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Adaptation of the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology to Perform Analysis of Cladding Rupture for High Burnup Fuel

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Revision 0

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Jeffrey R. Kobelak*	Safety Analysis
Kevin J. Barber*	LOCA Integrated Services
Andrew Bowman*	Licensing
Aaron M. Everhard*	LOCA Integrated Services
Brian P. Ising*	LOCA Integrated Services

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Prepared by: Jeffrey R. Kobelak**
Safety Analysis

Reviewer: Kevin J. Barber**
LOCA Integrated Services

Approved: Amy J. Colussy**, Manager
LOCA Integrated Services

*Contributor to content described in this topical report; formal signatures for topical report indicated by preparer, reviewer, and approver.

**Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

EXECUTIVE SUMMARY

The Westinghouse **FULL SPECTRUM™** LOCA (**FSLOCA™**) evaluation model (EM) was licensed for the purpose of allowing licensees to demonstrate compliance with the emergency core cooling system (ECCS) acceptance criteria prescribed in 10 CFR 50.46. The **FSLOCA** EM is a best-estimate LOCA (BELOCA) methodology, that was licensed to calculate the LOCA transient response across the full spectrum of break sizes (small-break LOCA, intermediate-break LOCA, and large-break LOCA). Since the **FSLOCA** EM was licensed, dozens of operating units have incorporated the methodology into their licensing bases.

Since the approval of the **FSLOCA** EM, Westinghouse has been working on the extension of the **FSLOCA** EM to higher burnup and higher initial fuel rod enrichments to support industry aspirations. That extension includes addressing new phenomena associated with high burnup fuel rod response during a postulated LOCA, such as the potential for fuel fragmentation, relocation, and dispersal (FFRD). While the licensed **FSLOCA** EM does account for fuel fragmentation and relocation, it does not account for fuel dispersal.

One means to address potential consequences of fuel dispersal is to prevent cladding rupture in fuel rods which are susceptible to fine fragmentation during a postulated LOCA. If the cladding does not rupture, then fine fuel fragments cannot be dispersed from the fuel rods. This topical report provides a method for application of the **FSLOCA** EM framework to the prediction of cladding rupture during a postulated LOCA. Rather than demonstrating compliance with the ECCS acceptance criteria, the **[**

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The cladding rupture calculations are **[** **]**^{a,c} similar to how the analyses are performed as part of the **FSLOCA** EM to demonstrate compliance with 10 CFR 50.46. Since explicit analysis of the intermediate-break region may be desired, the full break spectrum is divided into 3 regions; specifically Region I, Region IB (the new intermediate-break region defined within this topical report), and Region II. Following the methodology in this topical report, the cladding rupture calculations could be performed for **[**

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The thermal-hydraulic code used for the cladding rupture calculation methodology is the WCOBRA/TRAC-TF2 code, which was licensed as part of the **FSLOCA** EM for application to the full spectrum of LOCA break sizes. An updated version of that code, which is applicable for the modeling of higher burnup fuel and fuel rods with higher initial enrichments, is utilized as described within this topical report. The updated code is also capable of modeling the important phenomena associated with high burnup fuel rod response (such as transient fission gas release). The applicability of this topical report covers all Westinghouse and Combustion Engineering fuel designs, standard UO_2 and **ADOPT™** fuel pellets, and **AXIOM®** cladding.

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ACRONYMS and NOMENCLATURE

ANS/ANSI	American Nuclear Society / American National Standards Institute
ASTRUM	Automated Statistical Treatment of Uncertainty Method
BELOCA	best-estimate loss-of-coolant accident
BWR	boiling water reactor
CCFL	counter-current flow limitation
CDF	cumulative distribution function
CE	Combustion Engineering
CGE	V. C. Summer
CHF	critical heat flux
CFR	Code of Federal Regulations
CWO	core-wide oxidation
DEG	double-ended guillotine
DLW	Beaver Valley Unit 1
ECCS	emergency core cooling system
ECR	equivalent cladding reacted
EM	evaluation model
EMDAP	evaluation model development and assessment process
FFRD	fuel fragmentation, relocation, and dispersal
FGR	fission gas release
FSLOCA	**FULL SPECTRUM** LOCA
GEDM	Generalized Energy Deposition Model
H/U	hydrogen / uranium ratio
HFP	hot full power
IBLOCA	intermediate-break loss-of-coolant accident
IFBA	integral fuel burnable absorber
L&C	limitation and condition
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
MLO	maximum local oxidation
MSSV	main steam safety valve
MTC	moderator temperature coefficient
NFI	Nuclear Fuel Industries
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OFA	optimized fuel assembly
OPA	offsite power available
ORNL	Oak Ridge National Laboratory
PACC	accumulator cover pressure
PAD	Performance Analysis and Design
PBF	Power Burst Facility
PCT	peak cladding temperature
PIE	post irradiation exam

ACRONYMS and NOMENCLATURE (continued)

PIRT	phenomenon identification and ranking table
PRCS	reactor coolant system pressure
PWR	pressurized water reactor
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RIL	research information letter
ROSA	rig-of-safety assessment
SATS	severe accident test station
SBLOCA	small-break loss-of-coolant accident
SCIP	Studsvik Cladding Integrity Program
SER	safety evaluation report
SET	separate effects test
SG	steam generator
SGTP	steam generator tube plugging
SI	safety injection
TACC	accumulator temperature
TAVG	vessel average temperature
TCHF	critical heat flux temperature
Tcold	upper head liquid temperature near vessel inlet temperature
TDF	thermal design flow
tFGR	transient fission gas release
Thot	upper head liquid temperature near vessel outlet temperature
THTF	thermal-hydraulic test facility
TMIN	minimum film boiling temperature
TSI	safety injection temperature
UCP	upper core plate
UPI	upper plenum injection
VACC	accumulator water volume
WABA	wet annular burnable absorber

1 OVERVIEW AND METHODOLOGY ROADMAP

1.1 BACKGROUND

The Westinghouse **FULL SPECTRUM™** loss-of-coolant accident (**FSLOCA™**) evaluation model (EM) was licensed for the purpose of allowing licensees to demonstrate compliance with the emergency core cooling system (ECCS) acceptance criteria prescribed in Title 10 of the Code of Federal Regulations (CFR) Part 50.46. The **FSLOCA** EM is a best-estimate LOCA (BELOCA) methodology, that was licensed to calculate the LOCA transient response across the full spectrum of break sizes (small-break LOCA (SBLOCA), intermediate-break LOCA (IBLOCA), and large-break LOCA (LBLOCA)). Since the **FSLOCA** EM was licensed, dozens of operating units have incorporated the methodology into their licensing bases.

Since the approval of the **FSLOCA** EM, Westinghouse has been working on the extension of the **FSLOCA** EM to higher burnup and higher initial fuel rod enrichments to support industry aspirations. That extension includes addressing new phenomena associated with high burnup fuel rod response during a postulated LOCA, such as the potential for fuel fragmentation, relocation, and dispersal (FFRD). While the licensed **FSLOCA** EM does account for fuel fragmentation and relocation, it does not account for fuel dispersal.

One means to address potential consequences of fuel dispersal is to prevent cladding rupture in fuel rods which are susceptible to fine fragmentation during a postulated LOCA. If the cladding does not rupture, then fine fuel fragments cannot be dispersed from the fuel rods. This topical report provides a method for application of the **FSLOCA** EM framework to the prediction of cladding rupture during a postulated LOCA. Rather than demonstrating compliance with the ECCS acceptance criteria, the **[**

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The **FSLOCA** EM was developed following the Evaluation Model Development and Assessment Process (EMDAP) as described in (Kobelak et al., 2016). A similar, but abbreviated process is utilized in this topical report to develop a LOCA method for the prediction of cladding failure in high burnup fuel since the majority of the methodology in (Kobelak et al., 2016) remains applicable. The changes to the WCOBRA/TRAC-TF2 code (the thermal-hydraulic code licensed as part of the **FSLOCA** EM) described within this topical report are primarily focused on the decay heat and kinetics models relative to higher burnup fuel and the fuel rod models related to fission gas release, cladding deformation, pre-burst fuel relocation, and cladding rupture. The method described in this topical report is based on the **FSLOCA** EM framework and utilizes a modified version of the same thermal-hydraulic code.

Changes were implemented within the WCOBRA/TRAC-TF2 code to extend the applicability of the methodology described in this topical report beyond the allowable limits of the **FSLOCA** EM. This topical report provides justification for application of the fuel rod cladding rupture calculations to fuel rods with up to **[ ]**^{a,c} w/o initial enrichment, and up to **[ ]**^{a,c} GWd/MTU fuel rod average burnup.

The WCOBRA/TRAC-TF2 code models and the methodology described in this topical report apply to **AXIOM®** cladding, and both standard UO_2 and **ADOPT™** fuel pellets. The applicability includes un-poisoned fuel, fuel with integral fuel burnable absorber (IFBA), and fuel with Gadolinia. The cladding rupture methodology is applicable to 2-Loop Westinghouse pressurized water reactors (PWRs) equipped with upper plenum injection (UPI), 3-Loop Westinghouse PWRs with cold side injection, 4-Loop Westinghouse PWRs with cold side injection, and Combustion Engineering (CE) designed PWRs. A limitation is imposed on this topical report related to the applicable plant classes as discussed in Section 7.2.

Note that the term “higher burnup fuel” is used throughout this topical report. That term describes fuel rods with sufficient burnup to experience fine fragmentation and potential dispersal during a postulated LOCA. Specific discussion of the associated burnup threshold for fine fragmentation is presented in Section 3.5 herein.

1.2 ORGANIZATION OF THE REPORT

Section 1, herein, provides a roadmap of the topical report. The background was discussed in Section 1.1. This section describes the organization and content of the topical report. Section 1.3 maps the content of the topical report to available regulatory guidance and other relevant industry publications. Section 1.4 provides a list of approved topical reports and methods with burnup limits which are relevant to the methodology developed within this topical report.

Section 2 contains a discussion of the pertinent phenomena identification and ranking, with a **[**

]^{a,c} The updates to the fuel rod models and the kinetics and decay heat models are discussed in Sections 3 and 4, respectively.

The method for performing the cladding rupture calculations is discussed in Section 5. Some IBLOCA sensitivity studies, as well as a demonstration analysis using the method described in Section 5 is presented in Section 6.

Section 7 provides a summary of the information contained in the topical report, including the limitations on application of this topical report.

1.3 MAPPING OF TOPICAL REPORT TO REGULATORY GUIDANCE

There is no specific language with Parts 50, 52, and 100 to Title 10 of the Code of Federal Regulations (10 CFR) that imposes a maximum fuel rod burnup limit. A review of Title 10 was performed to substantiate this conclusion, with particular emphasis on 10 CFR 50.46, 10 CFR 50.49, 10 CFR 50.67, 10 CFR 100, 10 CFR 50 Appendix A General Design Criteria 10, 27, 28, and 35, and 10 CFR 50 Appendix K. Furthermore, there is also no specific part of 10 CFR that imposes a maximum fuel enrichment limit relative to LOCA analysis methods.

10 CFR 50.68 prescribes criticality accident requirements and limits the maximum nominal ^{235}U enrichment of fresh fuel assemblies to 5 wt%. This requirement, however, pertains to handling and storage of fuel and is outside the scope of this topical report. Similarly, 10 CFR 70.24, which provides criteria, separate from 10 CFR 50.68, for preventing criticality accidents, is outside the scope of this topical report.

There are several regulatory and industry documents which can inform the scope of this topical report, especially as it relates to the extension of the WCOBRA/TRAC-TF2 code to higher initial fuel rod enrichments which are operated to higher fuel rod average burnup. The documents considered within this topical report are discussed in this section.

1.3.1 Industry Phenomenon Identification and Ranking Tables

In the early 2000s, a phenomenon identification and ranking table (PIRT) was developed for loss-of-coolant accidents (LOCAs) in PWRs containing high burnup fuel. The resulting PIRT was issued in (Boyack et al., 2001), the key points of which are summarized in (Meyer, 2001). The key findings from these documents are considered in reviewing the prior Westinghouse LOCA PIRT as discussed in Section 2 of this topical report.

1.3.2 Research Information Letter Regarding FFRD

More recently, the Nuclear Regulatory Commission (NRC) issued Research Information Letter (RIL) 2021-13 (Bales et al., 2021), which contains a conservative interpretation of available research at the time regarding fuel fragmentation, relocation, and dispersal in high burnup fuel rods. The RIL provides conservative interpretations regarding thresholds for susceptibility to fine fragmentation, cladding strains at which fuel becomes mobile within high burnup fuel rods, the mass of fuel fragments which could potentially be dispersed into the coolant, evidence regarding transient fission gas release (tFGR), and a characterization of fuel packing fractions in the balloon region of a high burnup fuel rod. Some of these elements are only applicable after fuel rod rupture occurs and are therefore beyond the scope of this topical report. The other elements are addressed within Section 3 of this topical report.

1.4 SUPERSEDED LIMITATIONS AND CONDITIONS

This topical report does not supersede any existing limitations and conditions. Specifically, the limitations and conditions imposed on the **FSLOCA** EM for demonstration of compliance with the ECCS acceptance criteria are not affected by this topical report. However, the method described herein is applicable to a broader range of conditions than the NRC-approved **FSLOCA** EM. Therefore, the use of the **FSLOCA** EM as the basis for this rupture methodology necessitates reinterpretation of some of the limitations and conditions in (Kobelak, et al., 2016) for applicability to this topical report. A review of the limitations and conditions as they apply to this topical report is provided in Section 7.1.

Limitation and Condition #5 in Sections 4.6.3.1 and 5.0 of the NRC safety evaluation report (SER) included in (Kobelak et al., 2016) indicates that the maximum assembly average burnup is limited to **[ ]**^{a,c} and the maximum rod length-average burnup is limited to **[ ]**^{a,c}. The primary reason for the burnup limitation is related to the assessment of the decay heat model in the WCOBRA/TRAC-TF2 code (although it is acknowledged that the PAD5 fuel performance code used to initialize the fuel rods in WCOBRA/TRAC-TF2 was also limited to the same burnup level). The decay heat model is updated within this topical report as discussed in Section 4 herein. Updated limitations and conditions on this topical report, including the fuel rod performance data utilized in the calculations, the maximum fuel rod burnup, and the initial enrichment, are discussed in Section 7.2.

1.5 REFERENCES

1. Bales, M., et al., December 2021, "Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup," RIL 2021-13.

2. Boyack, B. E., et al., December 2001, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel," NUREG/CR-6744.

3. Kobelak, J. R., et al., November 2016, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1 (NRC Safety Evaluation dated September 12, 2017, ADAMS Accession Number (package) ML17207A124).

4. Meyer, R. O., September 2001, "Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel," NUREG-1749.

2 PHENOMENA IDENTIFICATION AND RANKING

The PIRTs from the **FSLOCA** EM as well as an industry PIRT for postulated LOCAs in PWRs with high burnup fuel were reviewed in Section 4.3 of (Kobelak et al., 2020) with the focus of identifying phenomena where the existing code models and/or modeling approach need to be considered for application to cladding rupture calculations for high burnup fuel. The discussion in this section is based on Section 4.3 of (Kobelak et al., 2020), modified as appropriate to reflect the content of this topical report. The **FSLOCA** EM PIRT is reviewed in Section 2.1, and the industry PIRT is reviewed in Section 2.2.

The LOCA scenario and transient class considered are the same as described in Section 1.2.1 of (Kobelak et al., 2016). The power plant class includes all Westinghouse-designed 2-loop PWRs equipped with UPI, 3-loop and 4-loop plants with ECCS injection into the cold legs, and Combustion Engineering designs. It is noted that the approval of the **FSLOCA** EM is presently limited to Westinghouse-designed 3-loop and 4-loop PWRs. As such, application of this method to Westinghouse-designed 2-loop PWRs with UPI and CE-designed PWRs requires that the licensing basis LOCA analyses utilize the **FSLOCA** EM as extended to those plant classes (see Limitation #1 in Section 7.2).

The updates to the code and method described in this topical report are primarily focused on phenomena related to the reactor core and the fuel rods which reside in the core region. Within previously licensed Westinghouse BELOCA methodologies, there are **[** **]**^{a,c} as evidenced by the approval of the prior Automated Statistical Treatment of Uncertainty Method (ASTRUM) (Nissley et al., 2005) for all of these plant classes. Therefore, the updates in this topical report are expected to be applicable across all the different plant classes identified. As further assurance of the applicability of this method across the different plant classes, Limitation #1 in Section 7.2 includes a requirement to reconcile any changes in the approved **FSLOCA** EM against the changes in this topical report for 2-loop Westinghouse-designed PWRs with UPI and CE-designed PWRs.

2.1 FULL SPECTRUM LOCA METHODOLOGY PIRT REVIEW

The **FSLOCA** EM phenomena identification and ranking for LOCA analysis is discussed in Section 2.3 of (Kobelak et al., 2016). The phenomena of importance and rankings for the LOCA calculation to address the 10 CFR 50.46 acceptance criteria generally apply for the cladding rupture calculation, since a higher predicted cladding temperature increases the likelihood of burst. However, some phenomena such as cladding rupture may be of more direct importance.

Since the primary impact of the cladding rupture calculation methodology developed in this topical report is related to the analysis of higher burnup fuel rods, the fuel rod and core phenomena discussed in Sections 2.3.2.1 and 2.3.2.2 of (Kobelak et al., 2016) are reviewed. The focus of the review is to identify phenomena where the current code models and/or modeling approach should be reviewed for adequacy relative to cladding rupture calculations for higher burnup fuel.

2.1.1 Fuel Rod

Stored Energy

For WCOBRA/TRAC-TF2, within the **FSLOCA** EM, the **[**

]^{a,c} to fuel performance data from PAD5 (Bowman et al., 2017) as described in Section 29.4.2.2 of (Kobelak et al., 2016).

The stored energy of the fuel is important for the method to calculate cladding rupture that is developed in this topical report. An assessment of the gap conductance model in WCOBRA/TRAC-TF2 is discussed in Section 3.1 herein. The approach for calibration of the fuel temperatures (stored energy) is discussed in Section 3.4.

Clad Oxidation

Cladding oxidation is considered highly important in the prediction of a postulated LOCA transient response. There are various ways in which cladding oxidation influences the LOCA transient response, from the effects during normal operation on the fuel rod initialization through the LOCA transient. Each of the various influences is discussed.

Corrosion occurs during normal operation, which results in the development of an oxide layer on the outer surface of the fuel rods. This can influence the fuel rod condition at the onset of the LOCA transient (see Section 3.4). The corrosion process also results in hydrogen uptake into the cladding, which can impact cladding ductility. The approved corrosion model for **AXIOM** cladding is described in Section 5.1 of (Pan et al., 2023), and the hydrogen pickup model is discussed in Section 5.2 therein. Consideration of the 10 CFR 50.46 criteria with respect to **AXIOM** cladding is discussed in Section 6.2.1.4 of (Pan et al., 2023). The cladding embrittlement criterion accounts for the pre-existing cladding hydrogen content which is a result of the steady-state corrosion process. With respect to the cladding rupture calculation, the effect of corrosion and hydrogen is discussed in Section 3.3.1 of this topical report. **[**

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During a LOCA transient, an exothermic metal-water reaction can occur between the cladding and the surrounding coolant. The exothermic metal-water reaction becomes an increasingly significant source of heat addition with increasing cladding temperature. **[**

]^{a,c} Furthermore, it is noted in Section 5.1.2 of (Billone et al., 2008) that for irradiated fuel, the corrosion layer was found to be partially protective with regard to the growth of a high-temperature oxidation layer with time. **[**

]^{a,c}

Decay Heat

The W̲COBRA/TRAC-TF2 code decay heat model is based on the American National Standards Institute / American Nuclear Society (ANSI/ANS) 5.1-1979 standard. **FSLOCA** EM Limitation and Condition #5 limits maximum assembly average burnup to **[** **]^{a,c}** and the maximum peak rod length average burnup to **[** **]^{a,c}** because the physics parameters supporting the decay heat model in W̲COBRA/TRAC-TF2 were only assessed to **[** **]^{a,c}**. The physics parameters supporting the decay heat model in W̲COBRA/TRAC-TF2 were also only valid up to **[** **]^{a,c}** The decay heat is important to the rupture calculations as it is a significant energy source that influences the rate of cladding heatup. The physics parameters supporting the W̲COBRA/TRAC-TF2 decay heat model (as well as other aspects of the kinetics and decay heat calculations) are discussed in Section 4 for the methodology developed in this topical report.

Clad Deformation

The cladding deformation (and conditions for rupture) are highly important for the LOCA methodology in this topical report. Deformation of the cladding can influence the thermal-hydraulic conditions in the fuel bundles. It can also change the free volume and hence the pressure inside the fuel rod cladding. The conditions leading to rupture will directly influence whether or not the cladding is predicted to fail for a given LOCA transient. The cladding deformation and rupture models are discussed in Sections 3.2 and 3.3, respectively.

2.1.2 Core

Critical Heat Flux

W̲COBRA/TRAC-TF2 was assessed against experimental data in Sections 13, 15, and 22 of (Kobelak et al., 2016). It was found that the **[**

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Post-CHF Heat Transfer / Steam Cooling

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of post-CHF heat transfer / steam cooling from (Kobelak et al., 2016). **[**

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Rewet / T_{min}

As discussed in Section 29.1.8 of (Kobelak et al., 2016), the **[**

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]a,c

Heat Transfer to a Covered Core

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of heat transfer to a covered core from (Kobelak et al., 2016). Cladding rupture would not occur when the core is covered.

Radiation Heat Transfer

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of radiation heat transfer from (Kobelak et al., 2016).

3-D Flow / Core Natural Circulation

Multidimensional effects are captured by the core nodalization scheme, which uses separate assembly groupings to capture the radial flow distribution. The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of 3-D flow and core natural circulation from (Kobelak et al., 2016).

Void Generation / Void Distribution

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of void generation / void distribution from (Kobelak et al., 2016).

Entrainment / De-entrainment

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of entrainment / de-entrainment from (Kobelak et al., 2016).

Flow Reversal / Stagnation

Flow reversal and stagnation in the core is affected via the sampling of global models in the **FSLOCA** EM, such as the **[**

]a,c (illustrated for some parameters in Section 28 of (Kobelak et al., 2016)). The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of flow reversal / stagnation from (Kobelak et al., 2016).

Flow Resistance

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of flow resistance from (Kobelak et al., 2016). Key resistances in the WCOBRA/TRAC-TF2 model are calibrated as discussed in Section 26.4 of (Kobelak et al., 2016). The approach to address flow reversal and stagnation in the core, discussed in the prior paragraph herein, captures the effect of variation in flow resistance as well.

Water Storage in Barrel / Baffle Region

The analysis of cladding rupture for higher burnup fuel does not impact the PIRT rankings or treatment of water storage in the barrel / baffle region from (Kobelak et al., 2016).

2.2 INDUSTRY PIRT REVIEW

The review of the **FSLOCA** EM PIRT discussed in Section 2.1 is supplemented with information from an industry PIRT relative to LOCAs in PWRs containing high burnup fuel (Boyack et al., 2001), the key points of which are summarized in (Meyer, 2001).

2.2.1 Plant Transient Phenomena

Section 4.1.1 of (Meyer, 2001) notes a small number of fuel-related models in plant transient codes that need to be scrutinized for LOCA calculations in PWRs with high burnup fuel, because they are thought to be of high importance.

Gas pressure and Rod Free Volume

The fuel rod initialization is based on the PAD5 fuel performance code (Bowman et al., 2017); see related Limitation #4 in Section 7.2. The related WCOBRA/TRAC-TF2 code models were assessed in response to requests for additional information (RAIs) 36 through 39 on the **FSLOCA** EM, and found to be reasonable for the analysis of high burnup fuel. One additional consideration pertinent to higher burnup fuel rods is the potential for the pellet-to-clad bond to impede axial gas communication within the fuel rod. This phenomenon is addressed in Section 3.6.3.

Cladding Temperature

Parameters important to the calculation of cladding temperature were already captured in the **FSLOCA** EM, as the peak cladding temperature is a figure of merit for licensing-basis LOCA analysis.

Burst Criteria

The burst criterion is highly important since it directly relates to the parameter of interest for the cladding rupture calculations within this topical report. The cladding rupture models within the WCOBRA/TRAC-TF2 code are discussed in Section 3.3.

Location of Burst

[**]**^{a,c} within this topical report. Therefore, the location of burst is not considered significant to this topical report since the occurrence of rupture would fail the required outcome of the cladding rupture calculations.

Time-Dependent Gap-Size Heat Transfer

As previously discussed, an assessment of the gap conductance (i.e., gap heat transfer) in WCOBRA/TRAC-TF2 is discussed in Section 3.1 herein, and an assessment of the cladding deformation in WCOBRA/TRAC-TF2 is discussed in Section 3.2 herein.

2.2.2 Transient Fuel Rod Phenomena

Section 4.1.3 of (Meyer, 2001) identifies a small number of transient fuel-related phenomena that need to be appropriately modeled for LOCA calculations in PWRs with high burnup fuel, because they are thought to be of high importance.

Heat Resistance in the Gap

As previously discussed, an assessment of the gap conductance (i.e., gap heat transfer) in WCOBRA/TRAC-TF2 is discussed in Section 3.1 herein.

Heat Resistance in the Oxide

The heat resistance of the oxide layer can increase the stored energy inside the fuel rod at the onset of the LOCA. This increase in stored energy is addressed in a conservative manner as discussed under "Clad Oxidation" in Section 2.1.1 and in Section 3.4.

Cladding Oxidation Magnitude

The PIRT for LOCAs in PWRs and boiling water reactors (BWRs) containing high burnup fuel (Boyack et al., 2001) discussed both cladding oxidation and pre-existing oxidation on the cladding in the Category A – Plant Transient Analysis PIRT and the Category B – Integral Testing PIRT.

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]a,c

It is also discussed in Section 5.1.2 of (Billone et al., 2008) that the corrosion layer was found to be partially protective with regard to growth of a high-temperature oxidation layer with time. [

]a,c

Size of Burst Opening

[ ]a,c within this topical report. Therefore, the size of the burst opening is not considered significant to this topical report since the occurrence of rupture would fail the required outcome.

Burst Criteria

As previously discussed, an assessment of the burst criteria in WCOBRA/TRAC-TF2 is discussed in Section 3.3 herein.

Time of Burst

[ ][a,c] within this topical report. Therefore, the time of burst is not considered significant to this topical report since the occurrence of rupture would fail the required outcome.

2.3 REFERENCES

1. Billone, M., et al., July 2008, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents," NUREG/CR-6967.

2. Bowman, A. B., et al., 2017, "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1.

3. Boyack, B. E., et al., December 2001, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel," NUREG/CR-6744.

4. Kobelak, J. R., et al., November 2016, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1 (NRC Safety Evaluation dated September 12, 2017, ADAMS Accession Number (package) ML17207A124).

5. Kobelak, J. R., et al., December 2020, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs," WCAP-18446-P.

6. Meyer, R. O., September 2001, "Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel," NUREG-1749.

7. Nissley, M. E., et al., January 2005, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," WCAP-16009-P-A.

8. Pan, G., et al., March 2023, "Westinghouse **AXIOM**® Cladding for Use in Pressurized Water Reactor Fuel," WCAP-18546-P-A.

3 WCOBRA/TRAC-TF2 FUEL ROD MODEL UPDATES

The fuel rod models in the code identified in Section 2 are assessed in this section relative to the analysis of higher burnup fuel. WCOBRA/TRAC-TF2 fuel rod models from Section 8.3 of (Kobelak et al., 2016) which are shown to be appropriate or conservative are maintained without modification. Any updated or new models are described within this section.

3.1 PELLET-CLADDING GAP CONDUCTANCE MODEL

The pellet-to-cladding gap conductance model is discussed in Section 8.3.2 of (Kobelak et al., 2016). Additional discussion of the gap conductance and gap width relative to the calculated fuel temperature is captured in the response to Part 3 of RAI #37 on the **FSLOCA** EM (Kobelak et al., 2016). It is noted in the response that **[**

]^{a,c}

In order to achieve an acceptable steady-state, calibration of the fuel stored energy in WCOBRA/TRAC-TF2 to values determined from PAD5 is necessary because accurate prediction along the length of the rod requires more complicated models than are present in WCOBRA/TRAC-TF2 (see Section 3.4). **[**

]^{a,c} The fuel rod initialization was the subject of NRC staff review during the **FSLOCA** EM licensing (e.g., RAI-37 on the **FSLOCA** EM). The WCOBRA/TRAC-TF2 fuel pellet average temperature initialization [

]^{a,c}

The reactor coolant system (RCS) and fuel rod conditions experience significant changes upon the LOCA transient initiation. The RCS depressurizes and approaches the containment pressure toward the end of the blowdown phase of a LBLOCA. Within the fuel rod, the cladding temperatures tend to increase as the heat transfer to the coolant reduces, the fuel temperatures decrease as the core becomes subcritical and stored energy is transferred from the fuel, and the rod internal pressure decreases given the changes in pressure and temperature (and the potential associated fuel rod deformation). Under these conditions, the cladding expands and the fuel pellet shrinks, which in turn can reduce the contact pressure or re-open the gap. The associated decrease in gap conductance can be represented by the difference in the **[**

]^{a,c}

<u>W</u>COBRA/TRAC-TF2 simulates the noted important phenomena following the initiation of the LOCA transient. Figure 3.1-2 shows <u>W</u>COBRA/TRAC-TF2-predicted the gap conductance and gap width for a high burnup rod for an example PWR plant. **[**

]^{a,c}

a,c

Figure 3.1-1: PAD5 Predicted Gap Conductance for Example PWR Plant as a function of Gap Width

a,c

Figure 3.1-2: WCOBRA/TRAC-TF2 Gap Conductance and Gap Width for Example PWR Plant High Burnup Fuel Rod

3.2 CLADDING DEFORMATION

[

]^{a,c}

[

][a,c]

a,c

Figure 3.2-1: Creep Rate of [

]a,c

3.3 CLADDING RUPTURE

[

]a,c Afterward, the existing models and necessary updates are discussed in Section 3.3.2.

3.3.1 Effect of Hydrogen Uptake into the Cladding

As discussed in (Billone, et al., 2008), hydrogen which enters the cladding during normal operation reduces the ductility of the cladding. Therefore, the cladding rupture temperature and associated circumferential strain could be influenced by the presence of any substantial hydrogen. Since hydrogen enters the cladding from the corrosion process during normal operation, high burnup **AXIOM** clad fuel would contain hydrogen. Therefore, it is necessary to assess the effect of hydrogen on the existing models.

In order to assess the impact of hydrogen on the **AXIOM** cladding burst behavior, **[**

]a,c

3.3.2 Cladding Rupture Models

The models approved as part of the **FSLOCA** EM are discussed in Section 8.4.1 of (Kobelak et al., 2016), and the approved model for **AXIOM** cladding is discussed in Section 6.2.1.2.2 of (Pan et al., 2023). All of the existing models for the cladding rupture are **[**

]a,c

[

]a,c

[

]^{a,c}

a,c

Figure 3.3-1: Comparison of LOCA Burst Testing for Pre-Hydrided versus As-Fabricated AXIOM Cladding

a,c

Figure 3.3-2: [ ]a,c Burst Temperature versus Engineering Hoop Stress Curve Compared to AXIOM Cladding Burst Data

a,c

Figure 3.3-3: [ ]^{a,c} Burst Temperature versus Engineering Hoop Stress Curve Compared to [ ]^{a,c} Burst Testing for AXIOM Cladding

a,c

Figure 3.3-4: [ ]a,c Temperature Curve

a,c

Figure 3.3-5: [

][a,c] Temperature Curve

3.4 FUEL ROD INITIALIZATION

In the **FSLOCA** EM, for calculations to satisfy the 10 CFR 50.46 acceptance criteria, the fuel rods **[**

]^{a,c} The same approach to fuel rod initialization is maintained for the cladding rupture calculation methodology described in this topical report.

The calibration of the initial fuel rod average temperatures in WCOBRA/TRAC-TF2 based on PAD5 fuel performance data is already **[**

]^{a,c} Therefore, no updates are required beyond demonstrating the applicability of PAD5 to higher burnup fuel; see Limitation #4 in Section 7.2.

[

]^{a,c}

3.5 SUSCEPTIBILITY TO FINE FRAGMENTATION

The burnup threshold for fine fragmentation has been studied across various experimental programs, and much of the data is also presented by the NRC in a RIL (Bales et al., 2021). The RIL generally takes a conservative view of the available data, and indicates that fine fragmentation could occur at pellet burnups as low as 55 GWd/MTU. That conclusion is based on extrapolation of the data reviewed therein and is anchored by a data point from the BWR rod sample 09-OL1L04 LOCA2 (Mileshina and Magnusson, August 2019). One aspect which influenced the threshold definition as noted in (Bales et al., 2021) following Figure 3 therein was that, at the time the RIL was written, "*no tests have quantified fragment size for comparison between 45 and 60 GWd/MTU.*"

[

]^{a,c}

[

][a,c]

[

]a,c

a,b,c

Figure 3.5-1: Mass Fractions of Fuel Fragments less than 1 mm (i.e. Fine Fragmentation) from Various Test Programs

a,b,c

Figure 3.5-2: Mass Fractions of Fuel Fragments less than 1 mm (i.e. Fine Fragmentation) from NRC Testing at Studsvik and the SCIP-IV Program

a,b,c

Figure 3.5-3: Mass Fractions of Fuel Fragments less than 1 mm (i.e. Fine Fragmentation) from [ ]a,c

3.6 LOCA TRANSIENT FISSION GAS RELEASE

3.6.1 Test Data Review and Model Development

There have been various tests over the last decade that have attempted to characterize the potential for additional fission gas release inside the fuel rod during a postulated LOCA. These programs include, but are not limited to, the Halden IFA-650 test series as well as several of the Studsvik Cladding Integrity Programs (SCIP). Fission gas release during pellet heating has been observed in these test programs; however, the **[**

][a,c]

[ ]a,c (3-1)

Where: **[**

]a,c

The WCOBRA/TRAC-TF2 code only models **[**

]a,c

3.6.2 Onset of Gas Release

[

]a,c

1) [

]a,c

[

]a,c

3.6.3 Axial Gas Communication

Fuel rod conditions evolve as they are irradiated during normal operation. When they are fabricated, there is a gap between the cladding and the fuel pellets which is typically filled with helium to pressures on the order of [ ]a,c. This is substantially less than the typical RCS operating pressure of approximately 2,250 psi. Therefore, as the fuel is burned, the cladding tends to creep down onto the fuel pellets due to the pressure differential, and the gap between the cladding and the fuel pellets closes. Once the gap closes, fuel rod design criteria preclude the gap from reopening during power operation which could otherwise result from the release of fission gas that occurs throughout the life of the fuel. As such, the fuel pellets and the cladding remain in contact for the remainder of the fuel rod operation. Over time, this allows for a bond to develop between the fuel pellets and the cladding (Billone et al., 2008).

[

]a,c

a,b,c

Figure 3.6-1: Transient Fission Gas Release Model with Associated Data

a,b,c

Figure 3.6-2: Transient Fission Gas Release Model with Associated Data [

]a,c

a,b,c

Figure 3.6-3: Comparison of Transient Fission Gas Release Model and [ ][a,c] to Figure 7 from Bales et al., 2021

a,b,c

Figure 3.6-4: Transient Fission Gas Release Model [ ]^{a,c} with Additional Data from SCIP-IV

3.7 PRE-BURST AXIAL FUEL RELOCATION

3.7.1 Packing Fraction Assessment with Burst

The existing model in the WCOBRA/TRAC-TF2 code for axial fuel relocation following fuel rod rupture is described in Section 8.6.1 of (Kobelak et al., 2016). **[**

]^{a,c}

3.7.2 Conditions for Relocation

[

]^{a,c}

[

]a,c

3.7.3 Pre-Burst Relocation Test Data Review

[

]a,c

[

]^{a,c}

3.7.4 Pre-Burst Fuel Relocation Model

[

]^{a,c}

- [

]a,c

a,b,c

Figure 3.7-1: Packing Fraction versus Burnup for the SCIP-III LOCA Tests[1] (Packing Fraction Ranges with Midpoint)

[1] Sample burnup of the SCIP-III rods refers to the uncorrected rod average burnup associated with the sample as provided in the respective Studsvik reports.

a,b,c

Figure 3.7-2: Packing Fraction versus Burnup for the PBF and SCIP-III LOCA Tests[2] (Packing Fraction Ranges)

[2] Sample burnup of the SCIP-III rods refers to the uncorrected rod average burnup associated with the sample as provided in the respective Studsvik reports.

3.8 PROPERTIES OF NUCLEAR FUEL ROD MATERIALS

The properties of nuclear fuel rod materials included in the WCOBRA/TRAC-TF2 code for uranium-dioxide fuel are discussed in Section 11.4 of (Kobelak et al., 2016), and the properties for **AXIOM** cladding are discussed in (Pan et al., 2023). There is no impact to the cladding material properties from (Pan et al., 2023) for the analysis of high burnup fuel. The changes to the fuel pellet models for analysis of high burnup fuel are discussed in the following subsection(s).

3.8.1 Uranium Dioxide Thermal Conductivity

The UO_2 thermal conductivity model utilized in WCOBRA/TRAC-TF2 for licensing-basis applications accounts for the effects of burnup on thermal conductivity. **[**

][a,c]

A fuel thermal conductivity model was also developed for PAD5 (Bowman et al., 2017) which incorporates thermal conductivity degradation with burnup. The burnup dependent term, f(BU), is modeled after the **[** **]**[a,c], to account for thermal conductivity degradation as a function of burnup. The model coefficients are based on calibration to measured fuel centerline temperatures. The PAD5 fuel thermal conductivity model is described in Equations 3-2 through 3-4 as follows:

a,c

[ ] (3-2)

where:

[**]**[a,c] (3-3)

K_{95}	=	thermal conductivity for fuel with 95% theoretical density (W/cm-°C)
Bu	=	local burnup (GWd/MTU)
TC	=	fuel temperature (°C)

For thermal conductivity for fuel with any other density,

$$K_{density} = \frac{1-P}{1+\beta P} K_{100} = \left(\frac{1-P}{1+\beta P}\right) \times 1.08 \times K_{95} \qquad (3\text{-}4)$$

where:

P	=	Fractional porosity of the fuel ring (1-density)
β	=	a coefficient which is equal to 0.5 for P≤ 0.05 and which is equal to 1.0 for P> 0.05
K_{100}	=	thermal conductivity for fuel with 100% theoretical density

[

]^{a,c}

3.8.2 Uranium Dioxide Density

The W̲COBRA/TRAC-TF2 model for the fuel pellet density is discussed in Section 11.4.1 of (Kobelak et al., 2016). Specifically, the (cold) density for uranium-dioxide is assumed to be:

$$\rho_{UO_2} = 684.86 f_D \tag{3-5}$$

where f_D is the fraction of theoretical density and is input by the user. The density ρ_{UO_2} has units of lbm/ft^3.

Typical theoretical densities for UO_2 fuel manufactured by Westinghouse can range from approximately **[ ]^{a,c}** The approximate theoretical density for **ADOPT** fuel pellets is **[ ]^{a,c}** per Section 1.1 of (Hallman et al., 2022). **[ ]^{a,c}**

a,c

Figure 3.8-1: Comparison of the Modified NFI and PAD5 UO_2 Pellet Thermal Conductivity Models up to 100 GWd/MTU Burnup for Fuel at 95% of Theoretical Density

3.9 REFERENCES

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9. **[**

]^{a,c}

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27. [

]^{a,c}

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33. [

34.

35.

36.

]^{a,c}

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4 WCOBRA/TRAC-TF2 KINETICS AND DECAY HEAT MODEL UPDATES

The WCOBRA/TRAC-TF2 kinetics and decay heat model is discussed in Section 9 of (Kobelak et al., 2016). As discussed therein, the ANSI/ANS 5.1-1979 standard (ANS, 1979) is utilized within the WCOBRA/TRAC-TF2 code. There is no specific burnup limitation associated with the use of the ANSI/ANS 5.1-1979 standard. However, the implementation of the standard requires detailed physics evaluations of PWR fuel lattice designs. The codes used to perform these calculations for the **FSLOCA** EM are discussed in the response to RAI #25 in (Kobelak et al., 2016). Additionally, there are some limitations related to the calculation of the neutron capture correction in the standard.

The WCOBRA/TRAC-TF2 kinetics and decay heat model is updated for analysis of higher burnup fuel with higher initial fuel rod enrichment. The various updates are discussed in the following sub-sections.

4.1 NUCLEAR PHYSICS DATA

It was noted in RAI #23 to (Kobelak et al., 2016) that in the WCOBRA/TRAC-TF2 modeling of various important physical parameters related to fuel burnup, the burnup range presented was limited to **[** **]**^{a,c} assembly average burnup. This resulted in Limitation and Condition #5 on the **FSLOCA** EM. Westinghouse indicated that the adequacy of the fitting parameters to the physics calculations presented in these figures would be revisited if seeking approval to rod average burnups beyond **[** **]**^{a,c}

The supporting physics data utilized in the WCOBRA/TRAC-TF2 are updated to be valid for rod average burnups up to **[** **]**^{a,c} and up to **[** **]**^{a,c} The updated physics data are based on the PARAGON2 code (Ouisloumen et al., 2021), which is NRC-approved to **[** **]**^{a,c} The nuclear physics data was added directly into the WCOBRA/TRAC-TF2 code rather than curve fitting the data. The information presented in Figures 9-1 through 9-3 and Figures 9-5 through 9-15 of (Kobelak et al., 2016) is presented in Figures 4.1-1 through 4.1-14 herein for the updated physics data up to a burnup of **[** **]**^{a,c} and up to **[** **]**^{a,c} Note that this is the same nuclear physics data that is described in (Harper, December 2022) for the incremental burnup extension and in the higher enrichment topical report (Kucukboyaci et al., 2023); the data presented herein is simply to a broader combination of burnups and enrichments.

a,c

Figure 4.1-1: U-235 Fission Fraction (Updated Figure 9-1 from (Kobelak et al., 2016))

a,c

Figure 4.1-2: Pu-239 Fission Fraction (Updated Figure 9-2 from (Kobelak et al., 2016))

a,c

Figure 4.1-3: U-238 Fission Fraction (Updated Figure 9-3 from (Kobelak et al., 2016))

a,c

Figure 4.1-4: $\overline{\beta}$ versus Burnup (Updated Figure 9-5 from (Kobelak et al., 2016))

a,c

Figure 4.1-5: Prompt Neutron Lifetime (Updated Figure 9-6 from (Kobelak et al., 2016))

a,c

Figure 4.1-6: Prompt Energy Release (Updated Figure 9-7 from (Kobelak et al., 2016))

a,c

Figure 4.1-7: Total Energy Release (Updated Figure 9-8 from (Kobelak et al., 2016))

a,c

Figure 4.1-8: Delayed Group I Lambda (Updated Figure 9-9 from (Kobelak et al., 2016))

a,c

Figure 4.1-9: Delayed Group II Lambda (Updated Figure 9-10 from (Kobelak et al., 2016))

a,c

Figure 4.1-10: Delayed Group III Lambda (Updated Figure 9-11 from (Kobelak et al., 2016))

a,c

Figure 4.1-11: Delayed Group IV Lambda (Updated Figure 9-12 from (Kobelak et al., 2016))

a,c

Figure 4.1-12: Delayed Group V Lambda (Updated Figure 9-13 from (Kobelak et al., 2016))

a,c

Figure 4.1-13: Delayed Group VI Lambda (Updated Figure 9-14 from (Kobelak et al., 2016))

a,c

Figure 4.1-14: U-238 Capture / Fission Ratio as a Function of Initial Enrichment and Burnup (Updated Figure 9-15 from (Kobelak et al., 2016))

4.2 NEUTRON CAPTURE CORRECTION

There are three conditions related to the use of Equation 11 from (ANS, 1979) to calculate the neutron capture correction. The first is that the equation is only valid for shutdown times up to 10,000 seconds. After 10,000 seconds, Table 10 of (ANS, 1979) lists maximum values which can be used. The WCOBRA/TRAC-TF2 code **[**

]^{a,c}

The second condition is a maximum operating time of 4 years; however, this limitation could be exceeded for various nuclear designs (e.g., for fuel assemblies which are operated through three 24-month cycles).
[

]^{a,c}

[ ] a,c

Where:

[

]^{a,c}

The third condition from the ANS standard related to the use of Equation 11 for the neutron capture correction is that the number of fissions per initial fissile atom is less than 3.0. **[**

]^{a,c}

[

]a,c

a,c

Figure 4.2-1A: Comparison of WCOBRA/TRAC-TF2 and [ ]^{a,c}

a,c

Figure 4.2-1B: Comparison of WCOBRA/TRAC-TF2 and [ ]^{a,c}

a,c

Figure 4.2-1C: Comparison of WCOBRA/TRAC-TF2 and [ ]^{a,c}

a,c

Figure 4.2-2A: Comparison of W̲COBRA/TRAC-TF2 and [

]^{a,c}

a,c

Figure 4.2-2B: Comparison of W̲COBRA/TRAC-TF2 and [

]^{a,c}

a,c

Figure 4.2-2C: Comparison of W̲COBRA/TRAC-TF2 and [

]^{a,c}

a,c

Figure 4.2-3A: Comparison of W̲COBRA/TRAC-TF2 and [
]a,c

a,c

Figure 4.2-3B: Comparison of W̲COBRA/TRAC-TF2 and [
]a,c

a,c

Figure 4.2-4A: Comparison of WCOBRA/TRAC-TF2 and [

]a,c

a,c

Figure 4.2-4B: Comparison of WCOBRA/TRAC-TF2 and [

]a,c

a,c

Figure 4.2-5: Number of Fission per Initial Fissile Atom for a Representative Fuel Array

4.3 NORMALIZED FISSION INTERACTION FREQUENCY

The normalized fission interaction frequency from the NRC-approved **FSLOCA** EM is discussed in Section 9.3 of (Kobelak et al., 2016), and is calculated based on the coefficients presented in Table 9-5 therein. It is noted that the model in (Kobelak et al., 2016) is **[**
]^{a,c}

Figure 4.3-1 shows the **[**

]^{a,c} Based on this observation, an updated model is developed based on the PARAGON2 data.

The normalized fission interaction frequency model is updated to use **[**

]^{a,c}

$$\left[\qquad \right]^{a,c} \qquad (4\text{-}1)$$

Where,

[

]^{a,c}

Figures 4.3-2 through 4.3-9 show comparisons of the PARAGON2 data to the results using the revised model. As seen from the figures, the model **[**
]^{a,c} As such, the proposed model is considered acceptable over the desired range of conditions.

Table 4.3-1 Updated Normalized Fission Interaction Frequency Model Coefficients

a,c

Table 4.3-2 Normalized Fission Interaction Frequency Difference Between Enrichment Values at each Moderator Density

a,c

a,c

Figure 4.3-1: Normalized Fission Interaction Frequency versus [ ]a,c

a,c

Figure 4.3-2: Comparison of [ ]a,c and Results of Revised Model

a,c

Figure 4.3-3: Comparison of [ ]a,c and Results of Revised Model

a,c

Figure 4.3-4: Comparison of [ ]^{a,c} and Results of Revised Model

a,c

Figure 4.3-5: Comparison of [ ]^{a,c} and Results of Revised Model

a,c

Figure 4.3-6: Comparison of [ ]a,c and Results of Revised Model

a,c

Figure 4.3-7: Comparison of [ ]a,c and Results of Revised Model

a,c

Figure 4.3-8: Comparison of [ ]a,c and Results of Revised Model

a,c

Figure 4.3-9: Comparison of [ ]a,c and Results of Revised Model

4.4 GAMMA ENERGY REDISTRIBUTION

The modeling approach for gamma energy redistribution discussed in Section 9.6 of (Kobelak et al., 2016) used the DOT code (Disney et al., 1970) and BUGLE-80 (BUGLE-80, 1980) library to derive the data presented therein. With the Nuclear Regulatory Commission's approval of the PARAGON2 code (Ouisloumen et al., 2021) and its cross-section library, the Generalized Energy Deposition Model (GEDM) transfer matrix and the Gamma Energy Spectrum data are re-generated.

The dimensional problem for the recalculation with PARAGON2 uses **[**

]a,c Thus, the information presented in Figures 9-16 through 9-19 of (Kobelak et al., 2016) is still valid to describe the methodology used, **[** **]a,c**

The Gamma Photon Energy Spectrum data are re-calculated with the PARAGON2 gamma module based on **[**

]a,c and replace Table 9-10 of (Kobelak et al., 2016).

The data based on PARAGON2 supersede the data generated using DOT methodology. Thus, **[**

]a,c presented in Table 9-11 and illustrated in Figure 9-20 of (Kobelak et al., 2016) no longer apply.

The updated GEDM transfer matrix results for the 15x15 fuel design are presented in Table 4.4-2, which replaces Table 9-12 of (Kobelak et al., 2016).

Finally, Section 9.6.2 of (Kobelak et al., 2016) indicates that **[**

]a,c This conclusion remains valid for PARAGON2; therefore, the information in Table 9-13 of (Kobelak et al., 2016) is not re-created.

Table 4.4-1 Normalized Gamma Photon Energy Based on PARAGON2

Group	Energy		Normalized
	Upper	Lower	
	(MeV)		Particle Source

a,c

Table 4.4-2 Typical 15x15 GEDM Gamma Transfer Matrix

a,c

4.5 REFERENCES

1. American Nuclear Society, 1979, "Decay Heat in Light Water Reactors," ANSI/ANS 5.1-1979.

2. BUGLE-80 Gamma Cross Sections, 1980, ORNL RSIC DLC-76.

3. Disney, R. K., et al., 1970, "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Preparation," Volume 5, Two Dimensional Discrete Ordinates Transport Techniques, WANL-PR-(LL)-034, Westinghouse Astro Nuclear Laboratory.

4. Harper, Z. S., December 2022, "Submittal of Set 1 of Responses to the Second Round of Requests for Additional Information on Westinghouse Topical Report WCAP-18446-P/NP, 'Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs' (Proprietary/Non-Proprietary)," LTR-NRC-22-47.

5. Kobelak, J. R., et al., 2016, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1 and WCAP-16996-NP-A, Revision 1.

6. Kucukboyaci, V. N., et al., 2023, "Higher Enrichment for Westinghouse and Combustion Engineering Fuel Designs," WCAP-18773-P.

7. ORNL, June 2011, "Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1.

8. Ouisloumen, M., et al., July 2021, "Qualification of the Two-Dimensional Transport Code PARAGON2," WCAP-18443-P-A.

5 FUEL ROD CLADDING RUPTURE CALCULATION METHOD

5.1 TREATMENT OF REGIONS

The LOCA break spectrum within the approved **FSLOCA** EM (Kobelak et al., 2016) is split into two different regions for the purpose of the uncertainty analyses. The term "Region I" refers to the SBLOCA range of break sizes, and the term "Region II" refers to the LBLOCA range of break sizes. **[**

]^{a,c} For example, it may be desirable to analyze break sizes up to a transition break size (see Tregoning et al., April 2008) relative to the prediction of cladding rupture.

There are some differences in the uncertainty analysis execution for Region I versus Region II **[**

]^{a,c} These differences include the following:

- **[**

]^{a,c}

Each of these differences is addressed in the following sub-sections. Additionally, it is **[**

]^{a,c} or the associated analysis approaches that have been approved as part of the **FSLOCA** EM.

Region II from the approved **FSLOCA** EM covers all break sizes from 1-ft^2 (which corresponds to approximately a 13.5-inch break) up to a double-ended guillotine (DEG) break. As such, **[**

]^{a,c}

5.1.1 Break Type and Size

The treatment of the break type and size for each of the regions is as follows.

[

]^{a,c}

[

]a,c

[3] Note that this trend is also observed for the 2-loop plants based on the hot rod behavior. However, due to the [

]a,c is also conservative. See Section 6.1.1 for more detail.

a,c

Figure 5.1-1: SBLOCA Core Boiloff Uncovery Peak Cladding Temperature versus Break Diameter for a 3-Loop Westinghouse-Designed PWR (Figure 31.2-1 of Kobelak et al., 2016)

a,c

Figure 5.1-2: SBLOCA Core Boiloff Uncovery Peak Cladding Temperature versus Break Diameter for a 4-Loop Westinghouse-Designed PWR (Figure 4-2 of Mercier, July 2018)

a,c

Figure 5.1-3: SBLOCA Core Boiloff Uncovery Peak Cladding Temperature versus Break Diameter for a 2-Loop Westinghouse-Designed PWR (Figure 3.4.3-8 of Schoedel, September 2021)

a,c

Figure 5.1-4: SBLOCA RCP Trip Uncovery Peak Cladding Temperature versus Break Diameter for a 2-Loop Westinghouse-Designed PWR (Figure 3.4.3-11 of Schoedel, September 2021)

a,c

Figure 5.1-5: IBLOCA Break Spectrum Peak Cladding Temperature versus Break Diameter for a 3-Loop Westinghouse-Designed PWR (Case A)

a,c

Figure 5.1-6: IBLOCA Break Spectrum Peak Cladding Temperature versus Break Diameter for a 3-Loop Westinghouse-Designed PWR (Case B)

a,c

Figure 5.1-7: IBLOCA Break Spectrum Peak Cladding Temperature versus Break Diameter for a 4-Loop Westinghouse-Designed PWR (Figure 4-3 of Mercier, July 2018)

a,c

Figure 5.1-8: IBLOCA Break Spectrum Peak Cladding Temperature versus Break Diameter for a 2-Loop Westinghouse-Designed PWR (Figure 3.4.3-9 of Schoedel, September 2021)

a,c

Figure 5.1-9: Illustration of Regions for the Cladding Rupture Calculations

5.2 UNCERTAINTY CONTRIBUTORS

Any changes to the various uncertainty contributors, uncertainty parameter distributions, and treatment from the approved **FSLOCA** EM (Kobelak et al., 2016) are discussed in this section. Any parameters which are not discussed are treated in a manner consistent with the approved **FSLOCA** EM.

5.2.1 Peaking Factor Uncertainty

The approach to account for peaking factor uncertainties within the **FSLOCA** EM is discussed in Section 25.2.1 of (Kobelak et al., 2016). The general approach described therein is maintained for the method to perform the cladding rupture calculations described in this topical report. However, it is noted in Table 25.2-3 of (Kobelak et al., 2016) that **[**

][a,c]

5.2.2 Decay Heat Uncertainty

The decay heat uncertainty sampled within the **FSLOCA** EM is presented in Table 29-4 of (Kobelak et al., 2016). The uncertainty is **[**

][a,c] is retained for the cladding rupture calculation method described in this topical report.

5.2.3 [ ][a,c]

[

][a,c]

[

]a,c

5.2.3.1 [ ]a,c

[

]a,c

[

][a,c]

5.2.3.2 [ ][a,c]

[

][a,c]

5.2.3.3 [ ]a,c

[

]a,c

[

]a,c

5.2.4 Offsite Power Availability

The offsite power availability assumption primarily influences the LOCA transient for two reasons. First, the operation of the RCPs is tied to the offsite power availability. Second, the timing of the safety injection entering the RCS is influenced by whether or not offsite power is available. The modeling of offsite power availability for the cladding rupture calculations is [

]a,c

Sensitivity studies describing the impact of the offsite power availability assumption on the IBLOCA transient response are discussed in Section 6.1.1 herein. The modeling of offsite power availability for [

]a,c

- **[**

]a,c

Table 5.2-1 Rod Bow F_Q Uncertainties

a,c

Notes:

1. Uncertainties are given in terms of one standard deviation divided by average value (coefficient of variation).
2. Argall et al. (1979)
3. [ ]^{a,c}
4. Bounding, minimum rod bow uncertainty
5. Combustion Engineering (1983)

a,c

Figure 5.2-1: Comparison of Predicted and Measured Void Profiles for ORNL – THTF Test 3.09.10CC from the As-Submitted FSLOCA EM (Figure 13.4.2-24 of Frepoli et al., 2010)

a,c

Figure 5.2-2: Comparison of Predicted and Measured Void Profiles for ORNL – THTF Test 3.09.10EE from the As-Submitted FSLOCA EM (Figure 13.4.2-26 of Frepoli et al., 2010)

a,c

Figure 5.2-3: Comparison of Predicted and Measured Void Profiles for ORNL – THTF Test 3.09.10CC from the Approved FSLOCA EM (Figure 13.4.2-24 of Kobelak et al., 2016)

a,c

Figure 5.2-4: Comparison of Predicted and Measured Void Profiles for ORNL – THTF Test 3.09.10EE from the Approved FSLOCA EM (Figure 13.4.2-26 of Kobelak et al., 2016)

a,c

Figure 5.2-5: IBLOCA [ ]a,c Sensitivity Study Results

a,c

Figure 5.2-6: Comparison of 2-Loop PWR IBLOCA Break Spectrum Studies with Different Offsite Power Availability Assumptions

a,c

Figure 5.2-7: Comparison of 3-Loop PWR IBLOCA Break Spectrum Studies with Different Offsite Power Availability Assumptions

a,c

Figure 5.2-8: Comparison of 4-Loop PWR IBLOCA Break Spectrum Studies with Different Offsite Power Availability Assumptions

5.3 MISCELLANEOUS CONSIDERATIONS

The various other considerations identified in Section 5.1 that can differ across regions are discussed in this section, as well as the modeling approach for steam generator tube plugging (SGTP).

5.3.1 [ ]a,c

[

]a,c

5.3.2 Treatment of [ ]a,c Fuel Assemblies

[

]a,c

5.3.3 Control Rod Insertion

[

]a,c

5.3.4 Counter-Current Flow Limitation

[

]a,c

5.3.5 Steam Generator Tube Plugging

[

]a,c

[

]a,c

5.4 REFERENCES

1. Argall, B. M., et al., 1979, "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1.

2. Combustion Engineering, June 1983, "Fuel & Poison Rod Bowing," CENPD-225-P-A.

3. Frepoli, C., et al., November 2010, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P and WCAP-16996-NP.

4. Harper, Z. S., June 2022, "Submittal of Set 3 of Responses and Clarifications to Requests for Additional Information on Westinghouse Topical Report WCAP-18446-P/NP, 'Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs' (Proprietary/Non-Proprietary)," LTR-NRC-22-22.

5. Kobelak, J. R., et al., 2016, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1 and WCAP-16996-NP-A, Revision 1.

6. Mercier, E. J., July 2018, "'Information to Satisfy the **FULL SPECTRUM** LOCA (**FSLOCA**) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)' (Proprietary/Non-Proprietary)," LTR-NRC-18-50.

7. Schoedel, A. J., September 2021, "Extension of **FULL SPECTRUM** LOCA (**FSLOCA**) Evaluation Methodology to 2-Loop Westinghouse Pressurized Water Reactors (PWRs) with

Information to Satisfy Limitations and Conditions Specific to 2-loop Plant Types (Proprietary/Non-Proprietary)," LTR-NRC-21-22.

8. Tregoning, R., et al., April 2008, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," NUREG-1829.

6 SENSITIVITY STUDIES AND DEMONSTRATION ANALYSIS

In this section, sensitivity studies are presented with the WCOBRA/TRAC-TF2 code to demonstrate the impact of variations in several different input parameters to the IBLOCA transient response. Some of these studies **[**

][a,c] Other studies are included simply to illustrate the impact of various key input parameters on IBLOCA transient progression.

A demonstration analysis is then presented for **[**

][a,c] within this topical report.

6.1 INTERMEDIATE BREAK LOCA SENSITIVITY STUDIES

6.1.1 Offsite Power Availability

In the context of modeling a LOCA transient response, the availability of offsite power can influence the operation of the RCPs, and the timing of pumped safety injection since diesel generators are required to support injection when offsite power is lost. Since the impact of RCP operation is break size-dependent, the offsite power availability studies are performed across a range of break sizes.

6.1.1.1 4-Loop PWR Study

The PCT results versus break size for the 4-loop study were presented in Figure 5.2-8. The PCT with **[**

][a,c]

[

]^a,c^

6.1.1.2 3-Loop PWR Study

The PCT results versus break size for the 3-loop study were presented in Figure 5.2-7. The transient behavior from two different break sizes is reviewed since the **[**

]a,c

[

]a,c

6.1.1.3 2-Loop PWR Study

The PCT results versus break size for the 2-loop study were presented in Figure 5.2-6. The transient behavior from two different break sizes is reviewed for the 2-loop PWR studies as well due to the
[

]a,c

[

]a,c

[

][a,c]

6.1.1.4 Conclusions

For breaks at the [

][a,c] These conclusions support the modeling approach for offsite power availability discussed in Section 5.2.4.

a,c

Figure 6.1.1-1: RCP Speed for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-2: Pumped SI Mass Flow Rate for the [ ]^{a,c} from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-3: Pressurizer Pressure for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-4: Void Fraction at the Break for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-5: Break Mass Flow Rate for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-6: Vessel Fluid Inventory for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-7: Void Fraction at the Top of the Steam Generator Tubes for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-8: Hot Assembly Void Distribution for the [ ]a,c LOOP Case from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-9: Hot Assembly Void Distribution for the [ ]a,c OPA Case from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-10: Hot Rod PCT for the [ ]a,c from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-11: Accumulator Injection for the [ ]^{a,c} from the 4-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-12: RCP Speed for the [ ]^{a,c} from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-13: Pumped SI Mass Flow Rate for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-14: Pressurizer Pressure for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-15: Vessel Fluid Inventory for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-16: Void Fraction at the Top of the Steam Generator Tubes for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-17: Hot Rod PCT for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-18: RCP Speed for the [ ]^{a,c} from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-19: Void Fraction at the Break for the [ ]^{a,c} from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-20: Break Mass Flow Rate for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-21: Vessel Fluid Inventory for the [ ]^{a,c} from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-22: Void Fraction at the Top of the Steam Generator Tubes for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-23: Hot Assembly Void Distribution for the [ ]a,c LOOP Case from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-24: Hot Assembly Void Distribution for the [ ]a,c OPA Case from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-25: Hot Rod PCT for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-26: Accumulator Injection for the [ ]a,c from the 3-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-27: RCP Speed for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-28: Break Void Fraction for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-29: Break Mass Flow Rate for the [ ]^{a,c} from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-30: Vessel Fluid Inventory for the [ ]^{a,c} from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-31: Void Fraction at the Top of the Steam Generator Tubes for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-32: Hot Assembly Void Distribution for the [ ]a,c LOOP Case from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-33: Hot Assembly Void Distribution for the [ ]a,c OPA Case from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-34: Hot Rod PCT for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-35: Hot Rod PCT for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-36: Dummy Rod PCT for the [ ]^{a,c} from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-37: TCHF for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-38: TMIN for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-39: Dummy Rod Blowdown PCT for the [ ][a,c] from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-40: Linear Heat Rate Near the PCT Elevation for the [ ]^{a,c} from the 2-Loop PWR Offsite Power Availability Study

a,c

Figure 6.1.1-41: Core Average Moderator Density for the [ ]a,c from the 2-Loop PWR Offsite Power Availability Study

6.1.2 Upper Head Temperature

After the initiation of the transient, the liquid inventory in the upper head drains primarily through the guide tubes. RCS depressurization causes flashing in this region, affecting the local pressure and the upper head draining rate. A two-phase mixture will then continue the draining process, until the mixture level drops below the top of the guide tubes. The bulk fluid temperature in the upper head can influence the timing of flashing, and thereby the draining behavior during the LOCA transient.

6.1.2.1 4-Loop PWR Study

A single sensitivity study was executed with a 4-loop PWR model, where in one case the **[**

]^{a,c}

6.1.2.2 3-Loop PWR Study

A single sensitivity study was executed with a 3-loop PWR model, where in one case the **[**

]^{a,c}

[

][a,c]

6.1.2.3 Conclusions

The studies with a **[**

][a,c]

a,c

Figure 6.1.2-1: Upper Head Temperature for the 4-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-2: Upper Head Void Fraction Above the Guide Tubes for the 4-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-3: Liquid Mass Flow Rate in the Guide Tubes for the 4-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-4: Hot Rod PCT for the 4-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-5: Upper Head Temperature for the 3-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-6: Upper Head Void Fraction Above the Guide Tubes for the 3-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-7: Vessel Fluid Inventory for the 3-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-8: Liquid Mass Flow Rate in the Guide Tubes for the 3-Loop PWR Upper Head Bulk Fluid Temperature Study

a,c

Figure 6.1.2-9: Hot Rod PCT for the 3-Loop PWR Upper Head Bulk Fluid Temperature Study

6.1.3 Steam Generator Tube Plugging

The SGTP level can influence the evolution of the transient via differences in resistance, primary-to-secondary side heat transfer, and primary side vessel fluid inventory. These studies contrast the impact of low versus high tube plugging on IBLOCA transient progression. The sensitivity studies were conducted at the different tube plugging levels as large runsets with different boundary conditions and sampled uncertainties across the simulations in each runset. The cases discussed in the following sub-sections portray the typical behavior observed across the various simulations for each PWR class and tube plugging level.

6.1.3.1 4-Loop PWR Study

For the 4-Loop PWR study, **[**

][a,c]

6.1.3.2 3-Loop PWR Study

For the 3-Loop PWR study, **[**

][a,c]

[

]a,c

6.1.3.3 2-Loop PWR Study

For the 2-Loop PWR study, **[**

]a,c

6.1.3.4 Competing Effects

There are competing effects from the change in the steam generator tube pugging level for intermediate breaks (similar to for small and large breaks). **[**

]a,c

[

] ^{a,c}

6.1.3.5 Conclusions

The impact of the modeled SGTP level **[**

] ^{a,c}

a,c

Figure 6.1.3-1: Steam Generator Tube Void Fraction for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-2: Break Mass Flow Rate for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-3: Break Void Fraction for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-4: Vessel Fluid Inventory for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-5: Differential Mass of Steam Flowing through the Hot Leg (High SGTP minus Low SGTP) for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-6: Pressurizer Pressure for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-7: Accumulator Injection for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-8: Peak Cladding Temperature for the 4-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-9: Steam Generator Tube Void Fraction for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-10: Break Mass Flow Rate for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-11: Break Void Fraction for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-12: Vessel Fluid Inventory for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-13: Differential Mass of Steam Flowing through the Hot Leg (High SGTP minus Low SGTP) for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-14: Pressurizer Pressure for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-15: Accumulator Injection for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-16: Peak Cladding Temperature for the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-17: Steam Generator Tube Void Fraction for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-18: Break Mass Flow Rate for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-19: Break Void Fraction for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-20: Vessel Fluid Inventory for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-21: Differential Mass of Steam Flowing through the Hot Leg (High SGTP minus Low SGTP) for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-22: Pressurizer Pressure for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-23: Accumulator Injection for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-24: Peak Cladding Temperature for the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-25: Pressurizer Pressure for a Case from the 2-Loop PWR Steam Generator Tube Plugging Study where [ ]a,c

a,c

Figure 6.1.3-26: Vessel Fluid Inventory for a Case from the 2-Loop PWR Steam Generator Tube Plugging Study where [ ][a,c]

a,c

Figure 6.1.3-27: Peak Cladding Temperature for a Case from the 2-Loop PWR Steam Generator Tube Plugging Study where [ ]^{a,c}

a,c

Figure 6.1.3-28: Cumulative Distribution Function of the PCTs from the 2-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-29: Cumulative Distribution Function of the PCTs from the 3-Loop PWR Steam Generator Tube Plugging Study

a,c

Figure 6.1.3-30: Cumulative Distribution Function of the PCTs from the 4-Loop PWR Steam Generator Tube Plugging Study

6.1.4 Upper Core Plate Flow Area

The flow area at the upper core plate (UCP) can influence the availability of liquid in the upper regions of the reactor vessel to reach the core. While this flow area is fixed for a given PWR, it can vary from reactor-to-reactor due to design differences. As such, sensitivity studies were conducted modeling the actual flow area at the upper core plate, and then **[** **]**[a,c]

6.1.4.1 4-Loop PWR Study

A single sensitivity study was executed with a 4-loop PWR model, where in one case the actual flow area through the upper core plate was modeled, and in the other case **[**

][a,c]

6.1.4.2 3-Loop PWR Study

A single sensitivity study was executed with a 3-loop PWR model, where in one case the actual flow area through the upper core plate was modeled, and in the other case **[**

][a,c]

6.1.4.3 Conclusions

[

]^a,c^

a,c

Figure 6.1.4-1: Average Channel Liquid Mass Flow Rate at the Upper Core Plate for the 4-Loop PWR Upper Core Plate Flow Area Study

a,c

Figure 6.1.4-2: Integrated Average Channel Liquid Mass Flow Rate at the Upper Core Plate for the 4-Loop PWR Upper Core Plate Flow Area Study

a,c

Figure 6.1.4-3: Hot Rod PCT for the 4-Loop PWR Upper Core Plate Flow Area Study

a,c

Figure 6.1.4-4: Average Channel Liquid Mass Flow Rate at the Upper Core Plate for the 3-Loop PWR Upper Core Plate Flow Area Study

a,c

Figure 6.1.4-5: Hot Rod PCT for the 3-Loop PWR Upper Core Plate Flow Area Study

a,c

Figure 6.1.4-6: Integrated Average Channel Liquid Mass Flow Rate at the Upper Core Plate for the 3-Loop PWR Upper Core Plate Flow Area Study

6.1.5 Main Steam Safety Valve Setpoint

Variations in the main steam safety valve (MSSV) setpoint can influence the energy transfer across the steam generator tubes, which in turn can influence the primary side pressure behavior throughout the LOCA transient.

6.1.5.1 4-Loop PWR Study

For the 4-loop Westinghouse-designed PWR, sensitivity studies were run with a setpoint pressure
[

]^{a,c}

6.1.5.2 3-Loop PWR Study

For the 3-loop Westinghouse-designed PWR, sensitivity studies were run with a setpoint pressure
[

]^{a,c}

6.1.5.3 Conclusions

The MSSV setpoint pressure [

]a,c

a,c

Figure 6.1.5-1: Steam Generator Primary-Side and Secondary-Side Pressures for the 4-Loop PWR MSSV Setpoint Pressure Study

a,c

Figure 6.1.5-2: Steam Generator Secondary-Side Liquid Temperature for the 4-Loop PWR MSSV Setpoint Pressure Study

a,c

Figure 6.1.5-3: Hot Rod PCT for the 4-Loop PWR MSSV Setpoint Pressure Study

a,c

Figure 6.1.5-4: Steam Generator Primary-Side and Secondary-Side Pressures for the 3-Loop PWR MSSV Setpoint Pressure Study

6.1.6 Accumulator Cover Pressure

Variations in the accumulator cover pressure can influence the accumulator injection timing, the injection rate, and the duration of the accumulator injection.

6.1.6.1 4-Loop PWR Study

For the 4-loop PWR accumulator cover pressure study, a nominal pressure of **[**

][a,c]

6.1.6.2 3-Loop PWR Study

For the 3-loop PWR accumulator cover pressure study, a nominal pressure of **[**

][a,c]

6.1.6.3 Conclusions

Variations in the accumulator cover pressure are observed to **[**

][a,c]

[

]a,c

a,c

Figure 6.1.6-1: Pressurizer Pressure for the 4-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-2: Integrated Accumulator Injection for the 4-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-3: Vessel Fluid Inventory for the 4-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-4: Hot Rod PCT for the 4-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-5: Pressurizer Pressure for the 3-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-6: Integrated Accumulator Injection for the 3-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-7: Vessel Fluid Inventory for the 3-Loop PWR Accumulator Cover Pressure Study

a,c

Figure 6.1.6-8: Hot Rod PCT for the 3-Loop PWR Accumulator Cover Pressure Study

6.1.7 Accumulator Water Volume

Variations in the accumulator water volume can influence the injection rate and duration of the accumulators.

6.1.7.1 4-Loop PWR Study

For a 4-loop Westinghouse-designed PWR, a typical accumulator water volume is **[**

][a,c]

6.1.7.2 3-Loop PWR Study

For a 3-loop Westinghouse-designed PWR, a typical accumulator water volume is **[**

][a,c]

[

]^{a,c}

6.1.7.3 Conclusions

Variations in the accumulator water volume can have a [

]^{a,c}

a,c

Figure 6.1.7-1: Accumulator Injection for the 4-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-2: Reactor Vessel Upper Head Pressure for the 4-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-3: Vessel Fluid Inventory for the 4-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-4: Hot Rod Peak Cladding Temperature for the 4-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-5: Accumulator Injection for the 3-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-6: Reactor Vessel Upper Head Pressure for the 3-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-7: Vessel Fluid Inventory for the 3-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-8: Hot Rod Peak Cladding Temperature for the 3-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-9: Hot Assembly Channel Collapsed Liquid Level for the 4-Loop PWR Accumulator Water Volume Study

a,c

Figure 6.1.7-10: Hot Assembly Channel Collapsed Liquid Level for the 3-Loop PWR Accumulator Water Volume Study

6.1.8 Accumulator Temperature

Variations in the accumulator temperature can impact the post-injection LOCA transient evolution. Typical accumulator temperatures ranges from approximately **[** **]**^{a,c}

6.1.8.1 4-Loop PWR Study

The 4-loop PWR study considered a nominal accumulator temperature of **[**

]^{a,c}

6.1.8.2 3-Loop PWR Study

The 3-loop PWR study considered a nominal accumulator temperature of **[**

]^{a,c}

[

]a,c

6.1.8.3 Conclusions

Variations in the accumulator temperature tend to have **[**

]a,c

a,c

Figure 6.1.8-1: Pressurizer Pressure for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-2: Liquid Temperature in the Accumulator Line Near the Cold Leg for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-3: Lower Plenum Liquid Temperature for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-4: Downcomer Collapsed Liquid Level for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-5: Vessel Fluid Inventory for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-6: Hot Rod PCT for the 4-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-7: Pressurizer Pressure for the 3-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-8: Liquid Temperature in the Accumulator Line Near the Cold Leg for the 3-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-9: Lower Plenum Liquid Temperature for the 3-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-10: Downcomer Collapsed Liquid Level for the 3-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-11: Vessel Fluid Inventory for the 3-Loop PWR Accumulator Temperature Study

a,c

Figure 6.1.8-12: Hot Rod PCT for the 3-Loop PWR Accumulator Temperature Study

6.1.9 Accumulator Line Resistance

Variations in the accumulator line resistance can influence the injection rate and duration of the accumulators.

6.1.9.1 4-Loop PWR Study

This sensitivity study considered a variation of **[**

][a,c]

6.1.9.2 3-Loop PWR Study

This sensitivity study also considered a variation of **[**

][a,c]

[

][a,c]

6.1.9.3 Conclusions

Variations in the accumulator line resistance can have a **[**

][a,c]

a,c

Figure 6.1.9-1: Accumulator Injection for the 4-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-2: Vessel Fluid Inventory for the 4-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-3: Hot Assembly Collapsed Liquid Level for the 4-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-4: Hot Rod PCT for the 4-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-5: Accumulator Injection for the 3-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-6: Vessel Fluid Mass for the 3-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-7: Hot Assembly Collapsed Liquid Level for the 3-Loop PWR Accumulator Line Resistance Study

a,c

Figure 6.1.9-8: Hot Rod PCT for the 3-Loop PWR Accumulator Line Resistance Study

6.2 DEMONSTRATION ANALYSIS

A demonstration analysis utilizing the cladding rupture calculation methodology developed in this topical report is performed for a 4-Loop Westinghouse-designed PWR with cold side injection. The demonstration analysis is performed for **[**

][a,c]

6.2.1 Demonstration Analysis Inputs

The inputs utilized for the demonstration analysis were selected to simplify the analysis, and **[**

][a,c] The plant operating ranges, peaking factors, and pumped safety injection flow considered in the analysis are presented in Tables 6.2.1-1 through 6.2.1-3.

[

][a,c]

Table 6.2.1-1 Plant Operating Range Analyzed in the Demonstration Analysis

a,c

Parameter		As-Analyzed Value or Range

Table 6.2.1-1 Plant Operating Range Analyzed in the Demonstration Analysis

a,c

Parameter		**As-Analyzed Value or Range**

Table 6.2.1-2 Peaking Factor Inputs Considered in the Demonstration Analysis

a,c

Table 6.2.1-3 Pumped Safety Injection Flows Modeled in the Demonstration Analysis

a,c

a,c

Figure 6.2.1-1: Demonstration Analysis PWR Upper Core Plate Structure Configuration, Including Low Power Peripheral Assembly Designation

6.2.2 Discussion of Results

[

][a,c]

a,c

Table 6.2.2-1 [ ]^{a,c}

a,c

Figure 6.2.2-1: Pressurizer Pressure for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-2: Vapor Mass Flow Rate through the Loop Seal Region for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-3: Vessel Fluid Inventory for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-4: Accumulator Injection Mass Flow Rate for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-5: Hot Assembly Two-Phase Mixture Level for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-6: PCT for all the Fuel Rods for the Analysis Case from the Demonstration Analysis

a,c

Figure 6.2.2-7: Dummy Rod PCT and Burst Temperature for the Analysis Case from the Demonstration Analysis

6.2.3 Conclusions from Demonstration Analysis

[

]a,c

6.3 REFERENCES

1. Kobelak, J. R., et al., 2016, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology),” WCAP-16996-P-A, Revision 1 and WCAP-16996-NP-A, Revision 1.

7 SUMMARY AND IMPLEMENTATION

In summary, this topical report has described the application of the **FSLOCA** EM (Kobelak et al., 2016) framework for the prediction of fuel rod cladding rupture (with a focus on high burnup fuel rods). Various updates were made to the WCOBRA/TRAC-TF2 code for the analysis of higher enrichment, high burnup fuel rods. Since the methodology developed in this topical report originates from the **FSLOCA** EM, it is prudent to review the limitations and conditions from that evaluation model for application to the methodology herein. The various updates made within this topical report provide justification for the modification of several limitations and conditions from the **FSLOCA** EM.

A review of the **FSLOCA** EM limitations and conditions is provided in Section 7.1. New limitations associated with this topical report are discussed in Section 7.2.

Clarity regarding the interaction between this topical report and the ECCS acceptance criteria (10 CFR 50.46 / 10 CFR 50.46c) is provided in Section 7.3.

7.1 REVIEW OF LIMITATIONS AND CONDITIONS ON THE FULL SPECTRUM LOCA METHODOLOGY

The SER for the **FSLOCA** EM contains 15 limitations and conditions on the NRC-approved **FSLOCA** EM. A summary of each limitation and condition (L&C) and an assessment on whether the L&C remains applicable to this topical report is provided in this section. L&Cs that remain applicable to this topical report in whole or in part are propagated into Section 7.2 and must be satisfied for implementation of the methodology described in this topical report.

Limitation and Condition Number 1

Summary

The **FSLOCA** EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

Assessment

The cladding rupture calculation method in this topical report is related to the short-term LOCA response. As such, this L&C is not applicable to this topical report.

Limitation and Condition Number 2

Summary

The **FSLOCA** EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Assessment

Limitation #1 in Section 7.2 extends a modified version of the requirements of this L&C to the LOCA cladding rupture calculation methodology. Therefore, this L&C is superseded by Limitation #1 in Section 7.2.

Limitation and Condition Number 3

Summary

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Assessment

This L&C remains applicable and is retained herein for the cladding rupture calculations.

Limitation and Condition Number 4

Summary

The decay heat uncertainty multiplier will be **[**
]^{a,c} The analysis simulations for the **FSLOCA** EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Assessment

The aspect of the L&C regarding the decay heat uncertainty sampling is applicable to this topical report and is retained. The aspect of the L&C that the WCOBRA/TRAC-TF2 code cannot be applied for transient time longer than 10,000 seconds following shutdown unless the decay heat model is shown to be acceptable is also applicable to this topical report and is retained. Sampled values for the decay heat multiplier have been provided to the NRC for an extensive number of analyses with the **FSLOCA** EM. As such, the specific decay heat multipliers sampled for analyses of cladding rupture following this topical report will not be reported. Limitation #2 in Section 7.2 is an updated version of this L&C which is applicable to the cladding rupture calculations.

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup is limited to **[**
]^{a,c} respectively.

Assessment

This L&C is not applicable to this topical report and is superseded by Limitation #3 in Section 7.2.

Limitation and Condition Number 6

Summary

The fuel performance data for analyses with the **FSLOCA** EM should be based on the PAD5 (Bowman et al., 2017) code (at present), or the latest NRC-approved version of a Westinghouse fuel performance code (future), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the fuel performance data should adhere to the NRC-approved methodology.

Assessment

This L&C remains applicable for the cladding rupture calculations. An updated version of this L&C regarding the fuel performance data for application of the cladding rupture calculation methodology described in this topical report is included as Limitation #4 in Section 7.2.

Limitation and Condition Number 7

Summary

The **[**

]^{a,c}

Assessment

This L&C is not applicable to the methodology developed in this topical report based on the discussion in Section 5.2.3.1.

Limitation and Condition Number 8

Summary

The **[**

]^{a,c}

Assessment

The aspect of this L&C that the **[** **]^{a,c}** remains applicable and is retained based on the discussed in Section 5.2.3.2. The aspect of this L&C that the **[** **]^{a,c}** is not applicable to the methodology developed in this topical report based on the discussion in Section 5.2.3.3. An updated version of this L&C regarding the **[** **]^{a,c}** for application of the cladding rupture calculation methodology described in this topical report is included as Limitation #5 in Section 7.2.

Limitation and Condition Number 9

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[** **]^{a,c}** for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Assessment

The **[** **]^{a,c}** based on the discussion in Section 5.2.3. The requested **[** **]^{a,c}** were completed and docketed with the NRC per (Mercier, July 2018) for 4-loop PWRs and (Schoedel, September 2021) for 2-loop PWRs. A similar demonstration must be submitted to support review of implementation of this methodology for CE-designed PWRs. As such, this limitation is not carried forward into the cladding rupture calculation methodology described in this topical report.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[**

][a,c]

must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant from the **FSLOCA** EM. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Assessment

The requested demonstrations were completed and docketed with the NRC per (Mercier, July 2018) for 4-loop Westinghouse-designed PWRs and (Schoedel, September 2021) for 2-loop Westinghouse-designed PWRs. It was found that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and that the **[**
][a,c]

The requested demonstration for the 4-loop Westinghouse-designed PWRs (Mercier, July 2018) has supported the incorporation of the **FSLOCA** EM into the licensing basis for numerous operating 4-loop PWRs. The requested demonstration for 2-Loop Westinghouse-designed PWRs (Schoedel, September 2021) is available for review in support of implementation of this methodology for 2-Loop Westinghouse-designed PWRs. A similar demonstration must be submitted to support review of implementation of this methodology for CE-designed PWRs. Limitation #1 in Section 7.2 requires that these demonstrations be NRC-approved to extend the **FSLOCA** EM to these classes of plants prior to the application of this methodology. Furthermore, with the **[**

][a,c] Therefore, the first aspect of this L&C is considered to have been satisfied. The minimum break area sampled for Region II analysis will remain 1 ft^2 consistent with this L&C as discussed in Section 5.1. As such, this limitation is not carried forward into the cladding rupture calculation methodology described in this topical report.

Limitation and Condition Number 11

Summary

There are various aspects of this Limitation and Condition, which are summarized below:

1. The **[** **]**[a,c] the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[**
][a,c] and the Region I and Region II analyses seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.

2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for PCT, maximum local oxidation (MLO), and core-wide oxidation (CWO) which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.
3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Assessment

This Limitation and Condition generally remains applicable to the methodology in this topical report, with some clarifications as follows:

1. The **[ ]**[a,c] analysis seed(s), and the analysis inputs will be declared and documented prior to performing the cladding rupture calculations. The **[ ]**[a,c] and the analysis seed(s) will not be changed once they have been declared and documented. Since there is **[ ]**[a,c] and this aspect of the L&C is not applicable to the cladding rupture methodology described in this topical report.
2. This element of the limitation is not applicable to the methodology described in this topical report.
3. Plant operating ranges which are sampled for the cladding rupture calculations will be provided in the analysis submittal associated with the cladding rupture calculations.

An updated version of this L&C for application of the cladding rupture calculation methodology described in this topical report is included as Limitation #6 in Section 7.2.

Limitation and Condition Number 12

Summary

The plant-specific dynamic pressure loss from the steam generator secondary-side to the MSSVs must be adequately accounted for in analysis with the **FSLOCA** EM.

Assessment

This limitation remains applicable for the methodology described in this topical report, and the plant-specific dynamic pressure loss from the steam generator secondary-side to the MSSVs will be adequately accounted for in the cladding rupture calculations.

Limitation and Condition Number 13

Summary

In plant-specific models for analysis with the **FSLOCA** EM: 1) the **[ ]**[a,c] and 2) the **[ ]**[a,c]

Assessment

This limitation remains applicable for the methodology described in this topical report, and the [

]a,c

Limitation and Condition Number 14

Summary

For analyses with the **FSLOCA** EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

Assessment

As noted in Section 7.3, this topical report does not address compliance with the ECCS acceptance criteria. As such, this L&C is not applicable to the cladding rupture calculation methodology described in this topical report.

Limitation and Condition Number 15

Summary

The Region II analysis will be executed twice; once assuming LOOP and once assuming OPA. The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

The [ ]a,c

Assessment

The first part of this limitation remains applicable to the methodology described in this topical report (see Section 5.2.4). The [

]a,c for the methodology described in this topical report. An updated version of this L&C for application of the cladding rupture calculation methodology described in this topical report is included as Limitation #7 in Section 7.2.

7.2 LIMITATIONS ASSOCIATED WITH THIS TOPICAL REPORT

The limitations for the application of the method described in this topical report are discussed in this section.

Limitation #1: This topical report is applicable to Westinghouse-designed 2-loop PWRs equipped with UPI, 3-loop PWRs with cold-side injection, and 4-loop PWRs with cold-side injection as well as CE-designed PWRs. The methodology for the LOCA cladding rupture calculations can only be applied to the Westinghouse 2-loop PWR and CE PWR designs once the **FSLOCA** EM is approved for these designs. Any applicable differences in the approved methodology for these plant designs must be addressed in the cladding rupture calculations. Furthermore, any deviations from the method described in this topical report should be described and justified.

Limitation #2: The following conditions apply with regard to decay heat modeling and sampling in the cladding rupture calculations for all regions: 1) the decay heat uncertainty will be **[ ]^{a,c}** and 2) the cladding rupture calculations cannot be performed for transient times longer than 10,000 seconds following shutdown unless the decay heat model is shown to be acceptable for the analyzed core conditions. The latter limitation is **[ ]^{a,c}**

Limitation #3: The maximum fuel rod length-average burnup and fuel assembly average burnup permitted with this topical report is **[ ]^{a,c}**. Details behind the various burnup-related limitations associated with this topical report are provided in Section 7.2.1.

Limitation #4: The fuel performance data utilized to initialize the fuel rods for the cladding rupture calculations should be from a fuel performance code which includes the effect of thermal conductivity degradation and is NRC-approved through the fuel rod average burnups and initial fuel enrichments that are analyzed. The **[ ]^{a,c}** and the generation of all the fuel performance data should adhere to the NRC-approved methodology.

Limitation #5: The **[ ]^{a,c}**

Limitation #6: For each cladding rupture analysis performed:

1. The **[ ]^{a,c}** analysis seed(s), and the analysis inputs will be declared and documented prior to performing the cladding rupture calculations. The **[ ]^{a,c}** and the analysis seed(s) will not be changed once they have been declared and documented.
2. Plant operating ranges which are sampled for the cladding rupture calculations will be provided in the analysis submittal associated with the cladding rupture calculations.

Limitation #7: Plant-specific applications of the cladding rupture methodology for Region II should include two complete sets of sampled statistical evaluations: 1) a complete set with offsite power available, and 2) a second complete set with a loss-of-offsite power. For each set, the calculated statistical results at the 95/95 probability, confidence level should result in margin to cladding rupture. The **[ ]^{a,c}** to provide the required 95/95 probability, confidence statement that addresses the margin to rupture.

Limitation #8: L&Cs Number 3, 12, and 13 from (Kobelak et al., 2016) must be satisfied for the application of this cladding rupture methodology.

Limitation #9: This topical report is applicable to standard UO_2 or **ADOPT** fuel with **AXIOM** cladding.

Limitation #10: This topical report is applicable to un-poisoned fuel, fuel with IFBA, and fuel with Gadolinia. This limitation does not preclude the use of wet annular burnable absorbers (WABAs) or other discrete burnable absorbers during the lifetime of an assembly.

Limitation #11: A maximum of **[** **]**^{a,c} is permitted with this topical report. This limitation results from the maximum enrichment that is supported by the WCOBRA/TRAC-TF2 kinetics and decay heat module.

7.2.1 Details Associated with the Fuel Rod Average Burnup Limitation

There are various models discussed within this topical report that have various burnup-related limitations associated with them. Those models and the associated burnup-related limitations are as follows:

- The fuel pellet thermal conductivity model discussed in Section 3.8.1 is applicable to a rod average burnup of **[** **]**^{a,c}
- The nuclear physics data supporting the kinetics and decay heat model was provided from PARAGON2, which is valid to a burnup of **[** **]**^{a,c} in this topical report.
- The neutron capture correction was shown to be conservative to at least a burnup of **[** **]**^{a,c} in this topical report.
- The data from the high burnup fuel rods supporting various fuel rod models in this topical report is from samples with burnups up to approximately **[** **]**^{a,c}

Based on these various burnups, the topical report is limited to rods with an average burnup of no more than **[** **]**^{a,c}

7.3 COMPLIANCE WITH 10 CFR 50.46 / 10 CFR 50.46C ACCEPTANCE CRITERIA

The method described in this topical report can specifically be used to determine, with high probability, that cladding rupture will not occur. This method does not directly satisfy any of the current 10 CFR 50.46 or proposed 10 CFR 50.46c acceptance criteria, although it could indirectly support a demonstration of core coolability.

7.4 REFERENCES

1. Bowman, A. B., et al., 2017, "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1.

2. Kobelak, J. R., et al., 2016, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1 and WCAP-16996-NP-A, Revision 1.

3. Mercier, E. J., July 2018, "'Information to Satisfy the **FULL SPECTRUM** LOCA (**FSLOCA**) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)' (Proprietary/Non-Proprietary)," LTR-NRC-18-50.

4. Schoedel, A. J., September 2021, "Extension of **FULL SPECTRUM** LOCA (**FSLOCA**) Evaluation Methodology to 2-loop Westinghouse Pressurized Water Reactors (PWRs) with Information to Satisfy Limitations and Conditions Specific to 2-loop Plant Types (Proprietary/Non-Proprietary)," LTR-NRC-21-22.