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LTR-NRC-20-70

December 14, 2020

Subject: Submittal of Topical Report WCAP-18446-P / WCAP-18446-NP, Revision 0, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs" (Proprietary / Non-Proprietary)

Reference: LTR-NRC-20-53 dated September 11, 2020, "Transmittal of the Pre-Submittal Meeting Slides for Topical Report WCAP-18446-P, 'Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs,'" ADAMS Accession Number ML20255A320

Enclosed are proprietary and non-proprietary versions of Topical Report WCAP-18446-P / WCAP-18446-NP, Revision 0, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs," dated December 2020, submitted for review and approval under the NRC's licensing topical report program for referencing in licensing actions. Approval of this topical report is requested by July 2022.

This submittal contains proprietary information of Westinghouse Electric Company LLC ("Westinghouse"). In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Nuclear Regulatory Commission's ("Commission's") regulations, we are enclosing with this submittal an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of this submittal or the Westinghouse Affidavit should reference AW-20-5120 and should be addressed to Korey L Hosack, Manager, Licensing, Analysis, and Testing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Suite 133, Cranberry Township, PA 16066.

Korey L. Hosack, Manager
Licensing, Analysis, and Testing

cc: Ekaterina Lenning (NRC)
Dennis Morey (NRC)

Enclosures

1. Affidavit AW-20-5120
2. WCAP-18446-P, Revision 0, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs" (Proprietary)
3. WCAP-18446-NP, Revision 0, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs" (Non-Proprietary)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Korey L. Hosack, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting that the proprietary WCAP-18446-P Enclosure 2 to LTR-NRC-20-70 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable

AFFIDAVIT

others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

AFFIDAVIT

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2020 12 14

Korey L. Hosack, Manager
Licensing, Analysis, and Testing

LTR-NRC-20-70
Enclosure 3

WCAP-18446-NP, Revision 0, "Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs"

(Non-Proprietary)

December 2020

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

WCAP-18446-NP December 2020

Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs

WCAP-18446-NP
Revision 0

Incremental Extension of Burnup Limit for Westinghouse and Combustion Engineering Fuel Designs

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December 2020

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EXECUTIVE SUMMARY

Core designs in the nuclear industry are frequently constrained by licensed fuel burnup limits. These fuel burnup limits can lead to an increased number of feed fuel assemblies each reload, which reduces fuel cycle efficiency. It is desirable to increase existing fuel rod burnup limits to improve the fuel cycle efficiency. The purpose of this topical report is to provide an integrated package of method updates required to supersede existing fuel rod burnup limits.

Many different licensed Westinghouse methodologies have limits on fuel rod average burnup (typically between 60 and 62 GWd/MTU). These limits exist for a variety of different reasons. This topical report provides justification to exceed those existing fuel rod burnup limits under a specific set of conditions. The new fuel rod average burnup limit associated with this topical report is **[ ]**^{a,c} (i.e., an incremental burnup extension). The rationale for the existing limitations and justification for their increase is provided within this topical report.

The databases supporting fuel rod performance calculations and the performance of the fuel assembly structure were reviewed. It was concluded that sufficient data exists to justify a fuel rod average burnup of **[ ]**^{a,c} for Westinghouse and Combustion Engineering fuel using the existing codes and models.

In order to implement the incremental burnup extension, various analyses and evaluations must be completed. The fuel assembly mechanical design criteria which must be met under the incremental burnup extension are defined within this topical report. The fuel rod design criteria which must be satisfied under the incremental burnup extension are also discussed in this topical report.

The impact of the incremental burnup extension on the various safety analyses was assessed. The main impacts of the incremental burnup extension on loss-of-coolant accident (LOCA) analysis were found to be primarily in two areas: 1) Assessment of fuel fragmentation, relocation, and dispersal (FFRD), and 2) Assessment of research findings supporting the proposed 10 CFR 50.46c criteria. In order to preclude concerns related to FFRD, **[**

]^{a,c}

For the non-LOCA transients such as rod ejection, locked rotor, and steamline break, it will be confirmed **[**

]^{a,c} The new reactivity insertion accident (RIA) criteria described in RG 1.236 will also be addressed for high burnup fuel during plant-specific implementation as described in this topical report.

The impact of the incremental burnup extension on the radiological consequence analyses was assessed. Finally, the applicability of the incremental burnup extension to different fuel designs, cladding materials, fuel materials, and burnable absorbers is discussed. The limitations for application of the incremental burnup extension are provided. A summary of the implementation requirements is also provided.

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ACRONYMS and NOMENCLATURE

ADFBN	advanced debris filter bottom nozzle
ADU	ammonium diuranate
ANS/ANSI	American Nuclear Society / American National Standards Institute
AO	axial offset
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BN	bottom nozzle
BOL	beginning-of-life
BPVC	Boiler and Pressure Vessel Code
BWR	boiling water reactor
C&G	creep and growth
CE	Combustion Engineering
CHF	critical heat flux
CFR	Code of Federal Regulations
COLR	core operating limits report
CQD	Code Qualification Document
CRE	control rod ejection
DEG	double-ended guillotine
DFBN	debris filter bottom nozzle
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DS	double sided
EDF	Électricité de France
EM	evaluation model
EMDAP	evaluation model development and assessment process
ECCS	emergency core cooling system
ECR	equivalent cladding reacted
EOL	end-of-life
FA	fuel assembly
FCEP	fuel criteria evaluation process
FEA	finite element analysis
FFRD	fuel fragmentation, relocation, and dispersal
FGR	fission gas release
FHA	fuel handling accident
FSLOCA	**FULL SPECTRUM** LOCA
GDC	general design criteria
GT	guide tube
GTRF	grid-to-rod fretting
IBLOCA	intermediate-break loss-of-coolant accident
ID	inner diameter
IFBA	integral fuel burnable absorber
IFM	intermediate flow mixing
IN	information notice

ACRONYMS and NOMENCLATURE (continued)

IRI	incomplete rod insertion
JAEA	Japan Atomic Energy Agency
L&C	limitation and condition
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
LTA	lead test assembly
LTZ	**Low Tin ZIRLO**
M&E	mass and energy
MLO	maximum local oxidation
MSLB	main steamline break
NFI	Nuclear Fuels Industries
NGF	next generation fuel
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
NSSS	nuclear steam supply system
OD	outer diameter
OE	operating experience
OFA	optimized fuel assembly
OPA	offsite power available
PBF	Power Burst Facility
PCI	pellet-cladding interaction
PCMI	pellet-cladding mechanical interaction
PCT	peak cladding temperature
PIE	post irradiation exam
PIRT	phenomenon identification and ranking table
PNNL	Pacific Northwest National Laboratory
pRXA	partial recrystallization anneal
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
R/B	release-to-birth
RAI	request for additional information
RCA	control rod assembly
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
REA	rod ejection accident
RFA	robust fuel assembly
RG	regulatory guide
RIA	reactivity insertion (initiated) accident
RTDP	revised thermal design procedure
RTN	reconstitutable top nozzle

ACRONYMS and NOMENCLATURE (continued)

RXA	full recrystallization anneal
SBLOCA	small-break loss-of-coolant accident
SCIP	Studsvik Cladding Integrity Program
SE(R)	safety evaluation (report)
SGTR	steam generator tube rupture
SLB	steamline break
SRA	stress relief annealed
SRP	standard review plan
SS	single sided
T/H	thermal hydraulic
TCD	thermal conductivity degradation
TN	top nozzle
TRD	thermal reaction accumulated duty
(U)FSAR	(updated) final safety analysis report
UPI	upper plenum injection
US	United States
WCT-TF2	WCOBRA/TRAC-TF2
WTDP	Westinghouse thermal design procedure

1 OVERVIEW AND METHODOLOGY ROADMAP

1.1 BACKGROUND

Core designs in the nuclear industry are frequently constrained by licensed fuel burnup limits. These fuel burnup limits can lead to an increased number of feed fuel assemblies each reload, which reduces fuel cycle efficiency. It is desirable to increase existing fuel rod burnup limits to improve the fuel cycle efficiency. The purpose of this topical report is to provide an integrated package of method updates required to supersede existing fuel rod burnup limits.

Many different licensed Westinghouse methodologies have limits on fuel rod average burnup (typically between 60 and 62 GWd/MTU). These limits exist for a variety of different reasons. This topical report provides justification to exceed those existing fuel rod burnup limits under a specific set of conditions. The new fuel rod average burnup limit associated with this topical report is **[** **]**^{a,c} (i.e., an incremental burnup extension). The rationale for the existing limitations and justification for their increase is provided within this topical report.

The topical reports and methods for which existing burnup limits will be superseded via this topical report are identified in Section 1.4. The new limits which will be imposed, discussed throughout the topical report, are summarized in Section 7.1.

Note that the term "high burnup fuel" is used throughout this topical report. That term describes fuel rods with average burnup in excess of current burnup limits **[** **]**^{a,c} (fuel rod average burnup limit associated with this topical report).

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 various available regulatory guidance. Section 1.4 provides a list of approved topical reports and methods with burnup limits which are extended under the provisions of this topical report. Considerations related to fuel fragmentation, relocation, and dispersal (FFRD) are described in Section 1.5.

Section 2 describes the effect of the incremental burnup extension on the fuel assembly mechanical design. The design criteria which must be met for the incremental burnup extension are described in Section 2. It is demonstrated that the design criteria are met for the 17x17 Optimized Fuel Assembly (OFA) fuel design for the incremental burnup extension. The 17x17 OFA fuel design was selected as it is tends to be more limiting relative to many of the design criteria potentially affected by the incremental burnup extension as compared to all other currently-approved Westinghouse and Combustion Engineering fuel array designs. Application of the incremental burnup extension will include a demonstration that the design criteria identified in Section 2 are met on a design-specific / plant-specific basis.

The fuel rod design, nuclear design, and thermal-hydraulic design areas are discussed in Section 3. It is explained that the approved PAD5 fuel rod performance code (Bowman et. al., 2017) is validated for analysis of fuel rods under the incremental burnup extension, and the impacts of the incremental burnup extension on the fuel rod design criteria are discussed. It is also discussed that the incremental burnup extension does not require modifications or updates to any previously Nuclear Regulatory Commission (NRC)-approved topical reports assessing neutronics and nuclear design, since the methods implemented in the neutronic codes for fuel depletion remain unchanged and already contain the depletion capability to very high burnups. Some example nuclear designs for the current burnup limit and under the incremental burnup extension are presented. Similarly, no changes to any approved topical reports in the area of thermal-hydraulic design are required to support the burnup extension addressed in this topical report. Some additional requirements are imposed relative to **[**

]^{a,c} Application of the incremental burnup extension will include a demonstration that sufficient margin and cycle management strategies are available to address the fuel rod performance criteria and additional T/H requirements.

Section 4 describes the effect of the incremental burnup extension on licensing-basis loss-of-coolant accident (LOCA) analyses. A requirement to demonstrate no cladding rupture is prescribed for high burnup fuel to preclude concerns associated with FFRD. The method to calculate whether cladding rupture occurs is described, and a set of demonstration calculations is provided. It is also generically shown that the current Title 10 of the Code of Federal Regulations (10 CFR) 50.46 and draft 10 CFR 50.46c criteria will be met for high burnup fuel if the cladding rupture criterion is met. Application of the incremental burnup extension will include a demonstration that cladding rupture does not occur for high burnup fuel on a plant-specific basis.

The non-LOCA analysis (i.e., transient and containment analysis) methods are reviewed in Section 5. It was found that there is no impact from the incremental burnup extension on the existing non-LOCA codes, methods, and acceptance criteria. Implications resulting from Regulatory Guide (RG) 1.236 are also discussed in this section.

The impact of the incremental burnup extension on radiological consequence analyses is discussed in Section 6. Considering the requirements imposed on high burnup fuel (discussed in Sections 3 and 4 of the topical report) regarding **[**

]^{a,c}

Section 7 provides a summary of the information contained in the topical report, including the limitations on application of this topical report, as well as requirements for implementation of the topical report.

1.3 MAPPING OF TOPICAL REPORT TO REGULATORY GUIDANCE

There is no specific part of 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. As such, no exemption to 10 CFR is required to implement this topical report. Similarly, there is no specific regulatory guide which describes the required elements of an incremental burnup extension. However, there are several regulatory guides and industry documents which can inform the scope of the incremental burnup extension. The documents considered within this topical report are discussed in this section.

1.3.1 Standard Review Plan

Although Standard Review Plan (SRP) Section 4.2 has no specific numerical burnup limit, page 4.2-2 of SRP 4.2 Rev. 3 provides the following high-level reviewer guidance under Design Evaluation:

> "New fuel designs, new operating limits (e.g., rod burnup and power), and the introduction of new materials to the fuel system require a review to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents. The review also evaluates operating experience, direct experimental comparisons, detailed mathematical analyses (including fuel performance codes), and other information."

This topical report addresses the SRP 4.2 guidance with respect to an incremental increase in the allowable peak rod average burnup.

Specific SRP 4.2 Acceptance Criteria tied to high burnup effects on irradiated material properties include the following:

II.1.A (Fuel System Damage) – See Sections 2.4, 2.5, and 3.1 of this topical report.

II.1.B (Fuel Rod Failure) – See Sections 2.5 and 3.1 of this topical report.

II.1.B.iv (Fuel Pellet Centerline Melt) – See Section 3.1.6 of this topical report.

II.1.B.vi (PCI / Cladding Strain) – See Sections 3.1.3 and 3.1.4 of this topical report.

II.1.C (Fuel Coolability) – See Section 4.2 of this topical report with respect to LOCA, and Section 5.1.2 of this topical report with respect to RIA / RG 1.236.

II.2 (Description and Design Drawings – Design-specific Burnup Limit) – See Tables 2.2-1 through 2.2-6, Figures 2.2-1 through 2.2-3, and Figures 2.4-1 through 2.4-10 of this topical report, as well as the specific limitations for application of this topical report which are identified in Section 7.1.

II.3.A/B (Maximum Burnup OE / Prototype Testing) – See Section 2.5 of this topical report.

II.3.C.i (ECCS Performance Models with respect to Fuel Temperatures) – See Section 4 of this topical report.

II.3.C.ix (Fission Product Inventory) – See Section 6 of this topical report.

1.3.2 Emergency Core Cooling System Analysis Methods

In the early 2000s, a phenomenon identification and ranking table (PIRT) was developed for loss-of-coolant accidents (LOCAs) in pressurized water reactors (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 4.3 of this topical report.

More recently in 2015, a document was written to evaluate FFRD under LOCA conditions. It was written in (McCree, 2015) that "… research has shown as burnup exceeds 62 GWd/MTU, fuel becomes increasingly susceptible to FFRD." It is also written that "Given the burnup and utilization limitations on existing fuel designs, the staff does not foresee the ongoing research would identify a need to withdraw approval of existing fuel designs." Concerns related to FFRD under a postulated LOCA are discussed in Section 4.2.3 of this topical report.

1.3.3 Non-LOCA Safety Analysis Methods

RG 1.236 (NRC, June 2020): This regulatory guide provides guidance on acceptable analytical methods, assumptions, and limits for evaluating the nuclear reactor's initial response to a postulated PWR Reactivity Insertion Accident (RIA) (i.e., a control rod ejection accident), based on empirical data from in-pile, prompt power pulse test programs and analyses from several international publications on fuel rod performance under prompt power excursion conditions. The applicability of this guidance is limited to a maximum fuel rod average burnup of 68 GWd/MTU as indicated in Section C.1.1.3 of RG 1.236.

1.3.4 Source Term

There are a number of RGs and draft RGs related to the source term:

RG 1.25 (US NRC, March 1972): This regulatory guide provided assumptions for use in evaluating the consequences of a fuel handling accident. The assumptions in Regulatory Position C.1 were based on a peak average assembly burnup of 25,000 MWd/ton, which corresponds to a peak local burnup of approximately 45,000 MWd/ton. Although RG 1.25 was withdrawn in December 2016 (replaced by RGs 1.183 and 1.195), it is still cited in the current licensing basis for some operating plants.

RG 1.183 (US NRC, July 2000): RG 1.183 provides guidance to those plants that choose to change their licensing basis by adopting the alternative source term discussed in (Soffer et al, 1995) and 10 CFR 50.67. The core inventory release fractions listed in Table 2 (PWRs) of Regulatory Position C.3.2 have been determined to be acceptable for use with currently approved fuel types with a peak fuel rod average burnup up to 62 GWd/MTU. The non-LOCA gap fractions listed in Table 3 have been determined to be acceptable for use with currently approved fuel types with a peak fuel rod average burnup up to 62 GWd/MTU, provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWd/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, those calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load.

DG-1199 (US NRC, October 2009): This draft regulatory guide (proposed Revision 1 to RG 1.183) revised the footnote for the core inventory release fractions and non-LOCA gap fractions to specify a peak pellet burnup up to 70,000 MWd/MTU for boiling water reactors (BWRs) with no change to the 62,000 MWd/MTU peak rod average burnup limit for PWRs. An additional Table 4 is included for fission product inventory release from the power excursion during a reactivity-initiated accident. Additionally, Figure 1 therein provides a maximum allowable power operating envelope for non-LOCA gap fractions which extends to a nodal burnup of 68 GWd/MTU.

RG 1.195 (NRC, May 2003): The core inventory release fractions and non-LOCA gap fractions in Regulatory Position C.3.2 of RG 1.195 (Tables 1 and 2) contain the same burnup limitations as found in RG 1.183 (peak fuel rod average burnup of 62,000 MWd/MTU).

DG-1327 (NRC, July 2019): The NRC updated the generic gap fraction guidance in Appendix B of the July 2019 version of DG-1327. This draft version of the regulatory guide is referenced because Appendix B was removed from the final version issued as RG 1.236 in June 2020. The basis for the updated guidance in Appendix B of the July 2019 version of DG-1327 is provided in (Beyer and Clifford, June 2011) with a revision provided in (Clifford, June 2019). The fuel rod power operating envelope associated with the gap fractions is provided in Figure B-1 of DG-1327 (July 2019) up to a rod average burnup of 65 GWd/MTU.

1.4 SUPERSEDED BURNUP LIMITS

The incremental burnup extension discussed in this topical report will supersede existing burnup limits in NRC-approved topical reports and applications of NRC-approved design and analysis codes and methods which are all currently limited to a fuel rod average burnup of 62 GWd/MTU or lower. Such burnup limits have been explicitly stated in the relevant topical reports. The superseded burnup limits include, but are not limited to, the following:

- **Fuel Assembly Design Topical Report(s)**: (Davidson and Iorii, 1982) specifies a 36 GWd/MTU region average burnup limit for Optimized Fuel Assemblies (OFA).

  (Davidson and Kramer, September 1985) and (Davidson, February 1989) in conjunction with Section 3 of the NRC safety evaluation reports (SERs) included in Sections B and F of (Davidson and Ryan, 1995) have VANTAGE 5 and VANTAGE+ fuel assembly limitations of a maximum rod average burnup of 60 GWd/MTU.

  (Davidson and Kramer, December 1985) specifies a 60 GWd/MTU lead rod average burnup limit for Westinghouse Zircaloy-clad fuel designs. Originally interpreted under the FCEP process (Davidson, October 1994, as revised January 2002) to allow a change to the peak rod average burnup limit not to exceed 62 GWd/MTU in a unilateral letter (Liparulo, August 1994), the peak rod average burnup limit for these topical reports was later increased by the NRC to 62 GWd/MTU (Peralta, 2006) contingent upon the use of PAD 4.0 (Foster et al., July 2000). The Westinghouse Fuel Criteria Evaluation Process (Davidson, October 1994, as revised January 2002) is discussed further in Section 7.2 of this topical report.

(Davidson et al., March 1995) contains an NRC-approved fuel rod creep collapse methodology that was based, in part, on the observed axial gaps in 58 rods of current (1994 and earlier) Westinghouse fuel designs irradiated in four different reactors with a range of burnup levels between 5.0 to 60 GWd/MTU.

(Barsic et al., 2011) specifies a 62 GWd/MTU licensing basis lead rod average burnup limit for Westinghouse 17x17 next generation fuel (NGF) assemblies; however, the NGF design limit is 75 GWd/MTU.

A peak pin average burnup limit of 60 MWd/kgU is specified in Section 4.1 of (Fiero, August 2004) for Combustion Engineering (CE)-designed PWR fuel assemblies.

Limitation and Condition #3 in Section 4.0 of the NRC SER included in (Book et al., August 2007) limits the CE 16x16 NGF fuel assembly to a peak rod average burnup of 62 GWd/MTU, although the fuel performance models and methods were used to evaluate the fuel assembly up to a peak rod average burnup of 70 GWd/MTU.

- **Cladding Material Topical Report(s)**: Limitation and Condition #2 in Section 5 of the June 2005 NRC SER included in the **Optimized ZIRLO**™ high performance cladding material topical report (Schueren, 2006) restricts the fuel rod average burnup to 62 GWd/MTU for Westinghouse fuel designs and 60 GWd/MTU for CE fuel designs. (Hosack, February 2019) requested that the limit for the **CE16NGF**™ fuel design, which also uses **Optimized ZIRLO** cladding, be increased to 62 GWd/MTU; the standard 14x14 and 16x16 CE fuel design burnup limit would remain at 60 GWd/MTU. This request was approved in (Morey, May 2019).

- **Fuel Performance Topical Report(s)**: Previous versions of the fuel performance code, PAD 3.4 (Weiner et al., 1988) and PAD 4.0 (Foster et al., 2000) were approved up to the peak rod average burnup limits of (Davidson and Kramer, 1985), which are 60 GWd/MTU and 62 GWd/MTU, respectively. There is a limitation in Section 4.1, Item a) of the NRC SER included in (Bowman et al., 2017) that the PAD5 code and methodology can only be used for rod average burnups up to 62 GWd/MTU for all approved types of cladding.

  While no specific burnup limit is cited in (Garde et al., 2013), NRC safety evaluation (SE) Section 5.0 Limitation and Condition #3 cites (Davidson and Ryan, April 1995) and (Schueren, July 2006) for the H_2 pickup limit, and those references are limited to a peak rod average burnup of 60 GWd/MTU. In addition, it is stated in the response to request for additional information (RAI) #8.2 (Gresham, September 2011) in Section D of (Garde et al., 2013) that:

  > "CENPD-404-P-A identifies how the **ZIRLO**® corrosion model is applied in the CE NSSS plant applications. Specifically, as discussed in Section 4.5.2 of CENPD-404-P-A, the approach licensed by CENP for **OPTIN**™ for a peak rod average burnup of up to 60 MWd/kgU will be applied to **ZIRLO**®. The corrosion model for **ZIRLO**® cladding will be used to ensure that maximum expected oxide thickness will not exceed the required design limit for CENP nuclear fuel designs for fuel rod average burnups of up to 62 MWd/kgU. Further, Section 5.3.10 of CENPD-404-P-A discusses oxide buildup from a mechanical design standpoint. Corrosion is conservatively addressed by assuming the

same maximum reduction in base-metal wall thickness for the **ZIRLO®** and **OPTIN™** cladding materials."

- **Non-LOCA RIA Analysis Method(s)**: In the NRC SER on the Westinghouse rod ejection accident analysis methodology using multi-dimensional kinetics (Beard et al., October 2003), the Staff concluded that although the previous RG 1.77 limits were not conservative for cladding failure at higher fuel burnups, the more realistic analyses performed by Westinghouse, which have been confirmed by NRC-sponsored calculations, provided reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to the currently approved 62 GWd/MTU will neither (1) result in damage to the reactor coolant pressure boundary nor, (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core as specified in regulatory requirements (GDC 28). The existing NRC-approved rod ejection analysis methods, such as that described in (Beard et al., October 2003), remain unchanged for applications to the higher fuel burnup in compliance with the new RG 1.236. Effects of fuel burnup are accounted for in design input to the accident analysis, including the reactor core loading plan, core initial conditions, and fuel temperatures from a fuel performance code.

- **LOCA Analysis Method(s)**: Limitation and Condition #5 in Sections 4.6.3.1 and 5.0 of the NRC 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}. Section 4.0 of the NRC SER included in (Nissley et al., 2005) retains the prior limitations and conditions from (Bajorek et al., 1998) and (Dederer et al., 1999), where there is no specific burnup limitation. However, the primary reason for the burnup limitation is related to the assessment of the decay heat model in the WCOBRA/TRAC-TF2 code (the thermal-hydraulic code licensed as part of the **FULL SPECTRUM™** LOCA (**FSLOCA™**) evaluation model (EM)). Since the decay heat model in the WCOBRA/TRAC code (thermal-hydraulic code associated with prior best-estimate LOCA evaluation models) is similar to the WCOBRA/TRAC-TF2 model, the same limitation would be applicable to both codes.

1.5 CONSIDERATION OF FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL

The potential for fine fuel fragmentation has become better understood over the last decade due in no small part to in-pile and out-of-pile testing on high burnup fuel that has been conducted at the Nuclear Safety Research Reactor (NSRR), the Power Burst Facility (PBF), the CABRI reactor, the Halden reactor (IFA-650 test series), and at Studsvik Nuclear Laboratory. Experimental evidence indicates that the potential for fine fragmentation increases with higher fuel rod burnup.

The NRC has previously assessed the potential safety impacts of FFRD on operating reactors for the existing burnup limits. A safety assessment for the Condition IV non-LOCA events is provided in Section 6.0 of (Clifford, November 2015). An evaluation of FFRD potential for postulated LOCAs was discussed in (McCree, November 2015). Several important conclusions are written in (McCree, November 2015) including: 1) "The experimental results have continued to support the hypothesis that FFRD phenomena are primarily a high burnup fuel issue and that the current licensing limits in the U.S. are adequate to prevent dispersal of large quantities of fine fuel fragments.", 2) "The research and assessments completed to date indicate that near-term regulatory action is not needed to address FFRD phenomena at this time. However, this conclusion is closely linked with current fuel design limits and assumptions on how high-burnup fuel is operated.", 3) "Research has shown as burnup exceeds 62 GWd/MTU, fuel becomes increasingly susceptible to FFRD.", and 4) "Research and analyses provide reasonable assurance that no imminent safety concern exists with operating reactors." It is also noted in (McCree, November 2015) regarding future activities that "Given the burnup and utilization limitations on existing fuel designs, the staff does not foresee the ongoing research would identify a need to withdraw approval of existing fuel designs. However, the industry continues to develop advanced fuel designs and more economical fuel loading patterns. The research findings described above indicate that changes in fuel design and plant operations may have an adverse impact with respect to FFRD phenomena."

As noted in the prior paragraph, the conclusions regarding the safety significance of FFRD for operating plants is tied to current fuel utilization and burnup limits. **[**

]^{a,c}

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2 FUEL ASSEMBLY MECHANICAL DESIGN

2.1 INTRODUCTION

The Standard Review Plan (SRP), specifically section SRP 4.2 (US NRC, March 2007), provides the guidance for demonstrating the acceptability of a fuel design for use in-reactor. Table 1-2 of SRP 4.2 provides an overview of those parameters that need to be addressed with a new fuel design or for an increase in fuel assembly (FA) burnup limits. This section provides the criteria and justification needed to demonstrate that the fuel assembly will meet all criteria up to **[ ]**^{a,c} rod average burnup under the provisions of the incremental burnup extension. This section also provides a list of designs capable for an extension in burnup up to **[ ]**^{a,c} lead fuel rod average and a discussion of all relevant parameters listed in SRP 4.2. Some of the parameters are discussed in the other analysis sections included in this topical report. Not every criterion applies to each component; however, where applicable, the SRP section will be called out and the appropriate criteria will be shown to be met.

2.2 FUEL SYSTEM DESIGN DESCRIPTION

The fuel assembly design data for Westinghouse designed plants are given in Tables 2.2-1 to 2.2-4. The fuel assembly design data for Combustion Engineering (CE) designed plants are given in Tables 2.2-5 to 2.2-7 using various designs as examples. The fuel assembly components for other CE-designed plants are similar to those listed in referenced tables. These tables can be used to show design similarities and differences.

In addition, this topical report will also apply to the **PRIME™** fuel advanced features package which, when implemented, will further enhance fuel reliability and enable even higher burnups, higher fuel duty, and improved performance. The three **PRIME** fuel features include:

- A reinforced dashpot at the bottom of the fuel assembly skeleton (guide tube section near bottom nozzle) to further resist bow and incomplete rod insertion at higher burnups. Adding an additional tube to reinforce the dashpot area has been successfully implemented in Europe and in the United States on multiple fuel designs such as the 15x15 Upgrade fuel design currently operating in several domestic plants.
- **Low Tin ZIRLO™** mid-grids and intermediate flow mixing (IFM) grids to reduce the overall growth and oxide buildup on the grid material and improve grid-to-rod-fretting performance. This is important when considering longer cycle lengths, higher burnup, and reducing hydrogen content in the structural materials for long term storage. **Low Tin ZIRLO** structural material has the same material specification as **Optimized ZIRLO** fuel rod cladding. The final heat treatment for **Low Tin ZIRLO** structural material is full recrystallization anneal (RXA) compared to stress relieved anneal (SRA) and partial recrystallization anneal (pRXA) for **Optimized ZIRLO** fuel rod cladding. The **Low Tin ZIRLO** mid-grids exhibit similar mechanical performance (i.e., crush strength and cell stiffness behavior) as the current **ZIRLO** mid-grids. **Low tin ZIRLO** structures were approved in (Barsic et al., 2011).
- Modifications to the bottom nozzle flow hole geometry which will result in a lower pressure drop across the bottom nozzle. This has been proven in the 17 Next Generation Fuel design and is part of the **PRIME** features package. The slightly increased flow will have a positive impact on

departure from nucleate boiling (DNB) margins. The Advanced Debris Filter Bottom Nozzle (ADFBN) side skirt (designed to mitigate outer row leaking fuel rods by adding a filter to the bottom nozzle skirt flow path) is included with the modifications to the bottom nozzle flow hole geometry.

While these **PRIME** features improve fuel assembly performance at elevated burnup, they are not required for the incremental burnup extension.

Each fuel assembly consists of a square array of fuel rods, guide thimbles or guide tubes and an instrumentation tube arranged within a supporting skeleton structure. The instrumentation tube is located in the center position and provides a channel for insertion of an incore neutron detector when the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly (RCCA), or a control rod assembly (RCA), or other core components, depending on the position of the particular fuel assembly in the core.

For the purpose of illustration the 17x17 OFA fuel assembly is utilized. The 17 OFA design has been selected compared to the other 17x17 fuel designs since the 17 OFA design **[**

]^{a,c}

Figure 2.2-1 shows a 17x17 OFA fuel assembly. The fuel rods (Figure 2.2-2) are axially positioned within the fuel assembly structure to provide clearance between the fuel rod ends and the top and bottom nozzles. A more detailed view of the skeleton structure is provided in Figure 2.2-3.

Table 2.2-1 17x17 Fuel Assemblies for Westinghouse Plants, 17 OFA, 17RFA, and 17NGF Design Comparison

	17x17 OFA	17x17 RFA	17x17 NGF
[			
			][a,c]

Table 2.2-1 17x17 Fuel Assemblies for Westinghouse Plants, 17 OFA, 17RFA, and 17NGF Design Comparison (continued)

	17x17 OFA	17x17 RFA	17x17 NGF
[			
			]a,c

Table 2.2-2 17x17 Fuel Assemblies for Westinghouse Plants, 17 XL and AP1000® Plant Design Comparison		
	17x17 XL	AP1000 Plant
[		
		]a,c

Table 2.2-2 17x17 Fuel Assemblies for Westinghouse Plants, 17 XL and AP1000 Plant Design Comparison (continued)

	17x17 XL	AP1000 Plant
[		
		][a,c]

Table 2.2-3 15x15 and 16x16 Fuel Assemblies for Westinghouse Plants, 15 Upgrade and 16 NGF Design Comparison		
	15x15 Upgrade	**16x16 NGF**
[		
		]a,c

Table 2.2-3 15x15 and 16x16 Fuel Assemblies for Westinghouse Plants, 15 Upgrade and 16 NGF Design Comparison (continued)		
	15x15 Upgrade	**16x16 NGF**
[		
		]a,c

Table 2.2-4 14x14 Fuel Assemblies for Westinghouse Plants, 14 OFA and 14 V+ 17NGF Design Comparison

	14x14 OFA	14x14 422 V+
[		
		][a,c]

Table 2.2-4 14x14 Fuel Assemblies for Westinghouse Plants, 14 OFA and 14 V+ 17NGF Design Comparison (continued)

	14x14 OFA	14x14 422 V+
[		
		][a,c]

Table 2.2-5 16x16 Fuel Assemblies for Combustion Engineering Plants, Palo Verde, PV 16CE NGF, and PV 16CE STD Design Comparison

	PV 16CE NGF	PV 16CE STD
[		
		]a,c

Table 2.2-5 16x16 Fuel Assemblies for Combustion Engineering Plants, Palo Verde, PV 16CE NGF and PV 16CE STD Design Comparison (continued)

	PV 16CE NGF	PV 16CE STD
[		
		]a,c

Table 2.2-5 16x16 Fuel Assemblies for Combustion Engineering Plants, Palo Verde, PV 16CE NGF and PV 16CE STD Design Comparison (continued)

	PV 16CE NGF	PV 16CE STD
[		
		]a,c

Table 2.2-6 16x16 Fuel Assemblies for Combustion Engineering Plants, 16CE STD and 16CE NGF Design Comparison		
	WTFD 16CE STD	**WTFD 16CE NGF**
[		
		][a,c]

Table 2.2-6 16x16 Fuel Assemblies for Combustion Engineering Plants, 16CE STD and 16CE NGF Design Comparison (continued)

	WTFD 16CE STD	WTFD 16CE NGF
[		
		]a,c

Table 2.2-7 16x16 Fuel Assemblies for Combustion Engineering Plants, 16CE NGF Design

	ANO2 16CE NGF
[	
	]^{a,c}

Table 2.2-7 16x16 Fuel Assemblies for Combustion Engineering Plants, 16CE NGF Design (continued)

	ANO2 16CE NGF
[	
	][a,c]

Note: All dimensions are in inches and are listed as reference values as would be done in a plant's FSAR.

a,c

Figure 2.2-1: Typical Fuel Assembly 17x17 OFA Design

a,c

Figure 2.2-2: Typical 17x17 OFA Fuel Rod

a,c

Figure 2.2-3: Typical Fuel Assembly 17x17 OFA Skeleton

2.3 17X17 OFA FUEL ASSEMBLY

The design bases for the 17x17 OFA fuel assembly and each of the assembly components are identical to those for the VANTAGE + design as given in (Davidson et al., April 1995). The 17x17 OFA fuel assembly was designed (i.e., sized) to achieve up to a **[ ]**^{a,c} lead fuel rod average burnup, consistent with the methodology in both (Davidson et al., December 1985) and (Foster and Sidener, July 2000), with an option to use **Optimized ZIRLO** cladding (Schueren, July 2006). Various features of the 17x17 OFA design have a licensing pedigree as noted below:

• 0.360 inch outer diameter fuel rods	(Davidson, S. L., February 1989)
• Axial blanket pellets (including annular blankets)	(Davidson and Kramer, September 1985) and (Davidson et al., April 1995)
• Three Intermediate Flow Mixing grids	(Davidson and Kramer, September 1985)
• High burnup Bottom Grid	(Davidson and Kramer, September 1985)
• Debris Filter Bottom Nozzle (DFBN)	(Davidson and Kramer, September 1985)
• Protective Grid (P-grid)	Implemented under 10 CFR 50.59 prior to FCEP
• Option for Oxide coating for mitigation of debris related failures	Implemented under 10 CFR 50.59 prior to FCEP (Some plants already employ the oxide coated cladding feature. Thus, it may be new for some plants.)
• Option for **Optimized ZIRLO** fuel rod cladding	(Schueren, July 2006)

2.3.1 Fuel Assembly Design Bases and Evaluations

2.3.1.1 Fuel Assembly Growth

Design Basis: Axial clearance between core plates and nozzle end plates should allow sufficient margin for fuel assembly and fuel rod irradiation growth to established design burnups.

Evaluation: This criterion assures that excessive forces on a fuel assembly will not be generated by the hard contact between the top nozzle and the upper core plate. Such forces could lead to fuel assembly bowing or guide thimble distortion. As such, the fuel assembly is typically sized to support growth up to a **[ ]**^{a,c} lead fuel rod average burnup, even though the design will only be licensed to **[ ]**^{a,c} lead fuel rod average burnup. Sizing accounts for the irradiation growth behavior of material used.

2.3.1.2 Fuel Assembly Hydraulic Stability

Design Basis: Flow through the assembly should not cause wear that exceeds the Westinghouse guideline of **[ ]**^{a,c}.

Evaluation: The 17x17 OFA fuel assembly has been flow tested in the Westinghouse VIPER Loop adjacent to another 17x17 OFA fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and IFM-grids is well within the Westinghouse guideline of limiting wear to less than [ ]a,c. Testing was also performed using the Westinghouse FACTS Loop to confirm the pressure drop characteristics across the entire assembly and individual components as well as verifying that the fuel assembly vibration amplitude was less than [

]a,c. Testing for resonant high frequency vibration of the mid-grid and IFM-grid straps was conducted using the Westinghouse VISTA Loop with no significant vibration response found.

2.3.1.3 Fuel Assembly Structural Integrity

Design Basis: The fuel assembly must maintain its structural integrity in response to seismic and loss-of-coolant accident (LOCA) loads.

Evaluation: Testing and analysis were performed for the fuel assembly at beginning-of life (BOL) conditions to verify that structural integrity would be maintained during seismic and LOCA loads. Dynamic crush testing of the mid-grids and IFM-grids at BOL conditions found an acceptable mid-grid structural performance.

The impact of the burnup extension on fuel assembly mechanical characteristics and seismic / LOCA analysis results at the end of life (EOL) conditions is also performed to address Information Notice (IN) 2012-09 (US NRC, June 2012) subsequently issued by the Nuclear Regulatory Commission (NRC) in 2012. Guidelines for this evaluation are provided in the Westinghouse and Pressurized Water Reactor Owners Group (PWROG) topical report (Lu and Jiang, November 2019).

The 17x17 OFA EOL gap sizes between the grid cells and fuel rods were calculated for incremental high burnup at EOL conditions, and it was found that the [

]a,c. According to (Lu and Jiang, November 2019), the 17x17 OFA EOL seismic and LOCA analysis demo case is performed following the process procedures. The demo case shows that the fuel assembly structural integrity at EOL would be maintained during seismic and LOCA loads. A plant-specific EOL seismic and LOCA analysis will be performed following the process given in the demo case for other fuel designs during implementation of the incremental burnup extension.

2.3.1.4 Fuel Assembly Shipping and Handling Loads

Design Basis: The design acceleration limit for the fuel assembly handling and shipping loads is a minimum of 4g's.

Evaluation: Testing and analysis were performed on the fuel assembly to verify that shipping and handling load requirements were met. Section 2.4 gives more detail on what was done for the different components.

2.3.1.5 Fuel Assembly Bow and RCCA Insertion

Design Basis: The guide thimbles provide channels for the insertion of either a RCCA or a RCA, and provide an insertion path with distortion limited so that in the event of a SCRAM, the RCCA drop time limits established by the overall reactor plant design basis are met and the component fully inserts.

Evaluation: This criterion assures that guide thimble bow does not prevent RCCA insertion such that in the event of a SCRAM, the RCCA drop time limits are met. The overall guide thimble bow corresponding to fuel assembly bow and local guide thimble bow create a mechanical resistance force which reduces RCCA drop time and ultimately could lead to Incomplete Rod Insertion (IRI). The mechanical resistance force effect to RCCA travel is characterized by RCCA drag work. The RCCA Drag Work limit has been established based on the available RCCA drop tests and RCCA drag force measurement. This limit does not depend on the fuel assembly burnup.

The fuel assembly bow depends on fuel assembly design, fuel management, and operating conditions.

The current Westinghouse fuel assembly designs have an acceptable fuel assembly bow as demonstrated by the favorable 20 + year RCCA insertion history. A combination of fuel assembly, RCCA and drive line designs is found to be adequate to meet the specification requirement for RCCA drop times and insertion. Thus, the current fuel assembly designs are capable of supporting the proposed burnup increase.

The RCCA drop time has been found to be acceptable for the current fuel operating conditions based on the favorable 20 + year RCCA insertion history. The plant uprates, steam generator replacements, and other plant modifications affecting the fuel operating conditions have been found to be acceptable and do not prevent meeting RCCA drop times and insertion.

The elimination of high burnup fuel restriction for IRI was previously discussed in (Aleshin, October 2013). However, the fuel management corresponding to the proposed burnup increase leads to a reduction in number of feed assemblies and an average fuel assembly burnup increase. As a result, RCCA insertion could be potentially affected and needs to be evaluated. To demonstrate that the current fuel assembly design can support increased burnup management, an assessment has been completed to assess RCCA drag force variation for the current and high burnup fuel management. The assessment has been performed for representative 17x17 fuel designs (17x17 OFA and 17x17 RFA).

Based on the assessment results for the current hardware configuration, it was found that the RCCA drag work limit will not be exceeded, and therefore, the RCCA insertion requirement will be met. For other Westinghouse and Combustion Engineering fuel designs, similar assessments will be completed (as needed) before implementation of the burnup increase.

Finally, the current fuel assembly designs should be able to meet RCCA drop time and insertion limits for fuel management for the lead fuel rod average burnup increase up to **[** **]**[a,c]

2.3.1.6 Fuel Rod Bow

Design Basis: The spacer grids shall not permit or cause rod bowing that exceeds the allowable limits for channel closure for the fuel assembly lifetime. In the case of IFM or protective grids, the rod bowing in any span shall not be increased compared to fuel that does not incorporate those types of grids.

Evaluation: Although some high magnitudes of channel closure due to rod bow were observed in the past, subsequent data indicates that this bow has been reduced with the implementation of **ZIRLO** grids and cladding and the use of reduced rod bow spring forces. The trend of rod bow with burnup does not increase a penalty on DNB margin at high burnup. The maximum rod bow penalty is determined at a fuel assembly burnup typically below 33,000 MWd/MTU. Beyond that burnup, a credit is taken for the fuel rod to be at sufficiently low power that it can no longer be DNB limiting. Thus, for the fuel rod average burnup extension to **[** **]**[a,c], margins to DNB are not impacted.

2.4 STRUCTURAL COMPONENTS DESIGN BASES AND EVALUATIONS

2.4.1 Bottom Nozzle

The Bottom Nozzle has the following functional requirements:

1. Provides for Positioning the Bottom of the Fuel Assembly
2. Provides for Positioning of Fuel Rod in Elevation
3. Provides for Distribution of Flow of Reactor Coolant
4. Provides for Fuel Assembly Handling
5. Provides for Positioning and Guiding In-Core Instrumentation in plants with bottom inserted instrumentation
6. Provides for Protecting Fuel Rods from Debris

The 17x17 OFA DFBN was designed for 4g shipping and handling loads. A typical 17x17 OFA bottom nozzle is shown in Figures 2.4-1 and 2.4-2.

Design Basis: The bottom nozzle design bases are the same as those given in Section 2.3.1.2 of (Davidson and Kramer, September 1985).

Evaluation: Confirmatory testing was performed to verify the load-deflection characteristics and the flatness of the 17x17 OFA DFBN under 4g loading conditions. The test results met the design requirements.

2.4.2 Top Nozzle

The Top Nozzle has the following functional requirements:

1. Provides for Positioning of the Top of the Fuel Assembly
2. Provides Vertical Holddown of Fuel Assembly
3. Provides for Distribution of Coolant Flow
4. Provides for Handling of the Fuel Assembly
5. Prevents Ejection of Fuel Rod
6. Provides a Means of Reconstitution of Fuel Assembly

The top nozzle assembly design for 17x17 OFA is the reconstitutable top nozzle (RTN) type (split insert with lock tube) joint connection for attachment to the skeleton assembly. A typical Top Nozzle is shown in Figure 2.4-3. A view of the Top Nozzle to skeleton attachment is shown in Figure 2.4-4.

Design Basis: The top nozzle design bases are the same as those given in Section 2.3.2.2 of (Davidson and Kramer, September 1985).

Evaluation: Shipping and handling loads for the 17 OFA top nozzle have been analyzed using finite element analysis (FEA) methods and the nozzle has been functionally tested. The top nozzle design was shown to meet all requirements.

2.4.3 Fuel Assembly Holddown Springs

All 17x17 OFA hold-down spring packs use leaves similar to the spring leaves used on the standard 15x15 and 17x17, 12-foot fuel assemblies.

Design Basis: The design bases for the holddown springs are the same as those given in Section 2.3.3.2 of (Davidson and Kramer, September 1985).

Evaluation: Hydraulic tests were performed to obtain the necessary inputs to determine the required holddown force. Load deflection testing of the spring packs was completed to determine the actual spring load deflection characteristics. A final verification analysis using standard Westinghouse methodology will be performed for plant specific requirements to verify that holddown requirements are met.

2.4.4 Guide Thimbles and Instrumentation Tube

The Guide Thimbles have the following function:

1. Provides for Structural Continuity of the Fuel Assembly Skeleton

2. Provides for Positioning the Rods of Core Components
3. Provides for Flow of Reactor Coolant Around the Core Component Rods
4. Provides for RCCA Insertion Motion and Dashpot Action

The Instrumentation Tube has these functions:

1. Provides for Positioning the Sensor of the Core Instrumentation
2. Provides for Flow of Reactor Coolant Around the Core Instrumentation

The 17x17 OFA guide thimble is an integral swaged thimble design. The integral swaged guide thimble assembly has two diameters: a larger diameter which extends over most of the length of the fuel assembly, and a smaller diameter which forms the dashpot region at the bottom of the thimble.

The instrumentation tube is fabricated with the same OD and ID as the larger diameter section of the guide thimble tube. The length of the 17x17 OFA instrumentation tube is larger than the guide thimble tube length because the guide thimble ends below the top nozzle; whereas, the instrumentation tube has to engage the top nozzle above the counter bore chamfer.

Design Basis: The general guide thimble and instrumentation tube design bases are the same as those given in Section 2.3.4.2 of (Davidson and Kramer, September 1985).

Evaluation: Stress analysis on the 17x17 OFA guide thimble tube show adequate margin on shipping and handling loads.

2.4.5 Joints and Connections

The 17x17 OFA design uses joints and connections that are similar to existing Westinghouse designs. The top nozzle to thimble joint uses a reconstitutable insert. The top-grid, bottom-grid, mid-grids, and IFM-grids are all bulged to the skeleton. The protective grid is joined using a spacer or insert. The bottom nozzle is connected to each of the thimble tubes using a high strength thimble screw.

Design Basis: For events expected during the life of the fuel assembly, the resulting Condition I and II loads shall not cause permanent deformation at the joints or connections nor prevent the continued use of the fuel assembly for its design life. For accident and unanticipated events, the resulting Condition III and IV loads shall not cause any deformations that would prevent emergency cooling of the fuel or prevent the safe shutdown of the reactor. In addition, the loads resulting from shipping and handling shall not cause any deformations that would prevent the fuel assembly from meeting all the operating requirements for its design life.

Evaluations: Confirmatory testing was completed to verify the integrity of the joints and connections during the life of the fuel assembly for any accident and unanticipated events and for any loads resulting from shipping and handling.

2.4.6 Grid Assemblies

The various spacer grid assemblies have the following functions:

1. Provide Fuel Rod Support
2. Maintain Fuel Rod Spacing
3. Form Part of the Fuel Assembly Skeleton Structure
4. Promote Mixing of the Coolant
5. Provide Lateral Support and Positioning for the RCCA Guide Thimbles
6. Provide Lateral Support and Positioning for the Instrumentation Tube
7. Prevent Damage During Handling Operations
8. Protect the Fuel Rods from Foreign Material in the Flow Stream (Protective grid)

Top and Bottom-Grids

The top (Figure 2.4-5) and bottom (Figure 2.4-6) grids, used on the 17x17 OFA design have **[ ]**^{a,c} straps. The grid provides 6-point rod support (2 vertical springs and 2 vertical dimple pairs per cell). Sleeves are brazed into the grid at thimble cell locations and these sleeves are then bulged to the thimble tube to fix the location of the grid.

Mid-Grid

The 17x17 OFA mid-grid design has a 6-point rod support system (2 vertical springs and 2 horizontal dimple pairs per cell). The 17x17 OFA design uses **[ ]**^{a,c} straps; the material is laser welded and has skeleton mid-grid attachment sleeves. Refer to Figure 2.4-7.

Intermediate Flow Mixers (IFM)

The 17x17 OFA IFM design provides additional coolant turbulence in the high temperature spans near the top of the fuel assembly. The 17x17 OFA design uses **[ ]**^{a,c} straps; the material is laser welded and has skeleton IFM-grid attachment sleeves. The 17 OFA IFM application places one IFM near the mid-point of the upper three mid-grid to mid-grid spans. Refer to Figure 2.4-8.

Protective-Grid (P-grid)

The protective grid is a welded grid with **[ ]**^{a,c} straps. The protective grid supports the fuel rod with four coplanar dimples in each cell. The primary function of the protective grid is debris mitigation. The protective grid accomplishes this by sectioning the flow holes in the bottom nozzle, stopping debris before it can reach the fuel rod cladding. Refer to Figures 2.4-9 and 2.4-10.

Design Basis: The grid design bases are the same as those given in Section 2.3.5.2 of (Davidson and Kramer, September 1985).

The grids must function acceptably under loading limits and not fail due to fatigue. In addition, the interaction between the grid and fuel rod should not result in conditions beyond the allowable fretting wear guidelines.

Evaluation: The evaluation of the 17x17 OFA grids is based on the extensive design and irradiation experience with previous grid designs and the component testing and analysis completed with the 17x17 OFA design.

Fatigue testing and analysis was satisfactorily completed for the rod support features.

The 17x17 OFA fuel assembly has been flow tested in the Westinghouse VIPER Loop adjacent to another 17x17 OFA fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and IFM-grids is well within the Westinghouse guideline of limiting wear to less than **[ ]**^{a,c}.

Specific grid-to-rod fretting (GTRF) wear performance for this incremental burnup limit extension was evaluated on the basis of fuel rod vibration, fuel assembly vibration, the wear couples of fuel rod-to-grid supports, resident time, and field performance. The conclusion is that increasing burnup to **[ ]**^{a,c} should not impact GTRF wear.

a,c

Figure 2.4-1: Typical 17 OFA Bottom Nozzle

a,c

Figure 2.4-2: 17 OFA Connection of Guide Tubes to Bottom Nozzle

a,c

Figure 2.4-3: Typical 17 OFA Top Nozzle Design

a,c

Figure 2.4-4: Connection of Guide Tubes to Top Nozzle

a,c

Figure 2.4-5: Top End Grid and Cutaway Showing Fuel Rod Interface

a,c

Figure 2.4-6: 17 OFA Bottom Grid

a,c

Figure 2.4-7: 17 OFA Mid-Grid and Mid-Grid Cutaway Section

a,c

Figure 2.4-8: Intermediate Flow Mixing Grid and Cutaway Showing Non-Contacting Grid Cells

a,c

Figure 2.4-9: Combination Grid with Bottom End Grid and Protective Grid and Cutaway showing Details of Fuel Rod Interface

a,c

Figure 2.4-10: Protective Grid

2.5 MATERIALS

2.5.1 Background

The extension of fuel rod average burnup limits from **[** **]**^{a,c} can potentially cause more corrosion, more hydrogen uptake, more irradiation growth of zirconium-based materials, and more fission gas release from the fuel. Some fuel assemblies have reached assembly burnup above 62 GWd/MTU as listed in Table 2.5-1. Table A-1 in Appendix A of this topical report shows a summary of the post irradiation exam (PIE) program with fuel rod burnup ≥ 62 GWd/MTU. This section of the topical report summarizes all the PIE data from these programs and shows the impacts on the fuel performance with the burnup extension.

2.5.2 Fuel Rod Materials

Design Basis: The fuel rod design will use design values for properties of materials as given in (Davidson et al., April 1995), (Davidson and Kramer, March 1986), (Davidson et al., October 1994), (Schueren, July 2006), and (Ewing and Smith, January 1983 / Rossi, September 1986) for **ZIRLO** cladding, **Optimized ZIRLO** cladding, IFBA, and Gadolina material.

Evaluation: The material properties of the UO_2 fuel are not affected by the presence of a thin [ ]^{a,c} ZrB_2 coating on the fuel pellet surface; therefore, the properties described in (Davidson et al., October 1994) for UO_2 are also applicable, with due consideration to temperature and irradiation effects. In the ZrB_2 rod, the fuel pellets [ ]^{a,c} in ^{10}B. The ^{10}B acts as a burnable absorber. The irradiation behavior of the thin IFBA coating material has been reviewed and approved for use in Westinghouse pressurized water reactors (PWRs) in (Davidson and Kramer, March 1986).

Some material properties of the UO_2 fuel are slightly affected by the presence of Gadolinia in the fuel matrix, while other material properties are negligibly impacted. (Ewing and Smith, January 1983 / Rossi, September 1986) describes the appropriate material properties for Gd_2O_3 in UO_2, with due consideration to temperature and irradiation effects. In the Gadolinia fuel rod, a small amount of Gd_2O_3 is mixed with the UO_2 and sintered together to act as a burnable absorber. The use of Gadolinia product has been reviewed and approved for use in Westinghouse PWRs in (Davidson et al., October 1994) and (Ewing and Smith, January 1983 / Rossi, September 1986).

ZIRLO alloy is a modification of the Zircaloy-4 alloy. The comparative properties of the **ZIRLO** and Zircaloy-4 alloy are described in detail in (Davidson et al., April 1995). Some of these properties, including density, thermal expansion, thermal conductivity, and specific heat have been verified in testing programs described therein. **[**

]^{a,c}

Optimized ZIRLO alloy is a modification of the **ZIRLO** alloy. The comparative properties of the **ZIRLO** and **Optimized ZIRLO** alloy are described in detail in (Schueren, July 2006). Some of these properties, including density, thermal expansion, thermal conductivity, and specific heat have been verified in testing programs described therein.

2.5.3 Vogtle Creep and Growth

In order to get high quality creep and growth data on fuel rod cladding to high fluences without interference from pellet cladding interaction, the Vogtle creep and growth program was implemented. The reason for this program was to gather data to produce improved cladding creep and irradiation growth models and to justify that creep in and creep out are equivalent. To gather this data the [

]^{a,c} The pressure differentials provided variations in hoop stresses to allow measurements of diametrical creep under controlled levels of hoop stresses. The free irradiation growth was also measured on [
]^{a,c}

The materials included were **ZIRLO** cladding and **Optimized ZIRLO** cladding. Table 2.5-2 is a summary of the fluence and its approximate equivalent burnup [
]^{a,c} Figure 2.5-1 shows a selection of irradiation creep diameter strain versus deviatoric hoop stress of **ZIRLO** cladding. [

]^{a,c} The response to request for additional information (RAI) 9e in (Bowman, et al., 2017) includes comparisons of model predictions to measurements at the higher fluences. Predictions are in good agreement with measurements.

Figure 2.5-2 shows that the diametral irradiation growth results for both **ZIRLO** and **Optimized ZIRLO** cladding. The [
]^{a,c}

2.5.4 Rod Oxide Thickness

The oxide thickness data for fuel rod average burnup ≥ 62 GWd/MTU is plotted in Figure 2.5-3, which is taken from the corrosion database for **ZIRLO** and **Optimized ZIRLO** cladding materials. [
]^{a,c} 100 microns, which is the design limit for the cladding corrosion criteria (Bowman, et al., 2017). The oxide thickness is well below 100 microns for the **Optimized ZIRLO** cladding up to a burnup of [ ]^{a,c}

2.5.5 Rod Growth

The single rod growth data for fuel rod average burnup ≥ 62 GWd/MTU is plotted in Figure 2.5-4, which is also taken from the **Optimized ZIRLO** cladding database. Figure 2.5-4a shows the rod growth versus burnup and Figure 2.5-4b shows the same data versus fluence including the best estimate and the lower and upper bound **ZIRLO** and **Optimized ZIRLO** cladding models. Though there is [

]^{a,c} for the corresponding cladding materials.

2.5.6 Hydrogen

The hydrogen content is plotted in Figure 2.5-5 for **ZIRLO** and **Optimized ZIRLO** cladding. Figure 2.5-5a is the plot of hydrogen content versus the oxide thickness, while Figure 2.5-5b is the plot of hydrogen content versus the rod average burnup. The **[ ]**[a,c] for **ZIRLO** and **Optimized ZIRLO** cladding. The hydrogen contents for **Optimized ZIRLO** cladding above 62 GWd/MTU are **[ ]**[a,c] because **ZIRLO** cladding has poorer corrosion resistance (higher oxide thickness) than **Optimized ZIRLO** cladding.

2.5.7 Mechanical Properties

The yield and ultimate stress of **ZIRLO** and **Optimized ZIRLO** cladding versus the rod average burnup are present in Figure 2.5-6a and 2.5-6b. The uniform plastic strain and the total plastic strain versus the rod average burnup are plotted in Figure 2.5-7a and 2.5-7b. These data were obtained by different test methods **[ ]**[a,c] The materials were from different lots and irradiated at different reactors; therefore, **[ ]**[a,c]. The uniform and the total plastic strain data do not exhibit any obvious trends before or after 62 GWd/MTU.

2.5.8 Grids

The oxide thickness vs. burnup for both **Low Tin ZIRLO** (LTZ) and **ZIRLO** grids is plotted in Figure 2.5-8. **ZIRLO** and **Low Tin ZIRLO** structural material have the same material specification as **ZIRLO** and **Optimized ZIRLO** fuel rod cladding, respectively. The final heat treatment for **ZIRLO** and **Low Tin ZIRLO** structural material is RXA compared to SRA and pRXA for **ZIRLO** and **Optimized ZIRLO** fuel rod cladding, respectively.

The highest burnup is about **[ ]**[a,c]. From the hotcell examination, the maximum observed two-sided oxide thickness is **[ ]**[a,c], which was irradiated at V. C. Summer. From the poolside examination, the maximum observed two-sided oxide thickness is **[ ]**[a,c], which was also irradiated at V. C. Summer. The corresponding grid metal wastage from those oxide thicknesses is below the 18% design limit (Bowman, et al., 2017).

2.5.9 Assembly Growth

The assembly growth PIE data for the assembly burnup ≥ 56 GWd/MTU up to **[ ]**[a,c] is plotted in Figure 2.5-9. The **ZIRLO** cladding fuel assembly growth models are also presented in Figure 2.5-9 for comparison. The data are consistent with the **ZIRLO** cladding models.

2.5.10 Summary

The PIE data for average fuel rods burnup greater than or equal to 62 GWd/MTU is summarized in this section.

1. **[**
   ]^{a,c}
2. All rod growth data for **ZIRLO** and **Optimized ZIRLO** cladding are bounded by the upper limits of the cladding growth model.
3. There are no obvious changes in mechanical properties for **ZIRLO** and **Optimized ZIRLO** cladding before and after 62 GWd/MTU.
4. The available PIE oxide thickness data in grids meet the design limit **[**
   ]^{a,c}
5. The assembly growth PIE data are consistent with the **ZIRLO** cladding models up to **[**
   ]^{a,c}

Table 2.5-1 Number of Fuel Assemblies with Greater than 60 GWd/MTU Assembly Burnup as of December 2018

[		
		]a,c

Table 2.5-2 Vogtle Creep and Growth Program

[		
		]a,c

a,c

Figure 2.5-1: ZIRLO Cladding Irradiation Creep Diameter Strain/Deviatoric Hoop Stress as a Function of Fluence

a,c

Figure 2.5-2: Irradiation Growth in the Diametral Direction for both ZIRLO and Optimized ZIRLO Cladding

a,c

Figure 2.5-3: Maximum Measured Oxide Thickness versus Fuel Average Burnup at Burnup ≥ 62 GWd/MTU

a,c

Figure 2.5-4: Fuel Rod Growth for Fuel Average Burnup ≥ 62 GWd/MTU
(a) Growth versus Burnup, (b) Growth versus Fluence

a,c

Figure 2.5-5: Hydrogen Content of ZIRLO and Optimized ZIRLO Cladding versus (a) Oxide Thickness, (b) Rod Average Burnup

a,c

Figure 2.5-6: (a) Yield Stress Data, (b) Ultimate Stress Data of Irradiated Optimized ZIRLO Cladding (OZ) and ZIRLO Cladding (Z) at Temperature above 300°C

a,c

Figure 2.5-7: (a) Uniform Plastic Strain (b) Total Plastic Strain of Irradiated Optimized ZIRLO Cladding and ZIRLO Cladding at Temperature above 300°C

a,c

Figure 2.5-8: Two-Sided Oxide Thickness for Grids

a,c

Figure 2.5-9: Assembly Growth versus Assembly Burnup ≥ 56 GWd/MTU

2.6 REFERENCES

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3 CORE AND FUEL ROD PERFORMANCE

3.1 FUEL ROD PERFORMANCE

Fuel performance analyses will be performed using PAD5, the most recent Nuclear Regulatory Commission (NRC)-approved Westinghouse Fuel Performance and Design Model, documented in (Bowman et al., November 2017). Corrosion and hydrogen pick-up evaluations will be performed using the NRC-approved Westinghouse Clad Corrosion Model for **ZIRLO** and **Optimized ZIRLO** cladding (Garde et al., October 2013). In the Safety Evaluation (SE) for the PAD5 topical report, the NRC included the following Conditions and Limitations that would require modification to accommodate a burnup extension to a rod average burnup of **[** **]**^{a,c}:

> Rod average burnups up to 62 GWd/MTU for all approved types of cladding.

Upon submittal of this topical report, the Condition and Limitation noted above will be requested as follows:

> Rod average burnups up to **[** **]**^{a,c} for all approved types of cladding.

PAD5 considered rod average burnups above **[** **]**^{a,c}. Fuel performance data for rod average burnups beyond **[** **]**^{a,c} were used in the calibration and validation of the models in the PAD5 code. Various fuel and cladding material phenomena at high burnup have already been considered in PAD5, specifically:

<u>Rim structure and its impact on FGR</u>

PAD5 includes both thermal and athermal fission gas release models. The athermal release model includes an enhanced release component for higher burnups that models release from the rim region of the fuel pellet. Figures A.2.2-5, A.2.2-7, A.2.2-9, and A.2.2-11 of (Bowman et al., November 2017) show that the PAD5 fission gas release model **[** **]**^{a,c}

<u>Continuing degradation of fuel thermal conductivity</u>

Fuel temperature data for burnups greater than **[** **]**^{a,c} were used in development of the model. Figure A.2.1-15 of (Bowman et al., November 2017) shows a comparison predicted minus measured results versus burnup for fuel temperatures using the PAD5 thermal model.

<u>Potentially enhanced rod growth</u>

[

]^{a,c} as shown in Figure 5.9-3 of (Bowman et al., November 2017). Figure 2.5-4 of this report shows the rod growth model and its comparison to high burnup data.

Potentially enhanced clad corrosion

The approved corrosion model for **ZIRLO** and **Optimized ZIRLO** cladding materials is documented in (Garde et al., October 2013). The corrosion database for **ZIRLO** and **Optimized ZIRLO** contains data for rod average burnups in the range from 62 GWd/MTU to approximately **[ ]**[a,c], as shown in Section 2.5.4 of this report. As can be seen in Garde et al., October 2013, the **[ ]**[a,c] This confirms that no new mechanisms of accelerated corrosion appear in this range, beyond what is captured in the model. It is noted that the corrosion model uses thermal reaction accumulated duty (TRD) to calculate corrosion. Figure 2.1-3 of (Garde et al., October 2013) shows the relationship between rod average burnup and TRD for a Lead Test Assembly (LTA).

Enhanced fission gas swelling

A model for enhanced fission gas swelling at high burnup is developed in Section 5.8 of (Bowman et al., 2017). Figures A.2.4-1 to Figure A.2.4-4 in Appendix A of (Bowman et al., November 2017) show that the model enables the excellent prediction of the ramp diameter results up to very high burnup.

Continuing degradation of mechanical strength due to higher hydrogen pickup

The fuel rod design procedures ensure that the hydrogen concentration is limited to the **[ ]**[a,c] design limit. It is expected that operation at higher burnup will result in some increase in cladding corrosion and, therefore, in hydrogen content. However, confirmation of this criterion on a plant-specific basis as part of the standard reload analysis will ensure that degradation of mechanical strength due to higher hydrogen pickup is not a concern.

It should be noted that the **[**

][a,c] Section 2.5.3 of this report contains more information on these data. High burnup cladding diameter data included in Figures A.2.4-4 and A.2.5-2 of (Bowman et al., November 2017) show that PAD5 predicts **[ ]**[a,c]

In summary, the thermal, corrosion, pellet deformation, cladding deformation, and fission gas release models in PAD5 all **[**

][a,c]

The methods described in (Bowman et al., November 2017) do not need to be modified for rod average burnups up to [ ]^{a,c}. The methods defining analysis interfaces remain the same as described in (Bowman et al., November 2017), with no additional data required for fuel performance analyses nor any additional data transmitted for transient analyses. Definitions for limiting cases and uncertainty development methods are unaffected by increasing the allowable rod average burnup to [ ]^{a,c}, and an assessment of required uncertainties in analyses has been conducted.

It is not necessary to revise the PAD5 topical report (Bowman et al., November 2017) to incorporate this revised limitation and condition since this limitation and condition will be reflected in this topical report, which will be incorporated into each plant's licensing basis. Additional details are discussed in the following sections. This discussion does not impose a limitation or restriction on the use of future NRC-approved fuel performance methods in place of PAD5 or supplements to PAD5.

The fuel rod design bases and criteria are described below. Those which are particularly affected by the increase in rod average burnup to [ ]^{a,c} are so noted in the following sections.

3.1.1 Fuel Rod Internal Pressure

Design basis and acceptance limit: The fuel system will not be damaged due to excessive fuel rod internal pressure. The internal pressure of the lead fuel rod in the reactor should be limited to a value below that which could (1) cause the diametral gap to increase due to outward clad creep during steady-state operation, (2) result in cladding hydride reorientation in the radial direction, and (3) lead to extensive departure from nucleate boiling (DNB) propagation to occur.

Evaluation: Fuel rod internal pressure is evaluated using the NRC-approved fuel rod performance code PAD5 (Bowman et al., November 2017). The evaluation of this criterion [

]^{a,c}

As described above, the PAD5 code and associated methodology conservatively account for all the high burnup phenomena that contribute to the rod internal pressure. It is expected that operation to higher burnups will result in higher rod internal pressures. This criterion is evaluated on a plant-specific basis as part of the standard reload analysis following fuel rod design procedures which ensure that all three acceptance limits mentioned previously are met. [

]^{a,c} which is evaluated on a cycle-specific basis. The criterion evaluation procedure for rod internal pressure, addressing all three acceptance limits mentioned previously, is described in Section 7.4.3 of (Bowman et al., November 2017).

3.1.2 Fuel Rod Clad Stress

Design basis and acceptance limit: The fuel system will not be damaged due to excessive fuel rod clad stress. The maximum cladding stress intensities, excluding pellet-cladding interaction (PCI)-induced stress, will be evaluated based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) guidelines.

Evaluation: This evaluation uses **[**

]^{a,c} The PAD5 code and methods provide adequate and conservative estimates for these quantities. The increased burnup will lead to some increase in rod internal pressure and corrosion; the latter will, in turn, result in an increase in cladding temperatures. Rod internal pressure is discussed in Section 3.1.1. Corrosion is discussed in Section 3.1.5. It is anticipated that the overall effect of the operation at higher rod average burnup **[**

]^{a,c} This will be confirmed on a plant-specific basis, as the stress criterion is evaluated as part of the standard reload analysis. The criterion evaluation procedure for cladding stress is described in Section 7.4.1 of (Bowman et al., November 2017).

3.1.3 Fuel Rod Clad Strain

Design basis and acceptance limit: The fuel rod will not fail due to excessive fuel clad strain. The design limit for the fuel rod clad strain is that the total tensile strain, elastic plus plastic, due to uniform cylindrical fuel pellet deformation during any single Condition I or II transient should be less than 1% from the pre-transient value.

Evaluation: Cladding strains are evaluated using the NRC-approved fuel rod performance code PAD5 (Bowman et al., November 2017). The cladding strain analysis **[**

]^{a,c} This will be confirmed on a plant-specific basis, as the cladding strain criterion is evaluated as part of the standard reload analysis. The criterion evaluation procedure for clad strain is described in Section 7.4.2 of (Bowman et al., November 2017).

3.1.4 Pellet-Cladding Interaction

Design basis and acceptance limit: The fuel rod will not fail due to PCI or pellet-cladding mechanical interaction (PCMI). There is no specific design criterion for PCI. Two related criteria, the one percent clad strain criterion and the fuel overheating criterion, must be met.

Evaluation: As noted in Sections 3.1.3 and 3.1.6, cladding strain and fuel overheating criteria will be confirmed on a plant-specific basis as part of the standard reload analysis to ensure that they continue to be met at high burnups.

3.1.5 Fuel Clad Oxidation and Hydriding

Design basis and acceptance limit: Fuel damage will not occur due to excessive clad oxidation and hydriding. For **ZIRLO** and **Optimized ZIRLO** cladding materials, the acceptance limit for cladding oxidation is that **[ ]^{a,c}** shall be no greater than 100 microns. Furthermore, the maximum TRDs are restricted to numbers corresponding to a cladding corrosion amount of 100 microns for licensing applications. The acceptance limit for cladding hydrogen pickup is that the best estimate, volume average hydrogen pickup level in the most limiting clad axial node will be less than or equal to **[ ]^{a,c}** at the end of fuel operation.

Evaluation: Fuel clad corrosion and hydriding are evaluated using the NRC-approved Westinghouse clad corrosion model for **ZIRLO** and **Optimized ZIRLO** cladding (Garde et al., October 2013). The clad corrosion and hydriding databases contain high-burnup data and continue to be updated with data from healthy fuel examinations and high-burnup programs, as discussed in Sections 2.5.4 and 2.5.6 of this report. The longer resident time associated with operation to higher burnup will increase the level of cladding corrosion and hydriding. **[ ]^{a,c}** This will be confirmed on a plant-specific basis using the NRC-approved Westinghouse clad corrosion model for **ZIRLO** and **Optimized ZIRLO** cladding (Garde et al., October 2013), as the corrosion and hydriding criteria are evaluated as part of the standard reload analysis. The criterion evaluation procedures for cladding corrosion and hydriding when utilizing PAD5 are described in Sections 7.4.5 and 7.4.6 of (Bowman et al., November 2017), respectively.

3.1.6 Fuel Temperature

Design basis and acceptance limit: The fuel rods will not fail due to fuel centerline melting for Condition I and Condition II events. The fuel rod centerline temperature shall not exceed the fuel melt temperature during Condition I and II operation, accounting for degradation of the melt temperature due to burnup and the addition of integral burnable absorbers.

Evaluation: Fuel temperatures are evaluated using the NRC-approved fuel rod performance code PAD5 (Bowman et al., November 2017). The PAD5 fuel temperature model incorporates the effects of fuel thermal conductivity degradation with burnup. The PAD5 thermal model calibration and validation database contains a considerable amount of data in the rod average burnup range from 62 to **[ ]^{a,c}**. The validation results in this burnup range show very good agreement, which indicates that the key phenomena have been adequately captured in the code models. The operation to higher burnups will impact several aspects related to the power-to-melt evaluation: **[ ]^{a,c}** All these impacts are adequately and conservatively accounted for with the PAD5 code and associated methodology. **[ ]^{a,c}** This will be confirmed on a plant-specific basis, as the fuel temperature criterion is evaluated as part of the standard reload analysis. The criterion evaluation procedure for fuel temperature is described in Section 7.4.10 and 7.5.1 of (Bowman et al., November 2017).

3.1.7 Clad Free Standing

Design basis and acceptance limit: The fuel system will not be damaged due to excessive fuel clad stress. The clad should be short-term free standing at beginning of life, at power, and during hot hydrostatic testing.

Evaluation: This criterion precludes the possibility of instantaneous collapse of the cladding onto the fuel pellet caused by the differential pressure across the cladding wall. The most limiting condition occurs at beginning of life, when the rod internal pressure is minimal. Westinghouse has evaluated the critical pressure differential that would lead to cladding instability for each Westinghouse fuel type. These evaluations **[**

]^{a,c} will not result in end-of-life conditions more limiting than those at beginning of life. This criterion is not a concern. The criterion evaluation procedure for clad free standing is described in Section 7.4.9 of (Bowman et al., November 2017).

3.1.8 Fuel Clad Fatigue

Design basis and acceptance limit: The fuel system will not be damaged due to fatigue. The fatigue use life factor is limited to less than 1.0 to prevent reaching the material fatigue limit, considering a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles, whichever is more limiting.

Evaluation: Clad fatigue is evaluated using the NRC-approved fuel rod performance code PAD5 (Bowman et al., November 2017). Computer modeling of the fuel duty **[**

]^{a,c} The Langer-O'Donnell fatigue model is used to determine the relationship between strain and fatigue cycles-to-failure.

Fatigue is driven by the accumulated effects of cyclic strains associated with daily load follow. The models in PAD5 have been showed to be adequate for the purpose of evaluating these loads. It is expected that the operation to higher burnup will lead to increased fatigue usage. **[**

]^{a,c} This will be confirmed on a plant-specific basis, as the fatigue criterion is evaluated as part of the standard reload analysis. The criterion evaluation procedure for cladding fatigue is described in Section 7.4.4 of (Bowman et al., November 2017).

3.1.9 Fuel Clad Flattening

Design basis and acceptance limit: Fuel rod failures will not occur due to clad flattening. The fuel rod design will preclude failures due to clad flattening during the projected exposure.

Evaluation: Westinghouse fabricated fuel is sufficiently stable with respect to fuel densification such that the axial column gaps that can form as a result of fuel densification and axial shrinkage are too small to allow clad flattening to occur. Axial column gaps that could occur are sufficiently small such that no densification power spike factor is required. Westinghouse fabrication processes are well controlled with respect to the parameters that impact fuel densification such that adverse fuel performance issues associated with clad flattening do not occur. While the operation to higher burnups will result in longer residence times for the fuel rods, this is not expected to lead to an increased risk of clad flattening, because of the absence of sufficiently large inter-pellet gaps in the fuel column. The basis for the current position on clad flattening is described in (Kersting and Oelrich Jr., March 1995).

3.1.10 Fuel Rod Axial Growth

Design basis and acceptance limit: The fuel system will not be damaged due to excessive axial interference between the fuel rods and the fuel assembly structure. The fuel rods shall be designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the assembly without interference.

Evaluation: Fuel rod growth is evaluated using the NRC-approved code PAD5 (Bowman et al., November 2017). The PAD5 fuel rod axial growth database contains data from a significant number **[ ]**^{a,c} of rods with rod average burnups in the range between 62 GWd/MTU and **[ ]**^{a,c}. These rods have operated without issues and the PAD5 fuel rod axial growth model has been shown to accurately describe these data. Figure 2.5-4 shows the existing fuel rod growth model predicts high burnup measurements well. The operation to higher burnups will lead to some increase in the fuel rod growth. **[**

]^{a,c} This will be confirmed on a plant-specific basis as the rod growth criterion is evaluated as part of the standard reload analysis. The criterion evaluation procedure for fuel rod axial growth is described in Section 7.4.7 of (Bowman et al., November 2017).

3.1.11 Fuel Clad Wear

Design basis and acceptance limit: The fuel system will not be damaged due to fuel rod clad fretting. Westinghouse uses a design **[ ]**^{a,c} as a general guide in **[**

]^{a,c} including fretting wear marks.

Evaluation: The evaluation of the 17x17 Optimized Fuel Assembly (OFA) grids is based on the extensive design and irradiation experience with previous grid designs and the component testing and analysis completed with the 17x17 OFA design.

[

]^{a,c} is well within the Westinghouse guideline of limiting wear to less than **[ ]**^{a,c}.

Specific grid-to-rod fretting (GTRF) wear performance for the incremental burnup limit extension was evaluated on the **[ ]**^{a,c}

[

]a,c

3.2 NUCLEAR DESIGN METHODS AND APPLICATION

Westinghouse nuclear design methods and their application for the incremental burnup extension are discussed in this section. A brief background on the current codes utilized for nuclear design is provided in Section 3.2.1. Section 3.2.2 describes the implications of the incremental burnup extension on the existing nuclear design codes. Example nuclear designs for current fuel management strategies, as well as fuel management with the incremental burnup extension are discussed in Section 3.2.3.

3.2.1 Background

Two principal computer codes have been used in the nuclear design. These are PARAGON or PHOENIX-P (two-dimensional) and the **ANC™** code (two-dimensional and three-dimensional). Descriptions and uses for these codes are given below.

PARAGON (Slagle, et al., August 2004) is a two-dimensional, multi-group transport theory code which utilizes a 70 energy-group cross-section library. It provides the capability for cell lattice modeling on an assembly level. In this design, PARAGON is used to provide homogenized, two-group cross-sections for nodal calculations and feedback models. PARAGON implements an improved flux solution compared to the historical PHOENIX-P code.

PHOENIX-P (Nguyen et al., June 1988) is a two-dimensional, multi-group transport theory code which utilizes a 70 energy-group cross-section library. It provides the capability for cell lattice modeling on an assembly level. In this design, PHOENIX-P is used to provide homogenized, two-group cross-sections for nodal calculations and feedback models. It is also used in a special geometry to generate appropriately weighted constants for the baffle/reflector regions.

The **ANC** code (Liu et al., September 1986) is an advanced nodal code capable of two-dimensional and three-dimensional calculations. In this design, the **ANC** code is employed as the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, the 3-D **ANC** code is used to validate one- and two-dimensional results and to provide information about radial (x-y) peaking factors as a function of axial position. It has the capability of calculating discrete pin powers from the nodal information as well.

3.2.2 Incremental Burnup Limit Extension

The application of rod average burnup limit extension up to [ ]a,c does not require modification or updating of any previously NRC-approved topical reports assessing neutronics and nuclear design. The methods implemented in the neutronic codes for fuel depletion remain unchanged since the mechanism for depleting fuel beyond 62,000 MWd/MTU rod burnup is identical to the methods needed for accruing burnup prior to a rod burnup of 62,000 MWd/MTU. Therefore, the codes and approved methods for core depletion and simulation remain applicable to the incremental burnup limit extension.

3.2.3 Example Nuclear Designs

Figure 3.2-1 illustrates a representative core design that conforms to the current 62,000 MWd/MTU rod average burnup limit for an 18-month cycle operation of a 4-loop core. The design utilized to meet the 62,000 MWd/MTU rod average burnup limit feeds 81/84 assemblies in an alternate cycle loading strategy. Figure 3.2-2 also shows a representative core design that conforms to the current burnup limit for a 3-loop core, which feeds 61/64 assemblies in an alternate cycle loading strategy. Both example patterns utilize a duplex strategy that alternately feeds the center or utilizes the same once burned assembly in the center location. The center assembly can't be used for the third cycle where it would exceed the 62,000 MWd/MTU rod burnup limit. In general, fuel loaded in the core interior for 2 cycles must then be discharged to meet the rod burnup of 62,000 MWd/MTU. There are fuel assemblies being utilized for the third cycle; however, the loading strategy for these assemblies is unique in that the assembly can only be loaded in the core interior for one cycle, and must be loaded in the lower power locations on the core periphery in the next two cycles to minimize the burnup accumulation to meet the 62,000 MWd/MTU peak rod average burnup limit.

Figures 3.2-3 and 3.2-4 illustrate the core design that conforms with the **[ ]**^{a,c} rod average burnup limit for an 18-month cycle operation for the identical 4 and 3-loop cores. The **[ ]**^{a,c} rod average burnup limit increases the average discharge burnup, maximizing the fuel utilization by reducing the number of feed assemblies per cycle and increasing the number of third cycle burnt fuel assemblies in the core.

The rod average burnup increase from **[ ]**^{a,c} Figure 3.2-5 and Figure 3.2-6 illustrate the peaking factors versus rod burnup characteristic of representative core designs that conform to the 62,000 MWd/MTU and the **[ ]**^{a,c} burnup limits for a 4-loop pressurized water reactor (PWR) with 18-month cycle operation. Similarly, Figure 3.2-7 and Figure 3.2-8 illustrate the peaking factors versus rod burnup characteristic of representative core designs that conform to the 62,000 MWd/MTU and the **[ ]**^{a,c} burnup limits for a 3-loop PWR with 18-month cycle operation. The comparisons show that both the $F_{\Delta H}$ and F_Q of the core designs which conform to the incremental burnup extension of **[ ]**^{a,c} are comparable to the current designs and operation. Therefore, the nuclear designs and fuel utilization for the fuel rods which do not exceed current burnup limits under the incremental burnup extension are similar to current operation, such that the conclusions regarding the safety limits remain applicable and margin to safety limits will largely remain unchanged.

A core design with widespread interior assemblies exceeding the 62,000 MWd/MTU rod average burnup limit is not supportable with the current enrichment limit of 5 w/o ^{235}U due to the lack of required fissile material in the core to maintain criticality. The **ADOPT** fuel pellet, which has a higher density than the currently manufactured fuel pellets, potentially provides a complimentary benefit when combined with the **[ ]**^{a,c} peak pin average burnup application. The higher pellet density increases fissile material while maintaining the enrichment under the 5.0 w/o limit. Even with **ADOPT** fuel pellets' high density, the assemblies that contain rod burnups exceeding the 62,000 MWd/MTU pin burnup limit are [

]^{a,c}

a,c

Figure 3.2-1: Representative Core Design for an 18-Month Cycle Operation of a 4-Loop Core, Current 62,000 MWd/MTU Rod Average Burnup Limit

a,c

Figure 3.2-2: Representative Core Design for an 18-Month Cycle Operation of a 3-Loop Core, Current 62,000 MWd/MTU Rod Average Burnup Limit

a,c

Figure 3.2-3: Representative Core Design for an 18-Month Cycle Operation of a 4-Loop Core, Incremental [ ]a,c Rod Average Burnup Limit

a,c

Figure 3.2-4: Representative Core Design for an 18-Month Cycle Operation of a 3-Loop Core, Incremental [ ]^{a,c} Rod Average Burnup Limit

a,c

Figure 3.2-5: $F_{\Delta H}$ Comparison for a 4-loop PWR with 18-Month Cycle Operation Between [ ]^{a,c} Representative Core Designs

a,c

Figure 3.2-6: F_{Q} Comparison for a 4-loop PWR with 18-Month Cycle Operation Between [ ]^{a,c} Representative Core Designs

a,c

Figure 3.2-7: $F_{\Delta H}$ Comparison for a 3-loop PWR with 18-Month Cycle Operation Between [ ]a,c Representative Core Designs

a,c

Figure 3.2-8: F_Q Comparison for a 3-loop PWR with 18-Month Cycle Operation Between [ ]a,c Representative Core Designs

3.3 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design methods applied to PWR DNB analyses consist of DNB correlations such as WRB-1 (Motley et al., July 1984), WRB-2 (Davidson and Kramer, September 1985), WRB-2M (Smith et al., April 1999), WSSV (Joffre et al., August 2007), and WNG-1 (Joffre et al., February 2010), a thermal-hydraulic (T/H) subchannel code such as the Westinghouse version of the VIPRE-01 code referred to as the VIPRE-W code (Sung et al., October 1999), and a statistical method for determination of a 95/95 DNB Ratio (DNBR) limit, such as the Revised Thermal Design Procedure (RTDP) (Friedland and Ray, April 1989) or the Westinghouse Thermal Design Procedure (WTDP) (Sung and Singh, April 2020). Thermal-hydraulic analysis can also be performed as part of the integrated non-loss-of-coolant accident (non-LOCA) analysis methodology described in (Beard et al., October 2003) and (Beard et al., August 2006). Section 5.1.2 of this topical report contains a discussion of the impact of Regulatory Guide (RG) 1.236 on reactivity insertion accident (control rod ejection) events.

No modification or update is required to any of the NRC-approved topical reports (identified in the prior paragraph) on DNB correlations and thermal-hydraulic analysis methods for application to fuel with higher burnups, including fuel designs containing the **ADOPT** fuel pellets. The fuel burnup increase does not affect DNB correlations developed from DNB experiments, or the method for DNBR calculations. The VIPRE-W code can perform steady-state and transient DNBR calculations and non-LOCA post-Critical Heat Flux (CHF) fuel rod transient analysis based on the fuel design input, including fuel temperatures, applicable to the fuel with higher burnups.

The method using the VIPRE-W code for the DNB propagation evaluation is described in (Sidener et al., June 2006). The fuel burnup does not change the acceptance criteria and conditions of the DNB propagation evaluation method in (Sidener et al., June 2006). DNB and DNB propagation are **[**

]^{a,c}

The impact of fuel rod bowing on DNB as a function of burnup is evaluated using an NRC-approved evaluation methodology, such as that in (Skaritka, July 1979, safety evaluation report (SER) included with Gresham, August 2011) and (Thomas, December 1982). As input to the rod bow DNBR penalty evaluation for the high burnup fuel, the following rod-to-rod gap closure correlation is used:

$$\text{Sgap} = A + B\,(\text{BU})$$

where Sgap is the upper 95% tolerance limit for the standard deviation of channel closure for the worst grid-to-grid span of a fuel assembly, A and B are fuel specific coefficients applicable to the high burnup fuel design, and BU is the average fuel assembly burnup. There is no change in the DNBR penalty as a function of the gap closure from DNB testing as listed in (Thomas, December 1982) for the high burnup fuel. The effect of rod bow on DNBR is addressed up to a certain burnup in a plant DNB analysis. The maximum rod bow DNBR penalty is determined at a fuel assembly burnup typically below 33,000 MWd/MTU. Beyond that burnup, a credit is taken for the fuel rod to be at sufficiently low power so that it can no longer be DNB limiting.

3.4 REFERENCES

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4 LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODS

The impact of the incremental burnup extension described in this topical report varies across different functional areas. For some previously discussed functional areas, such as mechanical design, existing data is used to show that applicable criteria continue to be met for higher burnup fuel. For other areas, such as fuel rod design, it is shown that the existing PAD5 code models and associated methods (Bowman et al., 2017) remain valid for higher burnup fuel. For many non-loss-of-coolant accident (non-LOCA) and containment analyses (discussed later), it is shown that the existing methods remain applicable for higher burnup fuel. For LOCA analysis, some limited code updates are needed to analyze higher burnup fuel as described later in this section.

Research findings related to fuel fragmentation, relocation, and dispersal (FFRD) under LOCA conditions are described in (Raynaud, 2012). The staff concluded therein that fuel fragmentation does occur in high burnup fuel, and pellet fragments could disperse from a high burnup fuel rod which ruptures during a postulated LOCA. The amount of potential dispersal is a function of the rupture opening size, and the size of the pellet fragments "which existing data show to decrease with increasing burnup" per (Raynaud, 2012). Since this topical report increases the fuel rod average burnup limit beyond 62 GWd/MTU, the potential for FFRD during a LOCA must be addressed.

A relatively simple method for evaluating the potential for cladding burst was previously outlined by Westinghouse in (Nissley, Frepoli, and Ohkawa, 2005). This work was recognized by the Nuclear Regulatory Commission (NRC) in (Raynaud, 2012). However, this work was based on the original Westinghouse Code Qualification Document (CQD) Evaluation Model (Bajorek et al., 1998) which focused on fuel near beginning of life and did not account for fuel pellet thermal conductivity degradation (TCD). Therefore, additional rigor is required to consider high burnup fuel conditions with TCD, such as the use of the most recently NRC approved Westinghouse thermal-hydraulic LOCA analysis codes.

4.1 INTRODUCTION

Compliance with the LOCA analysis acceptance criteria prescribed in 10 CFR 50.46 is discussed in Section 4.2 of this topical report. It is concluded in Section 4.2 that the **[**

]^{a,c} However, as previously discussed, the consequences of potential FFRD during a postulated LOCA must be addressed.

The latest Westinghouse LOCA evaluation model (EM) is the **FULL SPECTRUM** LOCA (**FSLOCA**) methodology (Kobelak et al., 2016), and the associated thermal-hydraulic code is WCOBRA/TRAC-TF2. 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 discussion in (Kobelak et al., 2016) remains applicable. The changes to the WCOBRA/TRAC-TF2 code described within this topical report are primarily focused on the decay heat models relative to higher burnup fuel and the fuel rod models related to cladding deformation and rupture. The method described in this topical report is based on the **FSLOCA** EM framework and will utilize a modified version of the same thermal-hydraulic code.

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 pressurized water reactors (PWRs) equipped with upper plenum injection (UPI), 3-loop and 4-loop plants with emergency core cooling system (ECCS) injection into the cold legs, and Combustion Engineering (CE) designs. 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 large-break LOCA (LBLOCA) analysis utilize the **FSLOCA** EM as extended to those plant classes.

4.2 ANALYSIS REQUIREMENTS

The figures of merit considered in the **FSLOCA** EM are consistent with the 10 CFR 50.46 acceptance criteria. It is demonstrated in Section 4.2.1 of this topical report that **[**

]^{a,c} Additionally, the US NRC has initiated the formal process to revise the ECCS acceptance criteria in Section 50.46c of Title 10 of the Code of Federal Regulations (10 CFR 50.46c) via issuance of proposed rulemaking, and has published a proposed rule (US NRC, 2016) hereafter referred to as the 10 CFR 50.46c rulemaking. It is demonstrated in Section 4.2.2 that the **[**

]^{a,c}

In addition to the 10 CFR 50.46 / proposed 10 CFR 50.46c acceptance criteria, there is an additional requirement which must be satisfied for higher burnup fuel under LOCA conditions to address the FFRD concerns which are the subject of (McCree, 2015). For the incremental burnup extension, this additional requirement is to demonstrate the absence of cladding rupture for any higher burnup rods during a postulated LOCA. If fuel rod cladding rupture is precluded for the high burnup fuel rods, then no fuel dispersal is possible since all of the fuel will be retained within the cladding. Additional information regarding this additional requirement for high burnup fuel is provided in Section 4.2.3.

4.2.1 Compliance with 10 CFR 50.46 Acceptance Criteria

This section describes how the ECCS acceptance criteria defined in 10 CFR 50.46 are met with a high degree of probability for high burnup fuel. The acceptance criteria prescribed in 10 CFR 50.46 are as follows:

- **Peak Cladding Temperature** – The peak cladding temperature (PCT) must be less than the regulatory limit of 2,200°F. **[**

  ]^{a,c}

- **Maximum Cladding Oxidation** – The maximum local oxidation (MLO) must be less than the regulatory limit of 17 percent of cladding thickness. The MLO criterion **[**

  ]^{a,c} (see Section 4.3.1.1).

- **Maximum Hydrogen Generation** – The hydrogen generated in the core, as determined by estimating the total volume of cladding oxidized for the limiting conditions, must be less than the regulatory limit of 0.01 times the maximum theoretical amount. **[**

  ]^{a,c}

- **Coolable Geometry** – This acceptance criterion is met by compliance with acceptance criteria (b)(1) (peak cladding temperature) and (b)(2) (maximum cladding oxidation), and accounting for any grid deformation due to combined seismic and LOCA loads that extends to the in-board assemblies. As previously discussed, the high burnup fuel **[**

  ]^{a,c} The calculation of grid deformation due to combined seismic and LOCA loads will consider end-of-life (EOL) conditions, as discussed in Section 2.3.1.3 of this topical report.

- **Long-Term Cooling** – The incremental burnup extension does not impact existing methods for the calculations of long-term cooling. Any violations of associated reload safety analysis limits will be addressed on a plant-specific basis during implementation as indicated in Section 7.2.

4.2.2 Compliance with Proposed 10 CFR 50.46c Acceptance Criteria

There are several new phenomena that are proposed to be included within 10 CFR 50.46c which are absent from the current acceptance criteria. These phenomena include cladding embrittlement due to hydrogen-enhanced oxygen absorption, breakaway oxidation, oxygen ingress to the inside of the cladding from the fuel bond layer, and the thermal effect of the crud and oxide layer. These various considerations are addressed in turn.

- **Cladding Embrittlement** – As discussed in Section 4.2.1, **[**

  ]^{a,c}

- **Breakaway Oxidation** – The high burnup fuel addressed within this topical report **[**

  ]^{a,c}

- **Oxygen Ingress from Fuel Bond Layer** – The topic of fuel pellet-to-cladding bonding is discussed in (Billone et al., July 2008). It is stated in Section 7.6 therein that for burnups above 30 GWD/MTU there will be some fuel-to-cladding bonding in pressurized water reactor fuel rods, and by 50 GWD/MTU fuel bonding is likely to be well developed, and oxygen pickup on the interior of the cladding from the bond and fuel may be as high as the oxygen pickup from steam flowing on the cladding exterior. The topic is further discussed in Section C.3.B of (US NRC, 2016b), where it is stated that an "acceptable approach to account for oxygen ingress on the cladding inside surface due to the fuel-cladding bond layer is to use twice the oxidation as on the exterior of the cladding for un-ruptured locations for fuel rods with a local exposure beyond 30 GWD/MTU." As discussed in Section 4.2.1, **[**

  ]^{a,c}

- **Thermal Effect of Crud and Oxide Layer** – The approved **ZIRLO** and **Optimized ZIRLO** cladding integral form corrosion model (Garde et al., 2013) was implemented into the PAD5 fuel performance code (Bowman et al., 2017). The PAD5 fuel performance code is used to initialize the fuel rods in the **FULL SPECTRUM** LOCA evaluation methodology (see Section 26.4 and Limitation and Condition (L&C) #6 of (Kobelak et al., 2016)). Additionally, the PAD5 fuel performance data used to initialize the fuel rods **[**

  ]^{a,c} based on the PAD5 fuel performance code (Bowman et al., 2017).

4.2.3 Fuel Fragmentation, Relocation, and Dispersal

The NRC staff evaluated FFRD under LOCA conditions as it related to the draft 10 CFR 50.46c rulemaking in (McCree, 2015). It is written in (McCree, 2015) that "… research has shown as burnup exceeds 62 GWd/MTU, fuel becomes increasingly susceptible to FFRD." It is also written therein that "Given the burnup and utilization limitations on existing fuel designs, the staff does not foresee the ongoing research would identify a need to withdraw approval of existing fuel designs." The simplest approach for addressing concerns related to FFRD is to demonstrate that the high burnup rods are sufficiently low in power that burst will not occur, thereby precluding dispersal.

A methodology for performing LOCA calculations is developed within this topical report to determine if any higher burnup fuel rods would be predicted to rupture during a postulated LOCA. The method utilizes the most recent NRC-approved Westinghouse best-estimate LOCA thermal-hydraulic code (with modifications described in this section of the topical report), which has the models necessary to perform the calculations. The objective for the methodology is to show that cladding rupture would not occur during a postulated LOCA for any rods which exceed an average burnup of 62 GWd/MTU, thus avoiding the need for further assessment of postulated FFRD consequences.

4.3 REVIEW OF PRIOR PHENOMENA IDENTIFICATION AND RANKING

In this section, the phenomenon identification and ranking tables (PIRTs) from the **FSLOCA** EM as well as an industry PIRT for postulated LOCAs in PWRs with high burnup fuel are reviewed. The **FSLOCA** EM PIRT is reviewed in Section 4.3.1, and the industry PIRT is reviewed in Section 4.3.2.

4.3.1 FULL SPECTRUM LOCA 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 high burnup program on LOCA analysis is related to the fuel rod response, 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 in the context of the incremental burnup extension. 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.

4.3.1.1 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 4.4.1 herein. The approach for calibration of the fuel temperatures (stored energy) is discussed in Section 4.7.3.2.1.

Clad Oxidation

Cladding oxidation is considered highly important in the prediction of a postulated LOCA transient response. The exothermic metal-water reaction becomes an increasingly significant source of heat addition with increasing cladding temperature. **[**

]^{a,c}

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]^{a,c}

With respect to pre-existing (or pre-accident) corrosion, the treatment within the **FSLOCA** EM is noted (Section 29.4.2.1 of Kobelak et al., 2016).

(1) The approved corrosion models for **ZIRLO** and **Optimized ZIRLO** cladding are incorporated into the PAD5 code as described in Section 3.3.1 of (Bowman et al., 2017). The corrosion model uncertainties were determined as a [

]^{a,c}

(2) Compliance with the 10 CFR 50.46 (b)(2) criterion is shown by summing the [ ]^{a,c} pre-accident corrosion and the transient equivalent cladding reacted (ECR).

The treatment of the corrosion in generation of PAD5 fuel performance data for WCOBRA/TRAC-TF2 initialization is already conservative as discussed in the response to request for additional information (RAI) #37 in (Kobelak et al., 2016). As discussed therein, [

]^{a,c} Therefore, no updates are required beyond demonstrating the applicability of PAD5 to higher burnup fuel, which is discussed in Section 3.1.

Decay Heat

The WCOBRA/TRAC-TF2 code decay heat model is based on the 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 WCOBRA/TRAC-TF2 were only assessed 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 WCOBRA/TRAC-TF2 decay heat model are discussed in Section 4.6 for extended burnup.

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 4.4.2 and 4.4.3, respectively.

4.3.1.2 Core

Critical Heat Flux

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

]^{a,c}

Post-CHF Heat Transfer / Steam Cooling

The analysis of 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 **[**

]^{a,c}

Heat Transfer to a Covered Core

The analysis of 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 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. For the cladding rupture calculations, a bounding approach is utilized to define the initial condition for the assemblies / fuel rods of interest as discussed in Section 4.7.3.2.1.

Void Generation / Void Distribution

The analysis of 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 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 split-break area, the break discharge coefficients, the broken loop nozzle resistance, and the broken loop pump resistance (illustrated for some parameters in Section 28 of (Kobelak et al., 2016)). While the analysis of higher burnup fuel does not influence this global behavior, the analytical approach for high burnup fuel must account for the uncertainty in these parameters. See Section 4.7.3.2.2 for a description of how the analytical approach accounts for uncertainty in the flow reversal and stagnation behavior in the core.

Flow Resistance

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

4.3.2 Industry PIRT Review

The review of the **FSLOCA** EM PIRT discussed in Section 4.3.1 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).

4.3.2.1 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), which was found to be appropriate for the analysis of higher burnup fuel per Section 3.1 herein. The related WCOBRA/TRAC-TF2 code models were assessed in response to RAIs 36 through 39 on the **FSLOCA** EM, and found to be reasonable for the analysis of high burnup fuel.

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 4.4.3.

Location of Burst

Cladding rupture is the parameter of interest for the high burnup assemblies addressed 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 4.4.1 herein, and an assessment of the cladding deformation in WCOBRA/TRAC-TF2 is discussed in Section 4.4.2 herein.

4.3.2.2 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 4.4.1 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 in Section 4.7.3.2.1.

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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Size of Burst Opening

Cladding rupture is the parameter of interest for the high burnup assemblies addressed 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. This particular item would be of increased significance in attempting to quantify the amount of fuel dispersal from a rod under accident conditions.

Burst Criteria

Burst criteria was already discussed in Section 4.3.2.1.

Time of Burst

Cladding rupture is the parameter of interest for the high burnup assemblies addressed 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.

4.4 WCOBRA/TRAC-TF2 FUEL ROD MODELS

The fuel rod models in the code identified in Section 4.3 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 models are described within this section.

4.4.1 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 **[**

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4.4.2 Cladding Deformation

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4.4.3 Cladding Rupture

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]a,c Afterward, the existing models and necessary updates are discussed in Sections 4.4.3.2 and 4.4.3.3.

4.4.3.1 Effect of Hydrogen Uptake into the Cladding

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4.4.3.2 Cladding Rupture Models

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4.4.3.3 Cladding Deformation Due to Rupture

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4.4.4 Fuel Rod Initialization

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4.4.5 Fission Gas Release

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]a,c

Table 4.4-1 Overview of Cladding Rupture Data for As-Fabricated, Pre-Hydrided, and Irradiated Cladding Samples

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a,c

Figure 4.4-1: Creep Rate of [

]a,c

a,b,c

Figure 4.4-2: Rupture Temperature versus Engineering Hoop Stress for As-Fabricated Cladding Samples from Table 4.4-1

a,c

Figure 4.4-3: Rupture Temperature versus Engineering Hoop Stress for Pre-Hydrided Cladding Samples from Table 4.4-1

a,b,c

Figure 4.4-4: Rupture Temperature versus Engineering Hoop Stress for Irradiated Cladding Samples from Table 4.4-1

a,b,c

Figure 4.4-5: Rupture Temperature versus Engineering Hoop Stress for All Pre-Hydrided and Irradiated Cladding Data from Table 4.4-1

a,b,c

Figure 4.4-6: Rupture Temperature versus Engineering Hoop Stress for Pre-Hydrided and Irradiated Cladding Data [ ]a,c

a,c

Figure 4.4-7: Burst Strain versus Rupture Temperature for Zircaloy-2 and Zircaloy-4 Cladding Samples from Table 4.4-1

a,b,c

Figure 4.4-8: Burst Strain versus Rupture Temperature for ZIRLO and Optimized ZIRLO Cladding Samples from Table 4.4-1

a,b,c

Figure 4.4-9: Rupture Temperature versus Engineering Hoop Stress for All Cladding Samples from Table 4.4-1 with WCOBRA/TRAC-TF2 Nominal Model

a,b,c

Figure 4.4-10: Rupture Temperature versus Engineering Hoop Stress for All Cladding Samples from Table 4.4-1 with [ ]a,c

a,b,c

Figure 4.4-11: Proposed Cladding Rupture Model for High Burnup Fuel Burst Calculations versus Datasets from Table 4.4-1 with [ ]^{a,c}

a,b,c

Figure 4.4-12: Proposed Cladding Rupture Model for High Burnup Fuel Burst Calculations versus [ ][a,c] from Table 4.4-1

a,c

Figure 4.4-13: Circumferential Burst Strain versus Burst Temperature for Zirc-2 and Zirc-4 Cladding Samples from Table 4.4-1 with WCOBRA/TRAC-TF2 Nominal Zirc-4 Cladding Model

a,b,c

Figure 4.4-14: Circumferential Burst Strain versus Burst Temperature for ZIRLO and Optimized ZIRLO Cladding Samples from Table 4.4-1 with WCOBRA/TRAC-TF2 Nominal ZIRLO and Optimized ZIRLO Cladding Model

a,c

Figure 4.4-15: Temperature Near the Outside of the Fuel Pellet at all 80 Axial Elevations (Each Line is a Single Elevation) in a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 1

a,c

Figure 4.4-16: Temperature Near the Outside of the Fuel Pellet at all 80 Axial Elevations (Each Line is a Single Elevation) in a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 2

a,c

Figure 4.4-17A: Pellet Radial Temperature Profile [ ]a,c During Blowdown for a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 3

a,c

Figure 4.4-17B: Pellet Radial Temperature Profile [ ]^{a,c} Through Quench for a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 3

a,c

Figure 4.4-18A: Pellet Radial Temperature Profile [ ]^{a,c} During Blowdown for a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 4

a,c

Figure 4.4-18B: Pellet Radial Temperature Profile [ ][a,c] Through Quench for a High Burnup Fuel Rod under the Incremental Burnup Extension Program, Case 4

4.5 THERMAL PROPERTIES OF NUCLEAR FUEL ROD MATERIALS

The thermal properties of nuclear fuel rod materials included in the WCOBRA/TRAC-TF2 code for uranium-dioxide, Zircaloy-4, and **ZIRLO** / **Optimized ZIRLO** cladding are discussed in Section 11.4 of (Kobelak et al., 2016). The changes to those models for analysis of high burnup fuel are discussed in the following subsection(s).

4.5.1 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 4-1 through 4-3 as follows:

a,c

[ ] (4-1)

where:

[ ]^{a,c} (4-2)

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,

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

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

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a,c

Figure 4.5-1: Comparison of the Modified NFI and PAD5 UO_2 Pellet Thermal Conductivity Models up to 80 GWd/MTU Burnup for Fuel at 95% of Theoretical Density

a,c

Figure 4.5-2: Comparison of the Modified NFI and PAD5 UO_2 Pellet Thermal Conductivity Models up to 80 GWd/MTU Burnup for Fuel at 95% of Theoretical Density, Temperature Range of Interest

4.6 WCOBRA/TRAC-TF2 KINETICS AND DECAY HEAT MODEL

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.

4.6.1 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}**. The updated physics data are based on ALPHA/PARAGON (Slagle et al., 2004) with cross-section library version ENDF/B-VI. The nuclear physics data was coded directly into the WCOBRA/TRAC-TF2 code rather than curve fitting the data as was done previously. 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 4.6-1 through 4.6-14 herein for the updated physics data up to a burnup of **[ ]^{a,c}**.

4.6.2 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 18-month cycles). **[ ]^{a,c}**

The third condition is that the number of fissions per initial fissile atom is less than 3.0. **[ ]^{a,c}**

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]a,c

In conclusion, it was determined that the existing WCOBRA/TRAC-TF2 neutron capture correction is valid for analysis of higher burnup fuel.

4.6.3 Normalized Fission Interaction Frequency

The normalized interaction frequency is discussed in Section 9.3 of (Kobelak et al., November 2016), and uses the coefficients presented in Table 9-5 therein. A comparison of the normalized fission interaction frequency with newer physics data calculated from ALPHA/PARAGON (Slagle et al., 2004) indicates that the WCOBRA/TRAC-TF2 model predicts a lower interaction frequency than the newer data. Therefore, the fitting coefficients from Table 9-5 of (Kobelak et al., November 2016) are updated as shown in Table 4.6-1 to reflect the nuclear physics data from ALPHA/PARAGON (Slagle et al., 2004). This update will tend to slightly increase the power at the lower moderator densities expected during a LOCA.

4.6.4 Gamma Energy Redistribution

The modeling approach for gamma energy redistribution approved as part of the **FSLOCA** EM remains valid for the incremental burnup extension. **[**

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4.6.5 Conclusions

The nuclear physics data within the WCOBRA/TRAC-TF2 code were updated as described in this section to extend the validity of the kinetics and decay heat model to an assembly and rod average burnup of **[**

]a,c. It was found that the existing modeling approaches for the neutron capture correction and gamma energy redistribution are valid for the analysis of higher burnup fuel.

Table 4.6-1 Typical Normalized Interaction Frequency Fit Data

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a,c

Figure 4.6-1: U-235 Fission Fraction (Updated Figure 9-1 from (Kobelak et al., 2016))

a,c

Figure 4.6-2: Pu-239 Fission Fraction (Updated Figure 9-2 from (Kobelak et al., 2016))

a,c

Figure 4.6-3: U-238 Fission Fraction (Updated Figure 9-3 from (Kobelak et al., 2016))

a,c

Figure 4.6-4: $\overline{\beta}$ versus Burnup (Updated Figure 9-5 from (Kobelak et al., 2016))

a,c

Figure 4.6-5: Prompt Neutron Lifetime (Updated Figure 9-6 from (Kobelak et al., 2016))

a,c

Figure 4.6-6: Prompt Energy Release (Updated Figure 9-7 from (Kobelak et al., 2016))

a,c

Figure 4.6-7: Total Energy Release (Updated Figure 9-8 from (Kobelak et al., 2016))

a,c

Figure 4.6-8: Delayed Group I Lambda (Updated Figure 9-9 from (Kobelak et al., 2016))

a,c

Figure 4.6-9: Delayed Group II Lambda (Updated Figure 9-10 from (Kobelak et al., 2016))

a,c

Figure 4.6-10: Delayed Group III Lambda (Updated Figure 9-11 from (Kobelak et al., 2016))

a,c

Figure 4.6-11: Delayed Group IV Lambda (Updated Figure 9-12 from (Kobelak et al., 2016))

a,c

Figure 4.6-12: Delayed Group V Lambda (Updated Figure 9-13 from (Kobelak et al., 2016))

a,c

Figure 4.6-13: Delayed Group VI Lambda (Updated Figure 9-14 from (Kobelak et al., 2016))

a,c

Figure 4.6-14: U-238 Capture / Fission Ratio as a Function of Initial Enrichment and Burnup (Updated Figure 9-15 from (Kobelak et al., 2016))

4.7 FUEL ROD CLADDING RUPTURE CALCULATION METHOD

4.7.1 Small-Break LOCA

Within the **FSLOCA** EM, the term “Region I” refers to the small-break loss-of-coolant accident (SBLOCA) range of break sizes and the term “Region II” refers to the LBLOCA range of break sizes. **[**

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4.7.2 Intermediate-Break LOCA

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4.7.3 Large-Break LOCA

4.7.3.1 Introduction

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4.7.3.2 Treatment for Uncertainty Contributors for Deterministic Rupture Calculations

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4.7.3.2.1 Fuel Rod Initialization

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4.7.3.2.2 Break Specification, Offsite Power Availability, Global Models, and Axial Power Distribution

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4.7.3.2.3 Decay Heat Uncertainty

The cladding rupture calculations will utilize **[**
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Table 4.7-1 Peak and Average Linear Heat Rates for Results Presented in Figures 4.7-1 and 4.7-2

Plant	Peak Linear Heat Rate (kW/ft)	Average Linear Heat Rate (kW/ft)
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		]^{a,c}

Table 4.7-2 [ ]a,c for Deterministic LBLOCA Cladding Rupture Calculations Run Matrix

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					]a,c

1) [
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a,c

Figure 4.7-1: Peak Cladding Temperature Results for 3-Loop PWR Break Spectrum Studies at Bounding Linear Heat Rates

(Note: Each set of plot points indicates a different analysis.)

a,c

Figure 4.7-2: Peak Cladding Temperature Results for 4-Loop PWR Break Spectrum Studies at Bounding Linear Heat Rates

(Note: Each set of plot points indicates a different analysis.)

a,c

Figure 4.7-3: SBLOCA, IBLOCA, and LBLOCA Transient Results for a 2-loop PWR

a,c

Figure 4.7-4: IBLOCA and LBLOCA Transient Results for a 3-loop PWR (Figure 31.4-1 from Kobelak et al., November 2016)

a,c

Figure 4.7-5: SBLOCA, IBLOCA, and LBLOCA Transient Results for a 4-loop PWR (Figure 4-5 from Mercier, July 2018)

4.8 DEMONSTRATION ANALYSIS

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]a,c A demonstration analysis is performed for a 3-loop Westinghouse-designed PWR using the deterministic rupture calculation method discussed in Section 4.7.3. For an actual plant application, the nuclear design and fuel rod design input, as well as the associated limitations on each cycle of operation, would be determined on a plant-specific basis.

4.8.1 Nuclear Design Inputs

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]a,c These parameters would be checked each reload to ensure continued applicability of the cladding rupture calculations. The axial power distributions analyzed in the cladding rupture calculations are presented in Figure 4.8-1.

4.8.2 Results

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4.8.3 Confirmation of Blowdown Flow Conditions

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4.8.4 Conclusions from Demonstration Analysis

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Table 4.8-1 Results for Deterministic LBLOCA Cladding Rupture Calculations

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Figure 4.8-1: Axial Power Distributions for the Deterministic Cladding Rupture Calculation Demonstration Analysis

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Figure 4.8-2: Cladding Temperature and Rupture Temperature Near Limiting Elevation for the Limiting Burst Margin Case from the Demonstration Analysis

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Figure 4.8-3: Comparison of the Liquid Mass Flow Rate at the Bottom of the Core from the Nine Run Deterministic Run Matrix (Red Lines) and the Top 50 PCT Cases (Black Lines) from the Statistical Analysis for the Demonstration Plant

(Note: The break occurs at 50 seconds on this plot.)

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Figure 4.8-4: Comparison of the Liquid Mass Flow Rate at the Top of the Core from the Nine Run Deterministic Run Matrix (Red Lines) and the Top 50 PCT Cases (Black Lines) from the Statistical Analysis for the Demonstration Plant

(Note: The break occurs at 50 seconds on this plot.)

4.9 REFERENCES

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5 NON-LOCA SAFETY ANALYSIS METHODS

In this section, the impact of fuel achieving average burnup beyond [ ]^{a,c} on affected safety analyses other than loss-of-coolant accident (LOCA) analysis is addressed.

The impact on non-LOCA transient analyses is addressed in Section 5.1. The impact on containment analyses is discussed in Section 5.2.

5.1 TRANSIENT ANALYSIS

This section discusses the effect of the increased fuel rod burnup limit on the non-LOCA transient analyses.

5.1.1 Transient Analyses

The implementation of the increased fuel rod burnup limit for the non-LOCA transient analyses has been evaluated. The impact of the burnup increase will be addressed through input changes to existing analyses. The approved non-LOCA codes and methods will remain applicable.

Non-LOCA transient analyses are performed to demonstrate that the acceptance criteria related to fuel rod failure and core coolability are met. No new fuel rod failure or accident phenomena are identified due to an increase of the maximum fuel rod burnup limit to [ ]^{a,c}

To assess the impact of an increase in fuel rod burnup, there are two categories of non-LOCA events that need to be considered:

1. Events that are dependent upon core-average effects, and
2. Events analyzed to address local effects in the fuel rods.

The first category of events is typically analyzed in a single step with a system code. For this category, the non-LOCA events are analyzed to address gross plant criteria, such as loss of shutdown margin, margin to hot leg saturation, overpressurization of the reactor coolant system (RCS), overpressurization of the secondary system, or overfilling of the pressurizer. [

]^{a,c}

For the second category of events, analyses are performed to address local effects in the fuel rods. Such analyses are performed in two steps: 1) predictions of average core response to an initiating event, and 2) hot channel or hot spot analyses for such local effects as fuel enthalpy (cal/g), minimum departure from nucleate boiling (DNB) ratio (DNBR), fuel melting, and peak cladding temperature (PCT). The increased fuel rod burnup limit does not impact the event-specific acceptance criteria used for the local condition analyses. The three-dimensional (3-D) kinetics method used to address the new reactivity-initiated accident criteria is assessed in Section 5.1.2.

5.1.2 Reactivity Insertion Accidents

A reactivity insertion accident (RIA) or reactivity initiated accident, which in the context of this topical report for pressurized water reactor (PWR) applications is a control rod ejection (CRE) accident, is classified as a Condition IV event by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (ANSI N18.2-1973) due to the extremely low probability of its occurrence. The accident is defined as mechanical failure of a control rod mechanism pressure housing such that the RCS pressure would cause ejection of a partially or fully inserted control rod and drive shaft to its fully withdrawn position. If the reactor is at or near critical, the consequences of this mechanical failure are a rapid reactivity insertion and core power increase together with an adverse core power distribution, potentially leading to localized fuel rod damage. The power increase is arrested primarily by the negative reactivity due to the Doppler feedback resulting from the fuel heatup, and the transient is terminated by a reactor trip which is initiated shortly after the beginning of the transient.

Regulatory Guide (RG) 1.236 (US NRC, June 2020) provides guidance on acceptable analytical methods, assumptions, and limits on applicability for evaluating the control rod ejection accident, applicable to the reactor cores containing the Nuclear Regulatory Commission (NRC)-approved fuel designs up to a maximum fuel rod average burnup of 68 GWd/MTU. The acceptable fuel failure thresholds during a CRE are the following:

1) Fuel Rod Cladding Failure Thresholds

   The high-temperature cladding failure threshold is shown in Figure 1 of RG 1.236, expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (MPa). For non-prompt critical excursions, or prompt critical scenarios which experience a prolonged power level following the prompt pulse, fuel cladding failure is presumed if the local heat flux reaches DNB as determined by the DNBR, which is the ratio of predicted DNB heat flux to the local heat flux, using an NRC-approved DNB correlation and its 95/95 DNBR limit.

   The pellet-clad mechanical interaction (PCMI) cladding failure thresholds for the stress relief annealed (SRA) cladding in Figures 3 and 5 of RG 1.236 are applicable to the fuel designs containing the **ZIRLO** or the **Optimized ZIRLO** cladding (refer to the response to Westinghouse comment #2 on applicability contained on page 6 of (US NRC, June 2020a) for **Optimized ZIRLO** cladding), and the **ADOPT** pellets (see Appendix B). The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content (wppm).

   Fuel cladding failure is also presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions.

2) Allowable Limits on Radiological Consequences

   RG 1.183 and RG 1.195 contain the accident dose radiological consequence assumptions for the PWR control rod ejection accident. The allowable limit is defined on a plant-specific basis. A fuel rod is assumed to fail for radiological consequence evaluation if any of the fuel rod cladding failure thresholds is exceeded.

   No fuel rod cladding failure, including no DNB occurrence, is expected to occur for any fuel rods having an average burnup greater than 62 GWd/MTU during a CRE event due to their low power factors and analyses demonstrating that DNBR values remain above the design limit DNBR. The conclusion that fuel rod cladding failure will not occur in higher burnup rods (> 62 GWd/MTU) after a CRE event will be confirmed on a plant-specific basis during implementation.

3) Allowable Limits on Reactor Coolant System Pressure

   The allowable limits for the RCS pressure boundary are specified in a plant's updated Final Safety Analysis Report (UFSAR). The allowable limits on RCS pressure are also subject to plant-specific confirmation for reactor cores containing higher burnup fuel.

4) Allowable Limits on Damaged Core Coolability

   The following limits are applicable to the reactor core containing higher burnup fuel:

   - Peak radial average fuel enthalpy should remain below 230 cal/g.
   - Fuel melting is limited to less than 10 percent of the fuel volume in the centerline region.

An analysis of the consequences of the control rod ejection accident is typically presented in Chapter 15 of a plant's UFSAR. The analysis is performed using an NRC-approved methodology, such as the Westinghouse analysis methodology using multi-dimensional kinetics (Beard et al., October 2003), for addressing fuel cladding failure thresholds and allowable limits on damaged core coolability. The methodology utilizes an NRC-approved neutron nodal code and an NRC-approved T/H code coupled to pass the necessary data for the nuclear, fluid and fuel temperature calculations. Based on the solution of the coupled code system, the hot rod analysis is performed using the VIPRE-W code with additional conservative assumptions imposed. The analytical inputs, assumptions and calculation methods of the Westinghouse multi-dimensional kinetics methodology for the CRE analysis comply with the guidance in Sections C.2.1 and C.2.2 of RG 1.236.

The evaluation for addressing the allowable limits on the radiological consequences relies on a confirmation of **[**

]^{a,c} Any fuel rod heat flux reaching DNB during the transient is conservatively assumed to fail for the radiological consequence evaluations.

The evaluation for addressing the allowable limits on the RCS pressure is based on the existing method applied to the previous plant evaluation, such as the method described in (Risher, January 1975). The existing method for the plant evaluations on the RCS pressure responses remains applicable to the reactor core containing the high burnup fuel.

5.2 CONTAINMENT INTEGRITY ANAYSES

This section discusses the effect of the incremental increase in peak rod average burnup from 62 GWd/MTU up to **[ ]**[a,c] on the containment integrity analyses. Any impact would be the result of a change in the mass and energy released to containment due to a pipe rupture accident because the containment integrity analyses themselves do not model the fuel. Containment integrity analyses consider the mass and energy released to containment from a LOCA or a steamline break (SLB) event.

5.2.1 Short Term LOCA Mass and Energy Releases

The short term LOCA mass and energy (M&E) releases are used to determine the maximum differential pressure for structural analyses within sub-compartments inside the containment building resulting from postulated pipe ruptures in the primary system piping. These transients are typically performed for 1 to 3 seconds duration and are governed by the mass flux at the break location. Therefore, the parameters that influence the short term LOCA M&E releases are the break location, the corresponding temperature of the fluid in the postulated ruptured pipe, the size of the break, and the initial reactor coolant system pressure. The fuel product and specific aspects of the fuel performance do not influence the short term LOCA M&E. Therefore, any change to the core design to allow for fuel cycle length for 12 months, 18 months, or 24 months at higher incremental burnup levels for an average core would not impact the short term LOCA M&E releases used for short term sub-compartment analyses.

5.2.2 Long Term LOCA Mass and Energy Releases

There are three licensed methodologies currently in use to generate the long term LOCA M&E releases used for long term containment integrity, maximum sump temperature, and equipment qualification analyses for Westinghouse and Combustion Engineering (CE) designs. Those licensed methodologies are:

A. WCAP-10325-P-A (Westinghouse, May 1983)
B. WCAP-17721-P-A (Logan, September 2015)
C. CENPD-132P (CE, August 1974 – March 2001; CE, August 1974 – June 1985; Aerojet, November 1972; and CE, February 1988)

<u>WCAP-10325-P-A Methodology</u>

The core is modeled as an average core for the generation of the long term LOCA M&E releases. There is no hot rod or hot assembly modeled when generating long term LOCA M&E. It is conservative for the long term LOCA M&E to maximize the rate of transfer of energy from the core into the coolant and out of the break. Thus, pellet and cladding interaction and rod burst are not modeled because this would retard the release of the energy stored in the fuel to the coolant and then the break flow. The specific fuel product is modeled with respect to rod inside and outside diameter, flow area through the core, proposed peaking factors, rod initial gas fractions, rod initial internal pressure, theoretical density of the pellet, the material properties of the pellet, the material properties of the cladding material, and the burnup where the highest fuel temperature during the proposed cycle would occur. The licensed LOCA mass and energy release methodology in (Westinghouse, May 1983) does not have any limitations defined with respect to individual rod average burnup. The data that comes from the fuel performance calculations is used as input for the generation of the LOCA mass and energy releases. It is the fuel performance methodology that has the burnup limitation. Therefore, the use of approved fuel performance methods at higher burnups will result in the generation of conservative long term LOCA M&E releases for use in the containment integrity analyses.

The decay heat generated by the core is included in the total energy released to the containment to maximize the long-term containment pressure and temperature response. The decay heat model used in (Westinghouse, May 1983) is the American Nuclear Society ANS-5.1-1979 standard plus 2 sigma uncertainty. The standard provides the flexibility to model a range of burnups up to [

]^{a,c}

Thus, no changes are needed for the (Westinghouse, May 1983) methodology that models an average core for a peak rod average burnup of [ ]^{a,c}

<u>WCAP-17721-P-A Methodology</u>

The methodology approved in (Logan, September 2015) uses the WCOBRA/TRAC (WC/T) code. The initial core stored energy is biased high for the LOCA M&E calculation. [

]^{a,c} Thus, the core is modeled as an average core. The licensed LOCA mass and energy release methodology in (Logan, September 2015) does not have any limitations defined with respect to individual rod average burnup. The data that comes from the fuel performance calculations is used as input for the generation of the LOCA mass and energy releases. It is the fuel performance methodology that has the burnup limitation. [

]^{a,c}

The decay heat generated by the core is included in the total energy released to the containment in order to maximize the long-term containment pressure and temperature response. The decay heat model used in (Logan, September 2015) is the ANS-5.1-1979 standard plus 2-sigma uncertainty. The standard provides the flexibility to model a range of burnups of [

]^{a,c}

Thus, no changes are needed for the (Logan, September 2015) methodology that models an average core for a peak rod average burnup of [ ]^{a,c}

CENPD-132P Methodology

The CE methodology is documented in (CE, August 1974 – March 2001). The CEFLASH-4A computer code is used for the blowdown portion of the transient for both the emergency core cooling system (ECCS) and LOCA M&E calculations. Nominal, cold conditions are the foundation for the fuel dimensions. This approved methodology is based on a hot rod. The fuel temperatures that are used are based on a bounding fuel centerline temperature versus linear heat rate over the entire fuel cycle. No burnup limit is listed for this methodology. The fuel performance data that is used as an input is generated using an approved methodology. It is the fuel performance methodology that has the burnup limitations.

The decay heat generated by the core is included in the total energy released to the containment to maximize the long-term containment pressure and temperature response. The fuel material properties are also an input into the code. Due to the conservatism in the methodology **[**

]^{a,c} no methodology changes will be needed for a full core with higher burnup fuel.

5.2.3 Steamline Break Mass and Energy Releases

The short-term SLB M&E releases are used to determine the short-term pressure increase transients for structural analyses within subcompartments inside or outside the containment building resulting from postulated secondary-side pipe ruptures. These transients are typically performed for 1 to 10 seconds in duration and are governed by the mass flux at the break location. Therefore, the parameters that influence the short-term SLB M&E releases are the break location corresponding to the initial secondary system pressure, temperature and quality of the fluid in the postulated ruptured pipe, and the size of the break. The fuel product and specific aspects of the fuel performance including an increased fuel rod burnup limit do not influence the short-term SLB M&E releases. Therefore, any change affecting the burnup increase does not impact the short-term SLB M&E releases used for short-term subcompartment analyses.

The long-term SLB M&E releases analyses use methods and models that are similar to those discussed for the non-LOCA analyses in Section 5.1. The approved computer codes and analysis methods used to calculate the long-term SLB M&E releases will remain applicable. The impact of the burnup increase will be addressed through input changes to existing analyses.

There are three licensed methodologies currently in use to calculate the long-term SLB M&E releases used for long-term pressure and temperature responses inside containment and long-term temperature response within compartments (steam tunnels or main steam valve vaults) outside containment. The SLB methodologies utilize the following codes to calculate the long-term M&E releases:

A. LOFTRAN (Land, September 1976; Thomas, August 1983; and Osborne and Love, September 1986)
B. RETRAN (Huegel et al., April 1999)
C. SGNIII (CE, January 1974 and US NRC, December 1975)

LOFTRAN and RETRAN Methodologies

The long-term SLB M&E releases safety analyses licensed codes and methods are not tied directly to any specific fuel performance limit or specific fuel design. Therefore, the safety analyses of the long-term SLB M&E releases are not specifically affected by an increase of the maximum fuel rod burnup limit up to **[ ]**^{a,c}. The SLB safety analyses assume bounding reactivity feedback modeling within the licensed computer models to conservatively bound plant operation at the end of core life. Related to the effect of the increased fuel burnup limit on the long-term SLB M&E releases safety analyses,

- there are no changes required in methods to accommodate the increased fuel burnup limit,
- there are no changes in any of the acceptance criteria due to the increased fuel burnup limit,
- there are no licensing or other documentation requiring possible revision and/or NRC approval for the increased fuel burnup limit, and
- there are no tests or analyses required to be performed to support the increased fuel burnup limit.

[

]^{a,c}

SGNIII Methodology

The heat effects in the reactor coolant system such as core stored energy, core to coolant heat transfer, and decay heat tend to maintain the temperature in the reactor coolant system following a steamline break. A wide variation in these parameters, however, has little effect on the rate of energy release from the steam generators. Due to the overall conservatism in the SGNIII methodology, no changes to the methodology are needed when modeling an increased core-wide fuel burnup limit.

5.2.4 Containment Integrity Response

The long term LOCA containment response for the overall peak containment pressure resulting from a LOCA uses the mass and energy releases that are generated for these specific transients as an input. All of the containment codes are executed in this manner. Therefore, the containment codes and methods do not model the peak rod average burnup directly. The containment response codes may generate steam releases due to the decay heat boil-off separately from the long-term LOCA mass and energy releases for long-term equipment qualification considerations well after the peak containment pressure occurs. As was described for the generation of the LOCA mass and energy releases for the three licensed methodologies described in Section 5.2.2, the decay heat energy is based on the ANS-5.1-1979 Standard and the standard is flexible to accommodate peak rod average burnups up to **[ ]**^{a,c} Therefore, no methodology changes will be needed for the containment response models for a full core with higher burnup fuel.

The long-term SLB containment response for the overall peak containment pressure and temperature resulting from a SLB uses the M&E releases that are generated for these specific transients as an input.

All the containment codes are executed in this manner. Therefore, the codes and methods used to analyze the containment response to SLB M&E releases do not model the peak rod average burnup directly. No methodology changes will be needed for the containment response models for a full core with higher burnup fuel.

5.2.5 Conclusions

The short term LOCA mass and energy releases are generated for 1 to 3 seconds. This methodology is not impacted by the burnup level so both methods do not require any changes for peak rod average burnups to **[ ]**^{a,c}

There are three separate approved methodologies for generating long term LOCA mass and energy releases for a containment integrity (aka peak pressure) analysis. There are two methodologies for Westinghouse plants and one for CE plants. All of the methodologies use fuel product specific geometric data and material property data. The core in the two Westinghouse methods are modeled as an average core. The core in the CE method is based on a hot rod model. None of the methods have a burnup limit defined as part of the NRC final safety evaluation. The limitations on burnup are within the methodology that generates and provides the fuel performance data that is used as input for the LOCA mass and energy methodologies and the generation of the decay heat curves. Therefore, the three licensed LOCA M&E methodologies are not limited by the core burnup and therefore do not require any modifications to the computer codes or methodologies if the peak rod average burnup were to be increased from **[ ]**^{a,c}

The short-term SLB M&E releases are generated for 1 to 10 seconds. This calculation is not impacted by the fuel burnup level, and there are no required changes for an increase up to **[ ]**^{a,c} for the peak rod average burnup.

There are three separate approved methodologies for generating long-term SLB M&E releases for a containment integrity pressure and temperature analysis. There are two methodologies for Westinghouse plants and one for CE plants. All the methodologies use decay heat to determine the long-term M&E release rates following a SLB. None of the methods have a fuel burnup limit defined as part of the NRC final safety evaluation. Therefore, the three licensed SLB M&E releases methodologies are not limited by the fuel burnup and therefore do not require any modifications to the computer codes or methodologies to support a peak rod average burnup increase from **[ ]**^{a,c}

The long term LOCA mass and energy releases are used as an input to the containment integrity analyses. There isn't any direct modeling of the fuel in the containment integrity peak pressure analyses. There can be a long term decay heat boil-off steam release that is modeled in the containment codes for long-term equipment qualification purposes via a decay heat curve as an input. Therefore, the containment analysis codes do not require any changes, and they remain valid for incremental burnup fuel designs up to a peak rod average burnup of **[ ]**^{a,c}

The long-term SLB M&E releases are used as an input to the containment integrity analysis. There is no direct modeling of the fuel in the containment integrity pressure and temperature analyses. Therefore, the containment analysis codes do not require any changes, and they remain valid for incremental fuel burnup designs for a peak rod average burnup of up to **[ ]**^{a,c}

5.3 REFERENCES

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6 RADIOLOGICAL CONSEQUENCES ANALYSES

This section discusses the effect of the incremental burnup extension on the radiological consequences analyses for design basis accidents.

The typical design basis events with associated radiological consequence analyses are:

A. Loss-of-Coolant Accident (LOCA)
B. Steam Generator Tube Rupture (SGTR)
C. Main Steamline Break (MSLB)
D. Locked Rotor
E. Rod Ejection
F. Fuel Handling Accident (FHA)

6.1 LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL CONSEQUENCES

The LOCA radiological consequences analyses are not dependent on the emergency core cooling system (ECCS) analysis, which is the subject of Section 4. The LOCA radiological consequences analyses follow regulatory guidance such as that provided in Regulatory Guide (RG) 1.195 (NRC, May 2003) and RG 1.183 (NRC, July 2000). These RGs specify fuel damage assumptions intended to satisfy a footnote to 10 CFR 100.11. The footnote states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

The incremental burnup extension would not significantly impact the fission product releases and associated timing provided in RG 1.195 and RG 1.183. [

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The nuclide inventory contained in the fuel and potentially available for release following a design basis accident is calculated with codes and methods that remain valid for the incremental burnup extension.

Therefore, the codes and methods used in the analyses of LOCA radiological consequences remain valid for the incremental burnup extension.

6.2 STEAM GENERATOR TUBE RUPTURE, MAIN STEAMLINE BREAK, LOCKED ROTOR AND ROD EJECTION RADIOLOGICAL CONSEQUENCES

The radiological consequences analyses of the MSLB, Locked Rotor, and Rod Ejection rely on transient analyses whose potential effects from the incremental burnup extension have already been discussed in Section 5.1 and for which the associated codes and methods remain valid for the incremental burnup extension. The radiological consequences analyses of the SGTR rely on transient analyses with methods and models similar to those discussed for the non-LOCA analyses in Section 5.1 and these too remain valid for the incremental burnup extension. For these accidents, although incremental burnup extension could potentially change the calculated atmospheric mass release results (i.e., steam releases from the steam generators), no codes or methods would have to be changed in order to perform plant specific analysis updates to reflect the change.

The technical specification limits on reactor coolant activity and steam generator secondary coolant activity will continue to ensure that normal operation activity levels are controlled, establishing the initial conditions for these design basis accidents.

The nuclide inventory contained in the fuel and potentially available for release following a design basis accident is calculated with codes and methods that remain valid for the incremental burnup extension.

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Therefore, the codes and methods used in the analyses of MSLB, Locked Rotor, Rod Ejection, and SGTR radiological consequences remain valid for the incremental burnup extension.

6.3 FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

An FHA involving the fuel with the incremental burnup extension cannot be ruled out. [

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The nuclide inventory contained in the fuel and potentially available for release following a design basis accident is calculated with codes and methods that remain valid for the incremental burnup extension.

The FHA radiological consequences analyses follow regulatory guidance such as that provided in RG 1.195 and RG 1.183. This guidance specifies the fractions of the core inventory assumed to be in the gap during steady-state operation (i.e., prior to a postulated accident) for the various radionuclides for rod average burnup up to 62 GWd/MTU and is based on the 1982 American Nuclear Society ANS5.4 standard (ANS, November 1982). The 2011 ANS5.4 standard published in 2011 (ANS, May 2011) will eventually replace the 1982 Standard. The technical basis for the standard is documented in Turnbull and Beyer, January 2010. This standard is limited to rod average burnup of 70 GWd/MTU. The Nuclear Regulatory Commission

(NRC) outlined an acceptable analytical technique for calculating steady-state fission product gap inventories in Appendix B of DG-1327 (US NRC, July 2019). The NRC also updated the generic gap fraction guideline in DG-1327 for rod average burnup up to 65 GWd/MTU. The basis for the guideline is documented in Beyer and Clifford, June 2011 with a revision documented in Clifford, June 2019. Seven [

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It is possible that the impact on the fuel pellets during the FHA could result in fragmentation of the higher burnup fuel and the release of additional fission gases from the fragmented pellets. This would effectively increase the gap fractions relative to those discussed above, and thereby increase the fission product release from the fuel rods with cladding damaged by the impact resulting from the accident.

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Both RG 1.195 and RG 1.183 specify that, in order to account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods. [

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]a,c (Equation 6-1)

The comparison is shown in Table 6-2.

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6.4 CONCLUSIONS

The computer codes and methods currently used in the analyses of radiological consequence of design basis accidents are valid for the incremental burnup extension, [

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6.5 REFERENCES

1. American Nuclear Society, November 1982, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," ANSI/ANS 5.4.

2. American Nuclear Society, May 2011, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," ANSI/ANS 5.4-2011.

3. Beyer, C. E. and Clifford, P. M., June 2011, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," PNNL-18212, Revision 1.

4. [

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5. Clifford, P. M., June 2019, "Revised Technical Basis for Fission Product Release Fractions," Letter from P. M. Clifford (NRC) to R. Lukes (NRC).

6. [

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8. Turnbull, J. A. and Beyer, C. E., January 2010, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," NUREG/CR-7003, PNNL-18490.

9. US NRC, July 2000, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, Revision 0.

10. US NRC, May 2003, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," RG 1.195, Revision 0.

11. US NRC, July 2019, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," DG-1327 (ADAMS Accession Number ML18302A106).

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Table 6-1	**Steady-State Gas Release Fractions**	
Nuclide Group	**RG 1.183 and RG 1.195**	**DG-1327 Recommended Values**
Kr-85	0.10	0.36
I-131	0.08	0.08
I-132	0.05	0.06
Other Nobles	0.05	0.05
Other Halogens	0.05	0.05

Table 6-2	**Effective Gap Release Fractions**		
	[		
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Figure 6-1: Maximum Allowable Power Operating Envelope for Non-LOCA Gap Fractions

7 SUMMARY AND IMPLEMENTATION

In summary, this topical report has described the impact of increasing the fuel rod average burnup limit to **[ ]^{a,c}** on impacted design criteria, codes and methods, and Nuclear Regulatory Commission (NRC)-approved topical reports. Justification for superseding the burnup-related limitations identified in Section 1.4 has been provided. The limitations constraining the application of this topical report are summarized in Section 7.1.

The application of this topical report requires various supplemental analytical work. This is covered in Section 7.2 under the implementation discussion.

7.1 LIMITATIONS FOR APPLICATION OF BURNUP EXTENSION

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 all currently manufactured Westinghouse and Combustion engineering fuel designs. The specific list of applicable designs is provided in Section 7.1.1.

Limitation #2: This topical report is applicable to UO_2 or **ADOPT** fuel with **ZIRLO** cladding or **Optimized ZIRLO** cladding.

Limitation #3: This topical report is applicable to un-poisoned fuel, fuel with integral fuel burnable absorber (IFBA), and fuel with gadolinia. This limitation does not preclude the use of WABA or other discrete burnable absorbers during the lifetime of an assembly.

Limitation #4: The maximum fuel rod length-average burnup and fuel assembly average burnup permitted with this topical report is **[ ]^{a,c}**.

Limitation #5: A maximum of 5 w/o fuel enrichment is permitted with this topical report.

Limitation #6: This topical report is applicable to Westinghouse-designed 2-loop, 3-loop, and 4-loop pressurized water reactors (PWRs) and Combustion Engineering (CE)-designed PWRs. The methodology for the LOCA rupture calculations can only be applied to the Westinghouse 2-loop PWR and CE PWR designs if the **FSLOCA** EM is approved for these designs. No change to the licensing basis LOCA analyses is necessary for Westinghouse-designed 3-loop and 4-loop PWRs.

Limitation #7: PAD5 (Bowman et al., 2017) models and methods must be incorporated into the plant licensing basis for the remainder of the fuel and safety analyses (i.e., all with the exception of LOCA).

7.1.1 Applicable Fuel Designs

Any of the following fuel arrays and skeletons, with or without intermediate flow mixing (IFM) grids, and with or without **PRIME** fuel features.

Westinghouse NSSS

17-OFA

17-V5 / Vantage 5 / Vantage+

17-RFA / RFA-2 / MRFA-2

17-**AP1000** plant

17-XL

17-NGF

16-NGF

15-Upgrade

14-OFA

14-422V+

Combustion Engineering NSSS

16x16 HID1

16x16 System 80

All 16x16 CENGF designs

7.2 IMPLEMENTATION OF BURNUP EXTENSION

Implementation of the incremental burnup extension will require some new calculations, as well as an evaluation of various existing calculations. The implementation of the incremental burnup extension is described in this section.

Fuel Assembly Mechanical Design

Implementation of an incremental increase in the maximum average fuel rod burnup limit requires demonstration that the fuel assembly mechanical design criteria described in Section 2 of this topical report are met.

Fuel Rod Design

Implementation of an incremental increase in the maximum average fuel rod burnup limit requires demonstration that the fuel rod design criteria described in Section 3.1 of this topical report are met.

Loss-of-Coolant Accident Analysis

As discussed in Section 4.2, the LOCA acceptance criteria prescribed in 10 CFR 50.46 **[**

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A demonstration that cladding rupture is precluded for high burnup fuel during a postulated LOCA is required as discussed in Section 4.2. In order to implement the incremental burnup program, it must be demonstrated that any fuel rods in excess of 62 GWd/MTU rod average burnup do not rupture according to the method described in Section 4.7.

Transient Analysis

An evaluation will be performed to confirm that **[**

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As discussed in Section 5.1.2, an analysis of the consequences of the control rod ejection accident will be performed for addressing the acceptance criteria in Regulatory Guide (RG) 1.236, including fuel cladding failure thresholds and allowable limits on damaged core coolability, radiological consequences, and the reactor coolant system (RCS) pressure. The evaluation for addressing the allowable limits on the radiological consequences **[**

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Containment Analysis

The incremental burnup extension does not impact any of the containment analyses.

Nuclear Operations

The decay heat models used for shutdown safety and natural circulation analyses, as well as loss of residual heat removal at mid-loop were found to be conservative for the time period of interest after shutdown. Therefore, the incremental burnup extension does not impact these nuclear operations analyses.

Reload Safety Evaluation

Nuclear designs under the incremental burnup extension are discussed in Section 3.2 of this topical report. All reload limits will be assessed relative to the nuclear designs under the incremental burnup extension to ensure that no reload limits are violated. Any violations would be evaluated prior to implementation.

Additional limitations on the rod and assembly average powers as a function of burnup will be imposed during the reload process such that any fuel in excess of 62 GWd/MTU rod average burnup will operate within the analyzed values.

Radiological Consequence Analyses

The incremental burnup extension does not have any direct impact on the codes and methods used in existing radiological consequence analyses. It has been shown that the LOCA source term for high burnup fuel is similar to lower burnup fuel. For the rod ejection, locked rotor, and steamline break accidents, [

]a,c The reactor specific calculated core nuclide inventory will be assessed to determine any changes, which will then be evaluated to determine impacts on the plant specific radiological consequences results. Any plants that utilize older regulatory guidance in their current radiological consequence analyses of record will need to evaluate their current licensing basis against updated regulatory guidance that is applicable to the higher burnup limit addressed in this topical report.

Vessel Fluence

A limited impact on the vessel fluence calculations would result from the incremental burnup extension. Existing analyses will need to be evaluated or reanalyzed accounting for the elevated fuel average burnup.

Spent Fuel Pool

No significant impact is expected on spent fuel pool criticality analyses since the existing enrichment limit is retained for this topical report, and assembly reactivity decreases with burnup. Spent fuel pool heat removal and time to boil calculations will need to be evaluated or reanalyzed accounting for the higher fuel discharge burnup.

Dry Cask Storage

Dry cask storage is not covered within this topical report, and would have to be addressed separately by utilities and dry cask vendors.

Westinghouse Fuel Criteria Evaluation Process (FCEP)

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7.3 REFERENCES

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

2. Davidson, S. L., October 1994, "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, and WCAP-12488-A, Addendum 1-A, Revision 1, "Addendum 1 to WCAP-12488-A Revision to Design Criteria," January 2002.

3. 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.

APPENDIX A SUMMARY OF HIGH BURNUP FUEL PIE PROGRAM

Table A-1 Summary of PIE Program with Fuel Rod Burnup in Excess of 62 GWd/MTU

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APPENDIX B ADOPT FUEL PELLET-SPECIFIC CONSIDERATIONS

The Westinghouse **ADOPT** fuel pellet technology (Hallman et al., May 2020) has been submitted to the Nuclear Regulatory Commission (NRC) for review and approval. This incremental burnup topical report is applicable to both UO_2 fuel and **ADOPT** fuel pellets as discussed within this topical report. While the discussion within this topical report pertains to both fuel pellet materials, some additional **ADOPT** fuel pellet-specific considerations are described in this appendix.

B.1 MECHANICAL DESIGN

An assessment of the seismic-loss-of-coolant accident (LOCA) calculations for a 17x17 Optimized Fuel Assembly (OFA) design with **ADOPT** fuel pellets was completed. The new models and dynamic features of a fuel assembly with **ADOPT** fuel pellets were compared to the existing 17 OFA models with UO_2 fuel and it was found that the **[** **]**^{a,c} The analysis concludes that the standard 17x17 OFA models **[** **]**^{a,c} can be applied for fuel assembly seismic-LOCA analyses with **ADOPT** fuel pellets.

B.2 FUEL ROD DESIGN

The differences in fuel performance between the **ADOPT** fuel pellets and standard UO_2 fuel pellets are described in (Hallman et al., May 2020). The only differences that require explicit modeling in fuel performance analysis are the slightly increased initial fuel density and reduced rate of fuel densification. Other differences, such as reduction in transient fission gas release (FGR) and higher melting point, represent performance improvements with respect to the fuel rod design criteria of interest at high burnup. No credit is being taken for these improvements in performance within this topical report.

Section 3.1 provides discussion for several high burnup phenomena for UO_2 fuel and justification of applicability of PAD5 for UO_2 fuel to rod average burnup of **[** **]**^{a,c}. These discussions and justifications are largely valid for **ADOPT** fuel pellets. The minor differences are in three high burnup phenomena:

<u>Rim structure and its impact on FGR</u>

The development of rim structure in **ADOPT** fuel will likely be delayed slightly due to larger grain size. Nevertheless, **[** **]**^{a,c} (Section 5.3 of Hallman et al., May 2020).

<u>Potentially enhanced rod growth</u>

As shown in Section 5.2.2 and (Hallman et al., May 2020), **ADOPT** fuel rods tend to have higher growth early on due to earlier pellet cladding contact. The growth for **ADOPT** fuel rods will remain higher at higher burnup, however, as the data presented in that reference show, the difference does not become larger at higher burnups.

Enhanced fission gas swelling

As discussed in Section 4.1.3 of (Hallman et al., May 2020), fission gas in **ADOPT** fuel shows different behavior from in UO_2 fuel. From ramp tests, **ADOPT** fuel has increased porosity within the grain while larger pores in UO_2 fuel were primarily at the grain boundaries. The overall volume change from the ramp tests is smaller for **ADOPT** fuel than for UO_2. Fission gas swelling occurs predominantly in the middle to the center of the pellet where the microstructure of the fuel is less impacted by high burnup rim restructure. Therefore, **ADOPT** fuel is expected to continue showing less volume change from transient or ramp tests at high burnup.

The key design criteria that impact the Westinghouse fuel rod performance are described in Section 3.1 of this report. The methodology to evaluate these criteria for **ADOPT** fuel are the same as UO_2. The difference in the results between **ADOPT** fuel and UO_2 are discussed in Section 6.1.2 of (Hallman et al., May 2020) and the conclusion remains valid for rod average burnup up to **[ ]**^{a,c}.

B.3 SAFETY ANALYSIS

B.3.1 Loss-of-Coolant Accident Analysis Methods

Since higher burnup fuel is not allowed to burst under the incremental burnup extension, there are minimal differences that must be accounted for in performing the cladding rupture calculations for **ADOPT** fuel. The key differences to be addressed are consideration of the material and mechanical properties, differences in the fuel rod conditions at the onset of the transient, and any differences in the potential for fission gas release during the transient.

The different density of the **ADOPT** fuel pellets will be accounted for as described in Section 6.2.1.1.1 of (Hallman et al., May 2020) in the cladding rupture calculations. The other material and mechanical properties for UO_2 fuel remain applicable for **ADOPT** fuel as described in Sections 6.2.1.1.1 and 6.2.1.1.2 therein.

Fuel rod performance data from PAD5 is utilized for the initialization. The fuel rod initialization (e.g., fuel temperatures, rod internal pressures) will account for differences in the **ADOPT** fuel pellets such as the difference in densification. The applicability of PAD5 to calculate the fuel rod performance for higher burnup **ADOPT** fuel pellets is discussed in Section B.2.

The potential for fission gas release during a postulated LOCA for higher burnup rods under the incremental burnup extension was discussed in Section 4.4.5 for UO_2 fuel pellets. **[**

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B.3.2 Transient Analysis Methods

As discussed in Section 6.2.2 of (Hallman et al., May 2020), the non-LOCA codes and methods remain applicable for the **ADOPT** fuel pellet design, and the non-LOCA acceptance criteria also continue to be applicable.

The **ADOPT** fuel pellets are considered equivalent to the UO_2 fuel pellets in terms of the impact on the core stored energy and decay heat. Any modifications required to support the implementation of **ADOPT** fuel pellets for non-LOCA analyses will be made as changes to inputs to existing methods. No new fuel rod failure or accident phenomena are expected due to doping UO_2-based fuel with alumina and chromia. The existing non-LOCA event-related acceptance criteria continue to apply to **ADOPT** fuel pellets. There are no planned changes to the non-LOCA methods to support the implementation of **ADOPT** fuel pellets. The evaluation provided in Section 5.1.1 for the acceptability of the increased fuel rod burnup limit for UO_2-based fuel also applies to **ADOPT** fuel. There is no change in the evaluation method, as described in Section 5.1.2, on addressing the RIA acceptance criteria in Regulatory Guide (RG) 1.236 for a fuel design containing the **ADOPT** fuel pellets.

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B.3.3 Containment Analysis Methods

As discussed in Section 6.2.3 of (Hallman et al., May 2020), the codes and methods used for the various containment analyses are applicable for analysis of **ADOPT** fuel. Therefore, **ADOPT** fuel does not impact any of the discussions in Section 5.2 of this topical report.

B.4 RADIOLOGICAL CONSEQUENCE ANALYSIS METHODS

As discussed in Section 6.2.4 of (Hallman et al., May 2020), the codes and methods used to calculate the radiological consequences of design basis accidents are valid for the **ADOPT** fuel pellet design. The 2011 American Nuclear Society ANS5.4 standard is applicable for UO_2 up to a rod average burnup of 70 GWd/MTU as discussed in Section 6.3. The standards correlate the release of volatile fission gas products to the release of stable fission gas. [

]^{a,c} Therefore, The approach in Section 6 of this topical report for radiological consequences is applicable to **ADOPT** fuel.

B.5 REFERENCES

1. Hallman, L. H., et al., May 2020, "Advanced Doped Pellet Technology (**ADOPT™**) Fuel," WCAP-18482-P.