1230, as well as RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling Performance," recommend the use of C-P based on its superior accuracy when compared to Baker-Just.

However, as noted in Research Information Letter (RIL) 02-02, Attachment 2, the original and confirmatory ring compression tests on which the 17 percent ECR criterion was based relied on

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an ECR value calculated using Baker-Just.[35] As noted on Page 9 of RIL 02-02, Attachment 2, "had the Cathcart-Pawel correlation – which did not exist at that time – been used, the cladding oxidation limit would have been about 13%. Therefore, the Baker-Just correlation must be used when comparing results with the old 17% limit."

The use of a 17 percent limit on ECR when applied to cladding oxidation values calculated using the C-P correlation does not provide the same level of assurance of cladding ductility as the same limit when applied to a result calculated using the Baker-Just correlation. In its present reviews of ECCS evaluation models, the NRC staff is imposing a limitation specifying that the ECR results calculated using the C-P correlation are considered acceptable in conformance with 10 CFR 50.46(b)(2) if the ECR value is less than 13 percent, which is equivalent to 17 percent ECR if calculated using the Baker-Just equation. This is Limitation 7, "13-Percent ECR: 1," as discussed in Chapter 4 of this SE.

Treatment of Pre-Transient Cladding Oxidation

AREVA began providing an **[**

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This treatment is necessary for reasons described in NRC Information Notice (IN) 1998-29, "Predicted Increase in Fuel Rod Cladding Oxidation."[36] As discussed in the IN:

> The acceptance criterion in 10 CFR 50.46(b)(2) requires that the calculated maximum total oxidation not exceed 0.17 times the total thickness of the cladding before oxidation. Total oxidation includes both pre-accident oxidation and oxidation occurring during a LOCA….
>
> Historically, the focus of compliance with 10 CFR 50.46 has been on 10 CFR 50.46(b)(1), "Peak Cladding Temperature," which is usually most limiting at the beginning of fuel life.[I] Because the oxidation rate is known to be dependent on temperature, total oxidation was also deemed most severe at the beginning of life (BOL). The contribution of pre-accident oxidation to the calculated total oxidation had not been previously thought to be significant, but as measured cladding oxidation thickness in the later stages of assembly service life increased faster than had been predicted, it became so.

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[I] As discussed in other sections of this SE, the assumption that PCT is usually most limiting at the beginning of life was shown to be unfounded subsequent to the issuance of IN 1998-29. Refer also to IN 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," and to IN 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Arising from Nuclear Fuel Thermal Conductivity Degradation" for additional discussion.

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Figure 2. [ ]

While the figure does not contain enough information to perform a rigorous analysis of the **[** **]**. In this case, however, AREVA proposed none.

To approximate the uncertainty **[** **]**, and further estimate a **[** **]**, the NRC staff performed a simple dimensional analysis. The analysis was performed to determine the effect of increasing the **[** **]**. The NRC staff evaluated the estimated percentage oxidation associated with the **[** **]** oxide layer thickness for three different PWR fuel cladding designs in use in the United States. In all three cases, a **[** **]** oxide layer thickness corresponded to approximately **[** **]** cladding thickness, compared to approximately **[** **]** associated with a **[** **]** thickness. [J]

[J] It is apparent that the study shows a roughly **[** **]** ratio for cladding thickness in microns to percentage cladding thickness, but the exact numbers are dependent on the actual cladding thickness, and there was some variability among the different fuel pin designs studied.

Despite that the **[**
], there are several reasons to consider this approach acceptable:

- The **[**
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- Evaluations performed to date have indicated that the cladding oxidation calculated for M5-clad fuel is very low. Several recent examples for once-burnt fuel, including pre-transient steady-state cladding corrosion, include:
    - 2.46 percent for a Combustion Engineering designed plant with a 14x14 fuel matrix[37]
    - 4.38 percent for a 3-loop Westinghouse Electric Company (Westinghouse) designed plant with a 17x17 fuel matrix[K, 38]
    - 3.38 percent for a Combustion Engineering plant with a 16x16 fuel matrix[39]

- The issue associated with pre-transient, steady-state cladding corrosion is more closely related to hydrogen pickup than to the growth of a brittle oxide layer, as is the case with transient cladding oxidation. When compared to Zr-4, M5® cladding has a significantly lower hydrogen pickup fraction per corrosion event.[40] The cladding material is therefore less susceptible to embrittlement, given the same corrosion layer thickness, when compared to Zr-4.

Based on these considerations, the NRC staff accepts the use of a **[**
]. One of the considerations relates to the total cladding oxidation associated with existing EM results. In plant-specific review activities, the NRC staff may request a more quantitative evaluation of **[** **]** if the estimated **[** **]**, as calculated using the C-P correlation. This review limitation, Limitation 8, "13-Percent ECR: 2," is discussed in Chapter 4 of this SE.

3.4. UNCERTAINTY EVALUATION

According to RG 1.157, "the purpose of the uncertainty evaluation is to provide assurance that for postulated loss-of-coolant accidents a given plant will not, with a probability of 95% or more, exceed the applicable limits specific in paragraph [10 CFR] 50.46(b)." The determination of total uncertainty within RLBLOCA includes uncertainty from important model parameters and plant state. This section of the SE addresses three topics: (1) the updated code assessment, (2) the treatment of plant-specific input or operating state uncertainty, and (3) the statistical combination of uncertainty.

[K] The results for this plant included ruptured fuel pins. If these results had been calculated in accordance with EMF-2103P, Revision 3, the calculated transient oxidation would be a different value, owing to an upgrade in the treatment of cladding oxidation for ruptured fuel pins.

3.4.1. Updated Code Assessment

Upon incorporating the changes described above, AREVA updated the S-RELAP5 code assessment commensurate with the added capabilities. Since the largest and most significant changes to the code, with respect to RLBLOCA, affect the predictive behavior associated with dispersed flow heat transfer, the assessment focuses on separate effects tests involving these phenomena. Principally, this includes FLECHT-SEASET tests, which are used to determine uncertainties and biases, and the ORNL THTF tests, which are used as confirmation. The NRC staff review of the assessment included consideration of the limitations associated with a one-dimensional (1-D) code like S-RELAP5 relative to the highly three-dimensional (3-D) core heat transfer phenomena that the code analyzes. At the request of the NRC staff, AREVA also provided a demonstration of the statistical combination of uncertainties by performing a statistical trial of the CCTF test to compare the statistical results to the test data. The updated code assessment shows that the predictive capabilities of S-RELAP5 have improved relative to EMF-2103(P)(A), Revision 0. The assessment also confirms that the EM remains acceptable regarding 10 CFR 50.46 requirements.

DFFB and FILMBL

The code assessment is used to determine the appropriate uncertainty treatment for highly ranked phenomena, including for dispersed flow film boiling and for inverted annular flow. AREVA used the same assessment methods described in EMF-2102, Revision 0, Section 5.1, but merely updated the assessment using the revised S-RELAP5 code. Based on the updated assessment, AREVA determined that new central values and distributions were appropriate for the FILMBL and DFFBHTC multipliers. The updated assessment, relative to Revision 2 of EMF-2103P, is discussed in Section 8.4.1 of EMF-2103P, Revision 3. The actual multipliers are presented in Section A.2.3.6.5 of EMF-2103P, Revision 3. The updated multipliers are compared to the Revision 2 multipliers in the response to RAI 35.

The updated assessment included the effects of implementation of the Wong-Hochreiter heat transfer correlation, the inclusion of the droplet shattering model, changes to the interfacial thermal hydraulic modeling, and upgrades to the radiation heat transfer modeling for the hot rod. As a result of the implementation of the upgrades associated with Revisions 2 and 3, AREVA **[ ]**. This is an expected change because the revised EM accounts more explicitly for the phenomena associated with disperse flow conditions.

Since AREVA repeated the uncertainty assessment for DFFBHTC and FILMBL using methods unchanged from those associated with EMF-2103(P)(A), Revision 0, the NRC staff determined that the assessment methods remain acceptable. The information contained in Section A.2.3.6.5 of EMF-2103, Revision 3, as supplemented by the response to RAI 35, demonstrates that the assessment was updated based on the Revision 3 EM. As noted in Section 8.4.1 of EMF-2103, Revision 3, the assessment is based on “low pressure industry-accepted and -evaluated experiments applicable to pressurized water reactor post-critical heat flux heat transfer.” Based on these considerations, the NRC staff determined that the revised heat transfer coefficient multipliers are acceptable.

TMINK

During its review, the NRC staff observed that AREVA had made changes to its treatment of the minimum temperature for stable film boiling, or TMINK. TMINK is an important parameter for determining the transition point at which droplet contact with heated walls is permitted. Above TMINK, the only means of heat transfer are convection to vapor and radiation from the heated surface. Below TMINK, droplet contact is permitted, and the heated wall benefits from conduction and convection to liquid droplets. If TMINK is set too high, the code will tend to predict lower peak cladding temperatures, increased steam cooling effects, and early quench times. All of these attributes can be non-conservative with respect to a PCT prediction.

In the response to RAI 16, AREVA clarified the following:

- From Revision 0 to Revision 3, TMINK was **[ ]**
- The uncertainty band was **[ ]**
- The S-RELAP5 assessments were performed using a TMINK value of **[ ]** (i.e., slightly **[ ]** than the production version of S-RELAP5)

Additionally, the code assessments provided in the response to RAIs 6 and 19 show that the code generally predicts delayed reflood times relative to FLECHT-SEASET benchmarks. Since these tests were simulated using a **[ ]**, yet the results still indicate a tendency for delayed reflood, the NRC staff determined that the revised TMINK value is acceptable.

Limitations on One-Dimensional Modeling in the Core

In its review, the NRC staff considered the adequacy of the 1-D hot channel fluid hydraulic model representation in the S-RELAP5 code. Since S-RELAP5, as well as many other industry thermal hydraulic (T/H) codes including TRAC-TF2, FLASH-4, RETRAN, and TRACE, contain a 1-D representation of the hot rod channel, the codes distribute vapor temperature and droplets evenly across the entire hot channel during the dispersed flow film boiling reflood phase of an RLBLOCA. The codes compute cross-section averaged quantities, which fail to provide an explicit representation of the very high temperature gradient in the vapor phase boundary layer near the wall.

The distribution of the evaporating water droplets plays a fundamental role in the heat transfer process. Noting the limitations above, the explicit distribution is not properly represented. In particular, interfacial heat transfer can be over-predicted, or otherwise not properly calculated with a 1-D model of the channel. This is a major limitation for all 1-D codes, and is also the subject of RAI 8.

Test data shows that the channel is 3-D with accumulation of drops in the central region and a highly superheated boundary layer region near the walls. Such variations in sub-channel behavior are observable, for example, in selected Achilles experiments.[41] Modeling this multi-dimensional behavior leads to a substantial reduction in the interfacial heat transfer and limiting of the droplet de-superheating to only the central core portion of the channel and not the highly superheated layer near the walls. S-RELAP5, like other T/H codes, suffers from this deficiency. There are no tractable model adjustments that can be made to the disperse flow film boiling (DFFB) model components to overcome this major discrepancy.

The sink temperature is not the entire average of the hot channel temperature for computing single phase heat transfer. Interfacial heat transfer between the drops and the vapor is controlled by the lower vapor temperature in the central core where the drops reside. Due to the simplified 1-D averaging of thermodynamic quantities in S-RELAP5 and the limited data in the literature, it is difficult to quantify all of the component contributions to DFFB. Of particular concern is that significant portions of the test data fail to measure the temperature distribution across the channel as well as the location of the drops in the channel. Without the knowledge of all of the individual component contributions to DFFB, it becomes very difficult to know and verify the magnitude of the droplet contribution in the S-RELAP5 model.

Without detailed knowledge of the magnitude of all of the components to DFFB, proper validation of this model against reflood data may result in including other phenomena and effects that are not pertinent to the heat transfer benefits from the droplet break up model, for example. Thus, there are limitations in applying a 1-D model to capture 3-D effects such as those described here and RAI 8. In its response to RAI 8, AREVA acknowledged the issue and stated that there is much uncertainty in the hot channel model because of these limitations.

The NRC staff notes that these uncertainties are accounted for through the use of multipliers on the heat transfer rate developed based on a comparison to a large range of reflood heat transfer data from the separate effects FLECHT tests. Thus, if the heat transfer is over-estimated, then the multipliers developed from the reflood data base will attempt to compensate for this deficiency. While the NRC staff recognizes that the use of a 1-D model of 3-D effects characteristic of DFFB is deficient as explained above, the treatment of this important heat transfer regime is adjusted through the use of multipliers and tuning of the other component model parameters to predict the bulk sink and clad surface temperatures.

The NRC staff also notes that there is also a lack of critical data (for example, radial sink temperature distribution in the hot channels of the separate effects reflood tests) to properly develop a DFFB model, and as such, this limits the ability to develop more exact DFFB models. Given these considerations, the NRC staff determined that the AREVA model for DFFB model and heat transfer multiplier uncertainty treatment provides an appropriate adjustment to compensate for the deficiencies in the 1-D model presentation. Comparisons to the FLECHT separate effects reflood tests demonstrate that the model captures and bounds the clad temperature response for a wide range of reflood conditions, including variations in pressure, reflood rate, power level, and axial shape characteristic of large break LOCAs. Based on these considerations, the NRC staff finds the DFFB model acceptable for application to RLBLOCA evaluations.

<u>CCTF Benchmark</u>

In response to RAI 7, AREVA exercised its statistical method used to determine combined uncertainty in an evaluation of Cylindrical Core Test Facility (CCTF) Test Run 62. The PCT results, provided in the figures accompanying the response to RAI 7, showed that a nominal value of the predicted PCT was in excellent agreement with the upper limit of the measured data. The upper limit of the predicted PCT was significantly higher than the upper limit of the measured data. In high power regions, the upper limit predicted PCT was **[** **]** measured values. In addition, and consistent with the various FLECHT-SEASET benchmarks, the CCTF comparison shows that S-RELAP5 consistently predicts delayed reflood times relative to the data.

These results provide a comparative illustration of the overall performance of the statistical methodology, relative to a large-scale, integral effects test. The simulation exercise also provides a code-to-test comparison separate from the FLECHT-SEASET benchmarks that are so pervasively used for evaluation throughout the remainder of this SE. The nominal result shows that the evaluation model, as a whole, retains conservatism in that the nominal result traces consistently with the upper range of test data. The upper limit results provide a qualitative sense of the margin added by the sampling approach.

3.4.2. Treatment of Plant-Specific Input or Operating State Uncertainty

Regarding parameters treated statistically, EMF-2103, Revision 3, states:

> The choice of distribution may be influenced by how a utility manages a given process parameter. For example, using a uniform distribution may properly reflect the control provided for a parameter if that control is random within a range. A uniform distribution is also considered a conservative approach in that equal likelihood is given for values at the limits of the distribution where the strongest influence is expected. However, if the [sic] there is an expectation that the true distribution is substantially non-uniform, the actual distribution can be used.

The modeling guidelines contained in Appendix A distinguish between model parameters and plant parameters. Model parameters have set uncertainty distributions. The uncertainty treatments for plant parameters are plant-specific. Since the uncertainty treatment of plant parameters is plant-specific, the NRC staff reviewed the approach in greater detail.

Regulatory Position 4 of RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," states, "In addition to the code uncertainty, various other sources of uncertainty are introduced when attempting to use best-estimate codes to predict full-scale plant thermal-hydraulic response. These sources include uncertainty associated with… the input boundary and initial conditions."

Regulatory Position 4.3.1 states, "Uncertainties associated with the boundary and initial conditions and the characterization and performance of equipment should be accounted for in the uncertainty evaluation."

Parameters controlled by facility Technical Specifications (TS) Limiting Conditions for Operation (LCOs) present a challenge with respect to the assignment of a distribution for a statistically treated input parameter. Administrative procedures may require controlling a specific parameter at a target value. Plant surveillance may produce sufficient data to reveal a true and potentially non-uniform distribution for that parameter. However, the TS LCO is required, pursuant to 10 CFR 50.36(c)(2)(ii)(B), to reflect the initial condition of the design basis event. The LCOs governing plant parameters used in the LBLOCA analysis do not typically prescribe a specific value and uncertainty range, but rather a bounding limit or an upper and a lower bound.

Other plant parameters, which are not controlled by facility TS, may be limited by the facility design basis in a manner potentially similar to an LCO. For example, a minimum and maximum RCS flow rate, as well as an uncertainty distribution, may already be applied in a statistical safety analysis.

In response to RAI 13, AREVA provided a general discussion of the interface between AREVA and utilities in identifying those elements of the facility design or licensing basis that are supported by the LBLOCA analyses and selecting the appropriate treatment for any particular plant parameter. AREVA also provided four categories into which plant inputs can be divided, and described each category. AREVA clarified that plant parameters covered by LCOs are considered in two of the possible four categories. Finally, AREVA provided additional detail concerning the appropriate application of both treatments.

The extent to which LBLOCA analysis supports a plant's licensing basis varies from plant to plant. AREVA indicated that utilities (i.e., licensees) are required to identify those limits of operation that necessitate support from a LBLOCA calculation through a formal design input transmittal. AREVA also stated that, while plant parameters are not limited to those identified in the TR, each licensing submittal will include a table, like Table B-8 of EMF-2103, Revision 3, which will be supplied for NRC review.

For the purposes of EMF-2103, Revision 3, AREVA divides plant inputs into four categories. These categories include **[**

].

AREVA stated, "Parameters defined by the TS Limiting Conditions for Operation… are considered in one of the first two categories (i.e., **[** **]**)."

Certain LCO-controlled parameters, such as the hot channel peaking factor ($F_{\Delta H}$), ECCS pumped injection flows, or the pumped ECCS temperatures, are treated **[** **]**. While such parameters may be controlled by TS LCOs, AREVA notes that the "values selected in practice… can be different [from the parameter limits] and are documented and justified in accordance with the goals of the calculation."

Ranged parameters are treated to "encompass the intended or allowed range of the parameter values during operation" and the "measurement and computational uncertainty of the controls provided to assure operation within the operational range." This parametric range is determined by the licensee, and typically corresponds to existing procedural controls or TS limits; however, a larger range may be selected if AREVA and licensee agree that the larger range is appropriate.

AREVA discusses several approaches for defining the probability distribution function for sampling the ranged parameters. One recommended approach is to rely on a **[** **]**. AREVA notes that other approaches may also be used, such as probability density functions derived from plant data, and also states that "**[** **]**."

AREVA concluded its response with the following summary paragraph:

> All operating ranges used in the analysis are supplied for review by the NRC in a table like Table B-8 of [EMF-2103(P), Revision 3]. The applicability of the analysis to support a plant's operating limits is the responsibility of the licensee. Changes by a licensee to the analyzed operating ranges or the assigned uncertainties, such as resulting from new instrumentation, are accommodated provided the sum of the intended range of operation and the uncertainties remains bounded by the limits of the distribution used in the analysis. Changes that cannot be accommodated within the applied range will require disposition by AREVA, a calculation of the expected impact, or a complete recalculation of the RLBLOCA analysis.

As discussed above, the applicable regulatory guidance, specifically Regulatory Position 4.3.1 of RG 1.157, recommends that the characterization and performance of equipment should be accounted for in the uncertainty evaluation. In addition, 10 CFR 50.36(c)(2)(ii)(B) requires that a TS LCO be established for "a process variable, design feature, or operating restriction that is an initial condition of a design basis accident…." AREVA's response provided a typical means by which plant-specific applications of the RLBLOCA methodology address these requirements, in that parametric ranges controlled by a TS LCO will be **[ ]** that meets or exceeds the LCO-controlled range. This approach tends to be conservative, since, as stated in the TR, "**[ ]**."

However, AREVA also noted that the applicability of the analysis to support a plant's operating limits is the responsibility of the licensee, and that adherence to the **[ ]** is not an EM requirement.

Since the treatment of LCO-controlled, or for that matter, facility design basis-controlled parameters, is a plant-specific approach determined by the licensee, the treatment must be reviewed on an application specific basis. Therefore, a table like Table B-8 of EMF-2103, Revision 3, must be provided for each request to implement EMF-2103, Revision 3. Requests will be considered acceptable if plant parameters are **[ ]** that meets or exceeds design basis or LCO limits, with uncertainties included, as appropriate. If plant parameters are treated differently, such as by sampling from a distribution derived from operational data, or by biasing in a manner that is not bounding of permissible facility operation, inclusive of uncertainties, additional justification is required and will be the subject of further NRC staff review. This is Limitation 9, "Uncertainty Treatment for Plant Parameters," as discussed in Chapter 4 of this SE.

Since the general approach recommended by AREVA ensures that characterization and performance of equipment is accounted for in the uncertainty evaluation, consistent with the guidance in RG 1.157, and with the requirements of 10 CFR 50.36, the NRC staff determined that the RAI response was acceptable. Plant-specific differences from the general approach will be addressed on an application-specific basis, Limitation 9, "Uncertainty Treatment for Plant Parameters," in Chapter 4 of this SE.

3.4.3. Statistical Combination of Uncertainty

This section of the SE addresses two topics. The first, "Treatment of Overall Calculational Uncertainty," addresses the method used to determine the upper tolerance limits in plant-specific analyses. This first section addresses the more familiar aspects of the statistical treatment of overall calculational uncertainty, as discussed in Regulatory Position 4.4 of RG 1.157. The second section, "Sample Size, Resampling, and Reanalysis," addresses relatively recent developments in the application of realistic ECCS evaluation models, which are (1) the extension of the methods to accommodate outlying results and (2) consideration of the processes used to address unacceptable results. Although the second section is not addressed by existing regulatory guidance, it reflects the fact that care must be taken when results exceed acceptance criteria, either because one or more outliers exist in the sample population, or because the analysis shows that the plant (as analyzed) does not conform to the NRC's acceptance criteria.

Treatment of Overall Calculational Uncertainty

Within an RLBLOCA analysis, uncertainty is estimated using non-parametric order statistics. The method relies on randomization to create a sample of cases defined by model and plant parameters for which individual biases and uncertainties are included in the total uncertainty evaluation. In other words, a set of cases is defined. Each case in the set has a randomly generated set of attributes for each statistically treated model or plant parameter. For example, the uncertainty in the film boiling heat transfer is treated by sampling the value for a multiplier within a range that is defined by the experimental assessment. The multiplier is used to increase or decrease the heat transfer coefficient. A similar approach is used to treat uncertainty in some plant parameters, such as accumulator cover pressure. When treated statistically, the plant parameters are generally sampled within design basis or TS allowable values.

The statistical evaluation in EMF-2103, Revision 3, has been modified to shift from a minimum sample size of 59 cases for univariate coverage. This prior approach was based on an assumption that the value of the 59th order statistic associated with the PCT (i.e., the highest-PCT result in the sample of 59 cases) could be used to show that the acceptance criteria for all three parameters could be met. This approach is an extension of the work of S. S. Wilks.[42]

The modification is intended to (1) provide high-confidence assurance for multivariate coverage and (2) further justify the use of larger sample sizes to reject the highest-ranked cases while maintaining acceptable statistical confidence. The present modification is based on the **[** **]**. AREVA revised Section 9.4 of EMF-2103, Revision 3, to provide a discussion of the application of the **[** **]** to the order statistics sampling method.

Using this method, **[**

][43] **[**

].

In an example provided by AREVA, a set of **[**
]. Based on this sample size, the **[**
]. Simultaneously, use of the **[**
].
A key feature of this example is the concept that each **[**
].

While the above discussion relates one example of the application of the **[** **]**, the general theory is discussed in Section 9.4.2 of EMF-2103, Revision 3. The theory can lead to the **[**

] This is limitation 10, "**[**
]" as discussed in Chapter 4 of this SE.

Other approaches to determining an appropriate sample size for setting joint tolerance limits, with specific regard to reactor safety analysis, have been published.[44, 45] These approaches, however, differ, and there is not universal agreement among the contributors.[46, 47, 48] In particular, Guba *et al.* (References 43 and 46) devise methods that return significantly higher sample sizes required for a given multivariate tolerance interval than do Nutt and Wallis (References 44 and 45). The NRC staff notes that, at the vendor's minimum sample size (**[**
]), the **[** **]**-derived upper tolerance limits are estimated by similar ranked order statistics as suggested by other methods published in open literature.

Although the specific approach described above is not explicitly discussed in RG 1.157, Regulatory Position 4.4 permits the use of tolerance intervals, and indicates that results at the 95 percent probability level are acceptable. With specific regard to the **[** **]** approach proposed by AREVA, the following additional considerations apply: (1) the approach is supported by a statistical proof, (2) it has been applied in other settings where joint tolerance limits are desired without regard for the correlation between or among specific figures or merit, and (3) application of the **[** **]** leads to sample sizes reasonably consistent with published literature. Based on these considerations, and subject to Limitation 10, "**[**
]," in the Chapter 4 of this SE, the NRC staff determined that the approach for determining the sample size to **[**
], are acceptable.

<u>Sample Size, Resampling, and Re-Analysis</u>

In its review, the NRC staff considered recent experience obtained with uses of order statistics-based methods. In previously approved evaluation models, the sample size would generally be accepted without significant constraints. In early applications of these models, the sample size applied in plant-specific evaluations corresponded to that discussed in the associated TR. However, and more recently, licensees have begun considering larger sample sizes.

According to the order statistics distribution-free tolerance intervals theory, the use of a larger sample size allows the desired tolerance in the results to be set by a lower order statistic. For example, when using a sample size of 59 cases for comparison to a single attribute, the top-ranked result provides a 95-percent probability result with 95-percent confidence. The use of a larger sample size – for example, 124 cases – allows for the same upper tolerance limit to be estimated by the third highest order statistic.

Application of this theory offers a benefit to the analyst in that use of larger sample sizes can help to accommodate the possibility of one or several outlying results that could be unnecessarily restrictive. Such flexibility is especially desirable for application to plants that have little margin to one or more of the 10 CFR 50.46(b)(1 – 3) acceptance criteria. Given advances in computing capabilities, this use of additional code runs can be an economical means to manage the risk of outlying results that either do not meet NRC acceptance criteria, or that unnecessarily constrain plant operation.

Broader adoption of the practice of using a larger sample size, however, requires due consideration of the overall sampling approach. As discussed in Section 24.11, "What's Wrong with this Picture," of NUREG-1475, Revision 1, Applying Statistics, increasing a sample size *a posteriori* reduces the confidence in the overall result. In other words, to maintain a certain upper tolerance limit based on binomial sampling, the sample size must be pre-determined. The identification, and potential elimination, of outliers after completion of the analysis requires a different treatment than merely adding additional cases to the sample.

AREVA's response to RAIs 22 and 23 address the issue described above. The following practices will be employed in determining a sample size and addressing unacceptable results:

- The sample size must be determined before the initiation of production safety analysis.

- If unacceptable results are obtained, changes to the plant input must be made to accommodate the unacceptable results. For example, the peak linear heat rate may require reduction.

- Generally, subsequent analyses should be repeated using the same sample (i.e., set of random numbers).

- Re-sampling should only be applied when significant changes to the plant are made. For example, the implementation of an extended power uprate may provide an appropriate circumstance to re-sample.

- In the event that unacceptable results are obtained, licensees submitting applications to the NRC staff must document the process used to disposition the unacceptable results as a demonstration that this process has been acceptably followed.

The above outlined approach will ensure that licensees applying RLBLOCA select the sample size prior to gaining inference regarding the characteristics of the sample population. The approach also ensures that the sample size is not modified *a posteriori*. Finally, the approach ensures that licensees communicate, in requests for licensing action, any approaches used to disposition unacceptable results. These practices ensure that the fidelity of the chosen tolerance level is preserved in the analysis, and based on this consideration, the NRC staff determined that the general approach for addressing outlying or unacceptable results is acceptable. Adherence to these practices is a required element of the methodology. The first four bullets listed above are required as per AREVA's response to RAIs 22 and 23; the final bullet is addressed by Limitation 11, "Re-Analysis," discussed in Chapter 4 of this SE.

3.5. ADDITIONAL MODELS AND FEATURES

3.5.1. Core Nodalization Shift

Relative to EMF-2103(P)(A), Revision 0, AREVA shifted the axial alignment of the core component node boundaries. Previously, the vertical node boundary had been located at grid spacer centerlines. In EMF-2103P, Revision 3, the node boundary is now located at the bottom of grid spacers. This change supports the implementation of the kinetic droplet model.

The nodalization shift is not explicitly evaluated at any point in the TR; however, the nodalization shift was employed in the update to the benchmarking. In response to RAI 19, AREVA also provided a series of analyses of the FLECHT-SEASET low reflood rate Test 31504, one of which evaluated the implementation of the revised core nodalization. Figure 2.19-3 of the RAI response shows that the effect of the nodalization shift slightly reduces the predicted PCT in the steam-cooled, upper regions of FLECHT test 31504, when compared to the previous nodalization. Even so, the code predictions remain above the thermocouple data. Further, the CCTF benchmark discussed in Section 3.4.1 of this SE was also performed using the core nodalization shift.

Successful completion of the benchmarks, as described above, using the revised core nodalization, demonstrates that the revised nodalization continues to perform acceptably. Based on this consideration, the NRC staff determined that the core nodalization shift is acceptable for implementation within RLBLOCA.

3.5.2. General Design Criterion 35 Compliance

According to EMF-2103(P)(A), Revision 0, offsite power availability was sampled as a binary plant parameter with equal probability assigned to onsite and offsite power supply. In plant-specific review activities subsequent to the NRC approval of EMF-2103(P)(A), Revision 0, the NRC staff determined that this approach does not provide the requisite assurance, pursuant to GDC 35, that:

> …for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the [emergency core cooling] system safety function can be accomplished, assuming a single failure.

The original approach attempted to treat power source in similar fashion to a sampled plant parameter. In accordance with RLBLOCA, sampled plant parameters are governed by a defined range. Typically, the parameters are **[** **]**. Since GDC 35 requires consideration of both on-site and off-site power availability, each assuming the other is unavailable, the distribution is binary (i.e., one or the other) and either possibility would be treated with equal probability.

In the absence of a known statistical distribution for the sampled parameter, the uniform distribution approach tends to introduce conservatism by increasing the frequency of the values sampled at the limits.[L] Practically, it is possible to verify that sampling a TS-controlled plant input parameter from a uniform, continuous distribution is acceptable by comparing the bounds of the sampling distribution to information like TS analytic limits or recent plant surveillance data.

As to the designation of a preferred power source following a LOCA, there is no such heuristic to validate the assumption. In addition, in response to RAI 20, AREVA provided the results of an analysis performed by analyzing a **[** **]**, the results were not so definitive as to permit a conclusion that **[** **]**, either for the reference plant analyzed, or generically.

There is, therefore, a distinction to be made between assumed power source and other statistically treated plant parameters or initial conditions. In its review, the NRC staff did not obtain sufficient information to confirm the adequacy of an assumed split fraction for the preferred power source, nor did the NRC staff establish a clearly conservative assumption to apply generically.

Thus, AREVA, in response to RAI 20, proposed to address GDC 35 compliance by **[** **]**.

This approach is acceptable to the NRC staff because it provides acceptably high assurance that ECCS performance meets the 10 CFR 50.46(b) acceptance criteria, given the availability of either on-site or off-site power, consistent with GDC 35 requirements. The NRC staff finding is based on two considerations: **[** **]**.

4.0 LIMITATIONS

This section of the SE summarizes the limitations[M] that apply to the NRC staff review of EMF-2103, Revision 3. Limitations 1 through 3 reflect the general applicability of the methodology in terms of the regulatory basis, plant designs, and fuel cladding properties considered in the NRC staff review. Limitation 4 is intended to ensure consistent and appropriate application of

[L] This behavior is characterized as a tendency, because a sampling range that relies on exceptionally large, unjustified limits can introduce unrealistic trends into the analysis that would be difficult to characterize as conservative.

[M] Although this SE has favored the term “limitations,” no distinction is made between “limitations” and “conditions” as used in Office of Nuclear Reactor Regulation Office Instruction LIC-500, “Topical Report Process.”

RLBLOCA on a plant-to-plant basis. The majority of the limitations applied to EMF-2103(P)(A), Revision 0, are reflected in the modeling guidelines referenced in Limitation 4. This limitation replaces the previous limitations, with the exception of the applicability limitations. The remaining limitations arose from the NRC staff review, and are discussed in specific sections of Chapter 3 of this SE.

4.1. ACCEPTANCE CRITERIA SATISFIED BY THIS EVALUATION MODEL

This EM was specifically reviewed in accordance with statements in EMF-2103, Revision 3. The NRC staff determined that the EM is acceptable for determining whether plant-specific results comply with the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3). AREVA did not request, and the NRC staff did not consider, whether this EM would be considered applicable if used to determine whether the requirements of 10 CFR 50.46(b)(4), regarding coolable geometry, or (b)(5), regarding long-term core cooling, are satisfied. Thus, this approval does not apply to the use of SRELAP5-based methods of evaluating the effects of grid deformation due to seismic or LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.

4.2. PLANT DESIGN APPLICABILITY

EMF-2103, Revision 3, approval is limited to application for 3-loop and 4-loop Westinghouse-designed nuclear steam supply systems (NSSSs), and to Combustion Engineering-designed NSSSs with cold leg ECCS injection, only. The NRC staff did not consider model applicability to other NSSS designs in its review.

4.3. FUEL CLADDING

The EM is approved based on models that are specific to AREVA proprietary M5® fuel cladding. The application of the model to other cladding types has not been reviewed.

4.4. MODELING GUIDELINES

Plant-specific applications will generally be considered acceptable if they follow the modeling guidelines contained in Appendix A to EMF 2103, Revision 3. Plant-specific licensing actions referencing EMF 2103, Revision 3, analyses should include a statement summarizing the extent to which the guidelines were followed, and justification for any departures.

Additional Discussion

Should NRC staff review determine that absolute adherence to the modeling guidelines is inappropriate for a specific plant, additional information may be requested using the RAI process. For example, if a specific plant shows heightened PCT sensitivity to containment parameters, the NRC staff may request additional information seeking justification for the application of the containment modeling guidelines to that particular plant.

4.5. BURNUP LIMITATION

The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from data that extend to currently licensed fuel burnup limits (i.e., rod average burnup of

[ ]). Thus, the approval of this method is limited to fuel burnup below this value. Extension beyond rod average burnup of [ ] would require a revision or supplement to EMF-2103, Revision 3, or plant-specific justification.

Reference

Refer to SE Section 3.3.2 for further detail.

4.6. PELLET RELOCATION PACKING FACTOR DATA SET

The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from currently available data. Should new data become available to suggest that fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data.

Such a request would be tendered by a letter from the NRC to AREVA identifying the newly available data and requesting an update to the model, or an assessment to demonstrate that such an update is not needed.

Reference

Refer to SE Section 3.3.2 for further detail.

4.7. 13-PERCENT C-P ECR: 1

The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. To account for the use of the C-P correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.

Additional Discussion

Should the NRC staff position regarding the application of the 17 percent Baker-Just acceptance criterion to the C-P correlation change, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.

Reference

Refer to SE Section 3.3.3 for further detail.

4.8. 13-PERCENT ECR: 2

In conjunction with Limitation 8 above, C-P oxidation results will be considered acceptable, provided plant-specific [ ]. If second-cycle fuel is identified in a plant-specific analysis, whose [ ], the NRC staff reviewing the plant-specific analysis may request technical justification or quantitative assessment, demonstrating that [ ].

Additional Discussion

This limitation ensures that the safety analysis retains sufficient margin to the ECR analytic limit to **[** **]**.

Reference

Refer to SE Section 3.3.3 for further detail.

4.9. UNCERTAINTY TREATMENT FOR PLANT PARAMETERS

The response to RAI 13 states that all operating ranges used in a plant-specific analysis are supplied for review by the NRC in a table like Table B-8 of EMF-2103, Revision 3. In plant-specific reviews, the uncertainty treatment for plant parameters will be considered acceptable if plant parameters are **[** **]**, as appropriate. Alternative approaches may be used, provided they are supported with appropriate justification.

Additional Discussion

This limitation ensures that the safety analysis adequately covers the range of permissible plant operation, as discussed in Section 3.4.2 of this SE. However, this limitation should not be construed to imply that exceeding limiting values by any amount is acceptable; sampling distributions for plant parameters should be realistic and justifiable.

Reference

Refer to SE Section 3.4.2 for further detail.

4.10. **[** **]**

[

]

Reference

Refer to SE Section 3.4.3 for further detail.

4.11. RE-ANALYSIS

Any plant submittal to the NRC using EMF-2103, Revision 3, which is not based on the first statistical calculation intended to be the analysis of record must state that a re-analysis has been performed and must identify the changes that were made to the evaluation model and/or input in order to obtain the results in the submitted analysis.

Additional Discussion

Adherence to this process ensures that the fidelity of the chosen tolerance level is preserved in the analysis.

Reference

Refer to SE Section 3.4.3 for further detail.

5.0 CONCLUSION

Based on the review described in the preceding sections, and subject to the limitations delineated in Section 4 of this SE, the NRC staff determined that EMF-2103, Revision 3, is acceptable for referencing in licensing actions. As documented in EMF-2103, Revision 3, RLBLOCA is an acceptable evaluation model for the purpose of compliance with 10 CFR 50.46 requirements. EMF-2103, Revision 3, is also approved for use.

EMF-2103, Revision 3, constitutes a separate and unique methodology. Any other version derived from this TR, such as designated by a new revision number, amendment number, addendum number or equivalent designation, would constitute a definition of a new methodology requiring NRC review and acceptance prior to generic application and prior to any specific plant licensing application of a new methodology derived from EMF-2103, Revision 3.

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Attachment: Resolution of Comments

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