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Higher Enrichment for Westinghouse and Combustion Engineering Fuel Designs

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EXECUTIVE SUMMARY

Core designs in the nuclear industry are frequently constrained by licensed fuel burnup limits. To achieve higher discharge burnup values, thereby improving fuel cycle efficiency, ^{235}U enrichment higher than the current regulatory limit of 5 percent by weight (wt%) ^{235}U will need to be employed. Certain core designs also benefit from higher fuel enrichments, even if not aiming for a higher discharge burnup, for power uprates or to meet the increased demand for strategically important medical and non-medical radioisotopes production. The purpose of this topical report is to document any required changes in the fuel evaluation methods, and/or provide justification for applicability of existing Nuclear Regulatory Commission (NRC)-approved Westinghouse methods, including nuclear core design, fuel performance, thermal hydraulic design, and Loss-of-Coolant Accident (LOCA) and non-LOCA analyses, to a higher enrichment limit of **[ ]**[a,c]

The Westinghouse higher enrichment topical report will include the following areas:

- **Regulatory Roadmap, including mapping to regulations and regulatory guidance:**
  A review of regulations and regulatory guidance, including Title 10 of the Code of Federal Regulation (10 CFR) Part 50 and NUREG-0800 (Standard Review Plan, SRP) will be performed and applicable requirements will be addressed. Specific SRP 4.2 Acceptance Criteria pertaining to this topical report will address II.1.A (Fuel System Damage); II.1.B (Fuel Rod Failure); II.1.B.iv (Fuel Pellet Centerline Melt); II.1.B.vi (Pellet-Cladding Interaction (PCI) / Cladding Strain); II.1.C (Fuel Coolability for LOCA and non-LOCA scenarios); II.3.C.i (Emergency Core Cooling System (ECCS) Performance Models with respect to Fuel Temperatures); and II.3.C.ix (Fission Product Inventory).

- **Interaction with NRC-approved topical reports:**
  A review of NRC-approved Westinghouse topical reports pertaining to the methods covered in this higher enrichment topical report.

- **Potential licensee implementation actions:**
  Implementation of the higher enriched fuel design will require new design calculations, as well as an evaluation of various existing calculations, including reload safety evaluation, fuel rod design, LOCA, transient analysis, radiological consequence analysis, vessel fluence. Spent fuel pool criticality analysis for higher enriched fuel will be addressed on a licensee-specific basis.

- **Fuel assembly mechanical design:**
  Fuel assembly and structural components design basis and evaluations will be addressed with respect to higher enrichment, including fuel system damage during normal operation, anticipated operational occurrences (AOOs) and postulated accidents, addressing any potential mechanical issues such as structural integrity and dimensional changes.

- **Nuclear design:**
  The existing NRC-approved nuclear design analysis methodology, Westinghouse Reload Safety Evaluation (RSE) and Combustion Engineering Physics Assessment Checklist (PAC) methodologies, will be reviewed and assessed with respect to higher enrichment. Also, the

applicability and impact of higher enrichment on NRC-approved nuclear design codes, PARAGON2/NEXUS/ANC9 will be evaluated. Additional discussion on the impact of higher enrichment on uncertainties and sensitivities will be considered.

- **Thermal-hydraulic design:**
  The existing NRC-approved thermal-hydraulic design evaluation methodology, including Departure from Nucleate Boiling (DNB) correlations, subchannel code, transient fuel rod modeling, DNB propagation evaluation method, and fuel rod bow evaluation methods are evaluated to demonstrate applicability to higher enriched fuel. There is no change in the current thermal-hydraulic design limits for the higher enriched fuel.

- **Fuel rod performance:**
  The NRC-approved PAD5 method will be utilized for fuel rod design. Impact of higher enrichment will be evaluated with respect to fuel performance models and fuel rod design criteria.

- **Safety Analysis:**
  - LOCA:
    Evaluation of small and large break LOCA best-estimate methods, as well as long-term cooling analysis methods with respect to higher enrichment will be provided. The higher enriched fuel is expected to result in an increase in the energy that is released to the containment via higher decay heat energy. Decay heat models extended to higher enrichments will be presented in the topical report.

  - Non-LOCA:
    Existing NRC-approved transient analysis methodologies, including Reactivity Insertion Accidents (RIA), will be evaluated to confirm they are applicable and acceptable to the higher enriched fuel and core design in compliance with the appropriate criteria. A justification will be provided to extend the Regulatory Guide (RG) 1.236 criteria to the higher enriched fuel.

  - Containment Integrity Analysis:
    The effect of higher enriched fuel on the mass and energy (M&E) released to the containment due to a pipe rupture accident will be evaluated. Containment integrity analyses also consider the short-term and long-term M&E released to containment from a LOCA or a steamline break (SLB) event.

- **Fluences/Heat Generation/Sources:**
  Methods used in calculation of core sources, reactor pressure vessel neutron fluence, and reactor internals heat generation rates will be evaluated. In determining core source, the ORIGEN-ARP cross section libraries have been updated for applicability to higher enrichments. The existing approved processes, methods, and codes will be verified and confirmed to have no known deficiencies relative to higher enrichment.

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 and will cover the analytical methodologies described herein. Spent fuel pool criticality impacts will be addressed in plant specific applications. Dry cask storage is not covered within this topical report and would have to be addressed separately by utilities and dry cask vendors.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS and NOMENCLATURE

ANS/ANSI	American Nuclear Society / American National Standards Institute
AOO	anticipated operational occurrence
BOL	beginning-of-life
CE	Combustion Engineering
CHF	critical heat flux
CFR	Code of Federal Regulations
DBA	design-basis accidents
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DRWM	Dynamic Rod Worth Measurement
EM	evaluation model
ECCS	emergency core cooling system
EOL	end-of-life
FCEP	fuel criteria evaluation process
FSLOCA	**FULL SPECTRUM** LOCA
GDC	general design criteria
GEDM	Generalized Energy Deposition Model
IFBA	integral fuel burnable absorber
IFM	intermediate flow mixing
IN	information notice
LOCA	loss-of-coolant accident
M&E	mass and energy
MTC	moderator temperature coefficient
NGF	next generation fuel
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OFA	optimized fuel assembly
PAC	Physics Assessment Checklist
PCI	pellet-cladding interaction
PCT	peak cladding temperature
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
RAI	request for additional information
RCCA	rod cluster control assembly
RCS	reactor coolant system
REA	rod ejection accident (analysis)
RG	regulatory guide
RIA	reactivity insertion (initiated) accident
RR	reformulation-rehomogenization
RSE	Reload Safety Evaluation
RSM	Resonance Scattering Model
RTDP	revised thermal design procedure

ACRONYMS and NOMENCLATURE (continued)

SCICR	Spatially Corrected Inverse Count Rate Method
SE(R)	safety evaluation (report)
SI	spectral index
SLB	steamline break
SRA	stress relief annealed
SRP	standard review plan
T/H	thermal hydraulic
TCD	thermal conductivity degradation
UFEML	Ultra-Fine Energy Mesh Library
US	United States
WC/T	WCOBRA/TRAC
WTDP	Westinghouse thermal design procedure

1 INTRODUCTION AND TOPICAL REPORT ORGANIZATION

1.1 INTRODUCTION

Core designs in the nuclear industry are frequently constrained by licensed fuel burnup limits. To achieve higher discharge burnup values, thereby improve fuel cycle efficiency, ^{235}U enrichment higher than the current regulatory limit of 5 percent by weight (wt%) ^{235}U will need to be employed. Certain core designs also benefit from higher fuel enrichments, even if not aiming higher discharge burnup, for power uprates or to meet the increased demand for strategically important medical and non-medical radioisotopes production. The purpose of this topical report is to document any required changes in the fuel evaluation methods, and/or provide justification for applicability of existing Nuclear Regulatory Commission (NRC)-approved Westinghouse methods, including nuclear core design, fuel performance, thermal hydraulic design, and Loss-of-Coolant Accident (LOCA) and non-LOCA analyses, to a higher enrichment limit of **[ ]**[a,c]

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 and any of the fuel assembly designs, cladding materials, and fuel pellets that are covered by the NRC-approved codes/methods and topical reports that are referenced in this topical report.

1.2 ORGANIZATION OF THE REPORT

Section 1, herein, introduces the topical report.

Section 2 provides an overview and regulatory roadmap of the topical report. Section 2.1 maps the content of the topical report to various available regulatory guidance. Section 2.2 provides a list of approved topical reports and the interaction with a higher enrichment limit under the provisions of this topical report. Considerations for potential licensee implementation actions are described in Section 2.3.

The fuel assembly mechanical design is discussed in Section 3. Impact of higher enrichment is evaluated with respect to fuel assembly design basis and performance parameters including fast fluence and time-at-temperature.

The impact of the higher enrichment fuel on nuclear design codes and methods, including peaking factor uncertainty are evaluated in Section 4.

Section 5 discusses the applicability of the existing thermal-hydraulic methods to analyze fuel designs containing a higher enrichment, evaluating thermal-hydraulic design methods, including subchannel analysis codes and departure from nucleate boiling (DNB) correlations.

Impact of higher enrichment on fuel rod performance is discussed in Section 6. Justification for applicability of the Westinghouse fuel performance code PAD5 is provided along with inputs and models, including pellet radial power distribution, fast flux and fluence, integral fuel burnable absorber (IFBA) helium production, and gap fraction, as well as fuel rod design criteria.

Safety analysis methods, including LOCA and non-LOCA transients, as well as containment integrity analysis are addressed in Section 7. Justifications for applicability of existing methods, and any decay heat and kinetics related changes for the transient methods are provided in this section.

Section 8 discusses the effect of the higher enrichment fuel on the radiation analysis methods which provide core sources and evaluate reactor pressure vessel fluences and reactor internals heating rates for radiological consequences analyses.

Finally, a brief summary is provided in Section 9, followed by limits of applicability of higher enriched fuel.

2 TOPICAL REPORT OVERVIEW AND REGULATORY ROADMAP

2.1 MAPPING TO REGULATIONS AND REGULATORY GUIDANCE

There is no specific part of Title 10 of the Code of Federal Regulations (10 CFR) that imposes a maximum fuel enrichment limit in relation to the methods covered in this topical report. A review of Title 10 was performed to substantiate this conclusion, with particular emphasis on 10 CFR 50.46 and 10 CFR 50 Appendix A General Design Criteria (GDC) 10.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which provides the safety limits that must be maintained by emergency core cooling systems in the event of a LOCA. Those requirements are: (1) peak cladding temperature; (2) maximum cladding oxidation; (3) maximum hydrogen generation; (4) coolable geometry; and (5) long-term cooling.
- General Design Criteria 10, "Reactor design," which states, "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, included the effects of anticipated operational occurrences."

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

There are several regulatory guides and industry documents which can inform the scope of the higher enrichment fuel designs. The documents considered within this topical report are discussed in this section.

2.1.1 Standard Review Plan

To ensure compliance with the regulatory requirements in the GDCs, the guidance provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (US NRC, 2007) is followed.

Although Standard Review Plan (SRP) Section 4.2 has no specific enrichment 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 increase in the ^{235}U enrichment beyond 5 wt%.

Specific SRP 4.2 Acceptance Criteria pertaining to this topical report include the following:

II.1.A (Fuel System Damage) – See Sections 3.2 and 6.1 of this topical report.
II.1.B (Fuel Rod Failure) – See Section 6.1 of this topical report.
II.1.B.iv (Fuel Pellet Centerline Melt) – See Section 6.1 of this topical report.
II.1.B.vi (Pellet-Cladding Interaction (PCI) / Cladding Strain) – See Sections 2.2.1 and 6.1 of this topical report.
II.1.C (Fuel Coolability) – See Section 7.1 of this topical report with respect to LOCA, and Section 7.2.1 of this topical report with respect to transient analysis/ Regulatory Guide (RG) 1.236.
II.2 (Description and Design Drawings) – Higher enrichment pellets do not require new fuel assembly or structural component designs. There are no new descriptions or design drawings in this topical report.
II.3.C.i (Emergency Core Cooling System (ECCS) Performance Models with respect to Fuel Temperatures) – See Section 7.1 of this topical report.

2.1.2 Non-LOCA Safety Analysis Criteria

RG 1.236 (US NRC, 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 approved light water reactor fuel rod designs comprising UO_2 ceramic pellets enriched up to 5 wt% ^{235}U as indicated in Section C.1.1.1 of RG 1.236. Its applicability to the higher enrichment fuel design **[ ]**^{a,c} is discussed in Section 7.2.2.

2.2 INTERACTION WITH NRC-APPROVED TOPICAL REPORTS

2.2.1 Fuel Assembly Design and Cladding Materials

Based on the demonstration to fuel assembly mechanical design criteria in Section 3, the methods described in this topical to implement higher enrichment pellets are applicable to existing Westinghouse fuel designs, including but not limited to the NRC-approved topicals listed below for fuel assembly designs, cladding materials, and fuel pellets.

Fuel Assembly Design Topical Report(s)

- (Davidson and Iorii, 1982) for Optimized Fuel Assemblies (OFA).
- (Davidson and Kramer, 1985) and (Davidson, 1989) in conjunction with Section 3 of the NRC safety evaluation reports (SERs) included in Sections B and F of (Davidson and Ryan, 1995) for VANTAGE 5 and VANTAGE+ fuel assembly.

- (Davidson and Kramer, 1985) for Westinghouse Zircaloy-clad fuel designs.
- Fuel Criteria Evaluation Process (FCEP) (Davidson, 1994, as revised 2002).
- (Davidson et al., 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) for Westinghouse 17x17 next generation fuel (NGF) assemblies.
- (Fiero, 2004) for CE-designed PWR fuel assemblies.
- (Book et al., 2007) for CE 16x16 NGF fuel assembly.
- (Harper, 2022) FCEP Notification of the 17x17 OFA **PRIME™** Fuel Product Implementation.

Cladding Material Topical Report(s)

- (Schueren, 2006), (Hosack, 2019) and (Morey, 2019) for **Optimized ZIRLO™** high performance cladding material for Westinghouse fuel designs and CE fuel designs.
- (Pan et al., 2023) for **AXIOM™** Cladding.

Fuel Pellet Topical Report(s)

- (Hallman et al., 2022) for **ADOPT™** Fuel.

2.2.2 Reload Methodology and Nuclear Design Methods

WCAP-9272-P-A (Bordelon et al., 1985) defines the methodology which is used for plants that have contractual arrangements with Westinghouse for reload designs. The Physics Assessment Checklist (PAC) methodology implements similar methodology for CE nuclear steam supply system (NSSS) plants. The reload safety evaluation (RSE) methodology is a systematic process to confirm that pertinent reload parameters are bounded by the corresponding value used in the reference safety analyses and to perform an evaluation of the effects on the reference safety analysis if a reload parameter is not bounded. Reference safety analyses have been performed using NRC approved analytical methodologies for NRC-approved fuel materials and designs. The reload methodologies do not include conditions or limitations associated with enrichment. Upon NRC approval of this topical report, these reload methodologies may be used to evaluate reloads containing higher enrichment **[ ]**[a,c] using analytical methodologies approved by the NRC.

2.2.3 Thermal-Hydraulic Design Methods

Implementation of the higher enrichment pellets in the current fuel assembly designs does not require modification or update to any existing NRC-approved topical reports for assessing margins to thermal-hydraulic design criteria such as the departure from nucleate boiling ratio (DNBR) limits of Westinghouse and CE fuel designs. Applicability of the existing evaluation methods is discussed in Section 5. Upon approval of this topical report, the existing Westinghouse and CE thermal-hydraulic design methods remain applicable to the enriched fuel pellets as an acceptable fuel material.

2.2.4 Fuel Performance Methods

No changes in fuel performance models are necessary for higher enriched fuel. Applicability of the Westinghouse approved fuel performance code is discussed in Section 6. Upon approval of this topical report, the Westinghouse Fuel Performance and Design model remains applicable to the enriched fuel pellets as an acceptable fuel material.

2.2.5 LOCA Analysis Methods

The nuclear physics data within the WCOBRA/TRAC-TF2 code were updated as described in Section 7.1.2 to extend the validity of the kinetics and decay heat model for fuel rods with greater than 5 wt% ^{235}U enrichment. Models of the neutron capture correction and normalized fission interaction frequency are modified for analysis of fuel rods with higher initial enrichment. Also, it is found that the updated gamma energy redistribution model is valid for the analysis of fuel rods with higher initial enrichment.

Finally, for the purpose of Post-LOCA Long Term Cooling Analysis, the Appendix K decay heat is confirmed to be bounding for higher enrichment fuel rods without degradation of its intended margin and conservatism.

2.2.6 Transient Analysis Methods

Implementation of fuel enrichment higher than the current limit of 5 wt% ^{235}U does not require any modifications to previously NRC-approved topical reports used to analyze non-LOCA analyses. Inputs to existing methods will be developed to incorporate the impacted parameters, such as the decay heat modeling. The current decay heat modeling, as supported in the currently utilized American Nuclear Society (ANS) standards, remains applicable; however, adjustments to the current inputs for decay heat may be required to account for the higher fuel enrichment. Additional discussion of the non-LOCA transient analysis is provided in Section 7.2.

2.2.7 Containment Integrity

No changes in short term or long term LOCA and steamline break (SLB) mass and energy (M&E) release methodologies or containment integrity analysis codes and methodologies are necessary for higher enriched fuel. Applicability of the Westinghouse and CE approved methods are discussed in Section 7.3.

2.3 POTENTIAL LICENSEE IMPLEMENTATION ACTIONS

Implementation of the higher enrichment fuel design will require some new calculations, as well as an evaluation of various existing calculations, as described in this section.

Fuel Assembly Mechanical Design
Implementation of a higher enriched fuel design requires demonstration that the fuel assembly mechanical design criteria described in Section 3 of this topical report are met.

Reload Safety Evaluation
Nuclear designs with higher enrichment fuel designs are discussed in Section 4 of this topical report. All reload limits will be assessed relative to the nuclear designs with higher enrichment fuel to ensure that no reload limits are violated. Any violations will be evaluated prior to implementation.

Fuel Rod Design
Implementation of a higher enriched fuel design requires demonstration that the fuel rod design criteria described in Section 6.2 of this topical report are met.

Loss-of-Coolant Accident Analysis
A demonstration that cladding rupture is precluded for higher enriched fuel design during a postulated LOCA is required with extended decay heat curves.

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

][a,c]

As discussed in Section 7.2.2, an analysis of the consequences of the control rod ejection accident (REA) will be performed for addressing the acceptance criteria in RG 1.236 (US NRC, 2020), including fuel cladding failure thresholds and allowable limits on damaged core coolability, radiological consequences, and the reactor coolant system (RCS) pressure.

Containment Analysis
Extended decay heat values and core stored energy with higher enriched fuel will impact long-term mass and energy releases.

Radiation Analysis
Using an enrichment greater than 5 wt% ^{235}U requires updates to radiation analysis data libraries while the underlying methods remain valid.

2.4 REFERENCES

1. Barsic, J. A., et al., May 2011, “17x17 Next Generation Fuel (17x17 NGF) Reference Core Report,” WCAP-16498-P-A (NRC Safety Evaluation dated March 16, 2011, ADAMS Accession Number (package) ML110610254).

2. Book, M. A., et al., August 2007, “CE 16x16 Next Generation Fuel Core Reference Report,” WCAP-16500-P-A (NRC Safety Evaluation dated July 30, 2007, ADAMS Accession Number ML071920269).

3. Bordelon, F. M., et al., July 1985, “Westinghouse Reload Safety Evaluation Methodology,” WCAP-9272-P-A (NRC Safety Evaluation dated May 28, 1985, ADAMS Accession Number (package) ML19269B536).

4. Davidson, S. L. and Iorii, J. A., May 1982, "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A (NRC Safety Evaluation dated May 22, 1981).

5. Davidson, S. L. and Kramer, W. R., September 1985, "Reference Core Report Vantage 5 Fuel Assembly," WCAP-10444-P-A (NRC Safety Evaluation dated July 18, 1985, ADAMS Accession Number (package) ML20126K599).

6. Davidson, S. L. and Kramer, W. R., December 1985, "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (NRC Safety Evaluation dated October 11, 1985, ADAMS Accession Number (package) ML20133H742).

7. Davidson, S. L. and Ryan, T. L., April 1995, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (NRC Safety Evaluation dated July 1, 1991).

8. Davidson, S. L., February 1989, "Vantage 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2-A (NRC Safety Evaluation dated November 1, 1988).

9. Davidson, S. L., October 1994, "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A (NRC Safety Evaluation dated July 27, 1994) and WCAP-12488-A, Addendum 1-A, Revision 1, "Addendum 1 to WCAP-12488-A Revision to Design Criteria," January 2002.

10. Davidson, S. L., et al., March 1995, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A (NRC Safety Evaluation dated January 30, 1995).

11. Fiero, I. B., August 2004, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," WCAP-16072-P-A (NRC Safety Evaluation dated May 6, 2004, ADAMS Accession Number ML041270102).

12. Hallman, L. H., et al., September 2022, "Westinghouse Advanced Doped Pellet Technology (**ADOPT™**) Fuel," WCAP-18482-P-A (NRC Safety Evaluation dated June 13, 2022, ADAMS Accession Number (package) ML22125A156).

13. Harper Z. S., February 2022, "Fuel Criterion Evaluation Process (FCEP) Notification of the 17x17 OFA **PRIME™** Fuel Product Implementation," (ADAMS Accession Number (package) ML22059B071).

14. Hosack, K. L., February 2019, "Request to Modify Condition and Limitation 2 to WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, 'Optimized ZIRLOTM'," LTR-NRC-19-7, Revision 1 (ADAMS Accession Number ML19039A010).

15. Morey, D. C., May 2019, "Modification of the Condition and Limitation 2 of the U.S. Nuclear Regulatory Commission Safety Evaluation for Westinghouse Electric Company Topical Report WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, '**Optimized ZIRLO™**' (EPID L-2019-TOP-0004)," Letter from D. C. Morey (NRC) to K. Hosack (Westinghouse) (ADAMS Accession Number ML19119A127).

16. Pan G., et al., March 2023, "Westinghouse **AXIOM®** Cladding for Use in Pressurized Water Reactor Fuel," WCAP-18546-P-A (NRC Safety Evaluation dated December 16, 2022, ADAMS Accession Number (package) ML22306A248).

17. Schueren, P., July 2006, "**Optimized ZIRLO™**," WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A (NRC Safety Evaluation dated June 10, 2005, ADAMS Accession Number (package) ML051670395).

18. US NRC, March 2007, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800.

19. US NRC, June 2020, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," RG 1.236.

3 FUEL ASSEMBLY MECHANICAL DESIGN

The SRP, specifically section SRP 4.2 (US NRC, 2007), provides the guidance for demonstrating the acceptability of a fuel design for use in-reactor. 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 burnup limits. The same list of parameters can be used in assessing the impact of fuel enrichments **[ ]**^{a,c} This section provides the criteria and justification needed to demonstrate that the fuel assembly will meet all criteria **[ ]**^{a,c}

Typically, in evaluating fuel assembly design performance parameters which experience change during operation, the main drivers are exposure to fast neutron flux, fluence, and time at operational temperatures.

Higher ^{235}U enrichment would tend to result in slightly less fast neutron fluence to various fuel assembly components. With higher enrichment, more fission is from ^{235}U and less fission is from Pu-239. Fission of Pu-239 emits higher energy fission neutrons than ^{235}U fission. For a core design with higher ^{235}U enrichment with no increase in burnup or power, the higher enrichment does not lead to higher fast neutron fluence.

With unchanged fuel burnup limits, the time and temperature histories would be similar to those the fuel is currently experiencing. The codes and methods used in fuel cycle evaluation are benchmarked to current operation and can, therefore, accurately predict thermal conditions under higher enrichment core designs.

American National Standards Institute (ANSI) standards are based on the GDC from 10 CFR 50 Appendix A. (ANS, 1983) provides definitions of four conditions used in defining design criteria. These are:

Condition I – Normal Operation
Condition II – Incidents of Moderate Frequency
Condition III – Infrequent Incidents
Condition IV – Limiting Faults

With respect to the fuel assembly structure evaluation this includes:

Fuel Assembly Damage (Normal Operation) - This would include shipping and handling and operation in Conditions I and II.

Fuel Assembly Damage (Anticipated Operational Occurrences (AOOs) and Postulated Accidents) - This would include operation in Conditions III and IV. For the fuel assembly this would include seismic and LOCA loads.

Table 3.0-1 provides an overview of the evaluations. More detailed evaluations are provided in Section 3.1 and Section 3.2 for fuel assembly and structural components respectively.

Table 3.0-1: High Energy Core Design Fuel Skeleton Impact Summary			
Parameter	**Criteria**	**Conditions of Evaluation**	**Impact of High Enrichment**
Fuel Assembly Design Bases and Evaluations			
Fuel Assembly Growth	Fuel assembly does not reach solid contact with both core plates.	Normal Operation – Conditions I & II	No impact: Fast neutron fluence is primary driving factor and is not increased with higher enrichment.
Fuel Assembly Hydraulic Stability	Flow through the fuel assembly should not cause wear that exceeds the guideline of **[ ]^{a,c}**	Normal Operation – Conditions I & II	Minimal impact: Core loading patterns would be similar and in-core axial and cross flows would be within the experience base.
Fuel Assembly Structural Integrity	The fuel assembly must maintain its structural integrity during Normal Operation and in response to Seismic & LOCA load.	Normal Operation – Conditions I & II and Seismic & LOCA events	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Operation within database. Fuel assembly mass changes will be minimal with high enrichment.
Fuel Assembly Shipping and Handling Loads	The design acceleration limit for the fuel assembly handling and shipping loads is a minimum of **[ ]^{a,c}**	During shipping and handling	Minimal impact: Fuel assembly mass changes will be minimal with high enrichment.
Fuel Assembly Bow and Rod Cluster Control Assembly (RCCA) Insertion	Limit bow and bow shape to maintain acceptable RCCA drop times.	Normal Operation – Conditions I & II	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Reduction in feed batch size can impact fuel assembly bow.
Fuel Rod Bow	DNBR analysis must account for bounding amount of rod bow. Use correlation of Channel Closure vs Burnup.	Normal Operation – Conditions I & II	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Operation within database.

Table 3.0-1: High Energy Core Design Fuel Skeleton Impact Summary (continued)			
Parameter	**Criteria**	**Conditions of Evaluation**	**Impact of High Enrichment**
Structural Components Design Bases and Evaluations			
Fuel Assembly Holddown Spring	At all operating temperatures, the holddown springs shall provide sufficient force to prevent the fuel assembly from lifting off the bottom core plate.	Normal Operation – Conditions I & II except coolant pump overspeed	Minimal impact: Fast fluence constrained by burnup and peaking limits. Operation within experience database.
Top Nozzle	Designed for **[ ]**^{a,c} shipping and handling loads.	During shipping & handling and Conditions I, II, III, & IV	No impact.
Guide Thimbles and Instrumentation Tube	Stress analyses on the guide thimble tubes show adequate margin on shipping & handling loads.	During shipping & handling and Conditions I, II, III, & IV	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Operation within database.
Grid Assemblies	Must function acceptably under loading limits, not fail due to fatigue and not result in excessive fuel rod wear.	During shipping & handling and Conditions I, II, III, & IV	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Operation within database.
Bottom Nozzle	Designed for **[ ]**^{a,c} shipping and handling loads.	During shipping & handling and Conditions I, II, III, & IV	No impact.
Joints and Connections	Stress analyses and testing on the joints and connection show adequate margin on shipping & handling loads and under all operating conditions.	During shipping & handling and Conditions I, II, III, & IV	Minimal impact: Integrated temperature and fast fluence history constrained by burnup and peaking limits. Operation within database.

3.1 FUEL ASSEMBLY DESIGN BASIS AND EVALUATIONS

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 **[**

]^{a,c} Sizing accounts for the irradiation growth behavior of material used. Fuel assembly growth is an empirical model using fast neutron fluence to predict both best estimate and upper bound growth. As part of core design, the fast neutron fluences will be validated using ANC9 (Zhang et al., 2005) and (Zhang, 2020) which can accommodate enrichments **[** **]**^{a,c} No impact is expected due to use of fuel enrichments **[** **]**^{a,c}

3.1.2 Fuel Assembly Hydraulic Stability

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

Evaluation: Fuel assembly designs are flow tested in the VIPER Loop adjacent to another fuel assembly. Results of these tests confirmed that the projected fuel rod wear due to contact with the mid-grids and intermediate flow mixing (IFM)-grids is well within the guideline of limiting wear to less than **[** **]**^{a,c}

Testing has been performed using the FACTS Loop to confirm the pressure drop characteristics across the entire assembly and individual components and confirming no resonant fuel assembly vibration phenomena is observed in reactor operating flow rates (+/- 15% of best estimate flow). Also, the overall vibration amplitude, for frequencies in the range of 0 - 100 Hz, is less than 2 mils RMS for reactor operating flow rates (+/- 15% of best estimate flow). No impact is expected due to use of fuel enrichments **[** **]**^{a,c}

3.1.3 Fuel Assembly Structural Integrity

3.1.3.1 Condition I & II

Design Basis: The fuel assembly shall maintain dimensional stability when subjected to the loads expected during the life of the fuel assembly (Conditions I and II). These loads shall not result in permanent deformation sufficient to affect the nuclear or the thermal and hydraulic performance of the core. These loads shall not result in any effects that prevent the continued use of the fuel assembly for its design life.

Evaluation: Testing and analysis were performed on the fuel assembly to verify that Condition I & II load requirements were met. There is minimal impact with an increase of enrichment **[ ]^{a,c}** since the overall change in fuel assembly mass will be minimal.

3.1.3.2 Condition III & IV

Design Basis: The fuel assembly must maintain its structural integrity in response to Seismic and 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 on fuel assembly mechanical characteristics and seismic / LOCA analysis results at the end of life (EOL) conditions has been performed to address Information Notice (IN) 2012-09 (US NRC, 2012) subsequently issued by the NRC in 2012. Results for this evaluation are provided in the and Pressurized Water Reactor Owners Group (PWROG) topical report (Lu and Jiang, 2019). For higher enrichments **[ ]^{a,c}** the impact is minimal since there are minimal changes in fast neutron fluence and fuel assembly thermal history.

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 **[ ]^{a,c}**

Evaluation: Testing and analysis were performed on the fuel assembly to verify that shipping and handling load requirements were met. There is minimal impact with an increase of enrichment **[ ]^{a,c}** since the overall change in fuel assembly mass will be minimal.

3.1.5 Fuel Assembly Bow and RCCA Insertion

Design Basis: The guide thimbles provide channels for the insertion of an RCCA 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. 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.

The current 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. There is minimal impact with an increase of enrichment **[** **]^{a,c}** since the overall thermal and fast fluence histories will be similar.

Reduction in feed batch size can impact fuel assembly bow. However, the current fuel assembly designs will be able to meet RCCA drop time for higher enrichments **[** **]^{a,c}**

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 24,000 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. Minimal impact is expected due to use of fuel enrichments **[**
]^{a,c}

3.2 STRUCTURAL COMPONENTS DESIGN BASES AND EVALUATIONS

3.2.1 Fuel Assembly Holddown Spring

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

Evaluation: Hydraulic tests are performed to obtain the necessary inputs to determine the required holddown force. Load vs. deflection testing of the spring packs was completed to determine the actual spring load deflection characteristics. A final verification analysis using standard methodology is performed for plant specific requirements to verify that holddown requirements are met. The main factors impacting the holddown force are the fuel assembly growth and holddown spring relaxation. Minimal impact is expected due to use of fuel enrichments **[** **]^{a,c}** since there will be minimal impact on fuel assembly growth and spring relaxation due to small changes in fast neutron fluence with higher enrichment.

3.2.2 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

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

Evaluation: All current top nozzle designs have been shown to meet all requirements. No impact is expected due to use of fuel enrichments **[ ]**^{a,c} since there will be minimal change to the fuel assembly mass and there will be no increase in either fast or thermal neutron fluences compared to the current 5 wt% ^{235}U enrichment.

3.2.3 Guide Thimbles and Instrumentation Tube

The Guide Thimbles have the following functional requirements:

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 the following functional requirements:

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

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 guide thimble tubes show adequate margin on shipping and handling loads. Minimal impact is expected due to use of fuel enrichments **[ ]**^{a,c} since there will be little change in the temperature history and the fast neutron fluence that the fuel assembly experiences.

3.2.4 Grid Assemblies

The various spacer grid assemblies have the following functional requirements:

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)

Design Basis: The grid design bases are the same as those given in Section 2.3.5.2 of (Davidson and Kramer, 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 grids is based on the extensive design and irradiation experience with previous grid designs and the component testing and analysis completed with each design.

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

The fuel assemblies are flow tested in the VIPER Loop adjacent to another 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 guideline of limiting wear to less than **[ ]^{a,c}**

Minimal impact is expected due to use of fuel enrichments **[ ]^{a,c}** since the thermal history and the fast fluence that the fuel is exposed to will change only slightly.

3.2.5 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

Bottom nozzles are designed for **[ ]^{a,c}** shipping and handling loads.

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

Evaluation: Confirmatory testing is performed to verify the load vs deflection characteristics and the flatness of the bottom nozzle under 4g loading conditions. The test results meet the design requirements. No impact is expected due to use of fuel enrichments **[ ]^{a,c}** since the fast and thermal neutron fluences will not increase as a function of burnup compared to current 5 wt% ^{235}U enrichment limit.

3.2.6 Joints and Connections

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 is performed 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. Minimal impact is expected due to use of fuel enrichments **[ ]**[a,c] since there will be little change in the temperature history and the fast neutron fluence that the fuel assembly experiences.

3.3 OVERALL MECHANICAL DESIGN SUMMARY AND CONCLUSIONS

The use of high enrichment fuel **[ ]**[a,c] will have negligible impact on the performance and margins to criteria of the fuel assembly structure.

3.4 REFERENCES

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4 NUCLEAR DESIGN METHODS

This section of the topical report addresses Westinghouse and Combustion Engineering nuclear design methodologies and code capabilities in the context of higher fuel enrichment and its effect.

4.1 NUCLEAR DESIGN CODES

Westinghouse employs various software packages to perform neutronics calculations to simulate the nuclear reactor core and predict its behavior during normal operation, AOOs and design-basis accidents (DBAs). The following present the PARAGON2/NEXUS/ ANC9 code suite and its applicability for higher fuel pellet enrichment **[ ]^{a,c}**

PARAGON2 (Ouisloumen et al., 2021), a two-dimensional multi-group neutron transport code, is an improvement over its predecessor, PARAGON (Slagle et al., 2004), by using the Ultra-Fine Energy Mesh Library (UFEML) and implementing the Resonance Scattering Model (RSM). PARAGON2 may be executed in a standalone mode or as cross section generation tool for NEXUS/ANC9. The capabilities and performance of PARAGON2 demonstrated in (Ouisloumen et al., 2021) were approved by the NRC for use in higher fuel enrichments **[ ]^{a,c}** for both Westinghouse and Combustion Engineering fuel lattices and for several fuel lattice types, fuel rod diameters, and burnable absorber combinations (See Section 5.2 of (Ouisloumen et al., 2021) for a comprehensive list). Given that PARAGON2 is applicable to enrichments **[ ]^{a,c}** it supports the application in this topical report **[ ]^{a,c}**

The NEXUS methodology (Zhang et al., 2005) parameterizes basic cross section data supplied by a lattice physics code like PARAGON2 into a set of fitting coefficients as a function of various core parameters which are supplied to ANC9 to solve the neutron diffusion equation. The implementation is in two forms: spectral index (SI) and reformulation-rehomogenization (RR). These methodologies were approved by the NRC in (Zhang et al., 2005) and (Zhang, 2020), respectively, for generic use. The use of higher fuel enrichment does not impact the approved methodologies as the neutron diffusion approach was deemed robust in modeling of a variety of unit assemblies, core configurations, and reactor designs based on cross section data provided by PARAGON2. Moreover, the accuracy of the core neutronics calculations is proportional to the accuracy of the cross-section data; the robustness and accuracy of PARAGON2 contributes to the performance of NEXUS/ANC9 through the fitting coefficients and neutronics. Therefore, NEXUS/ANC9 and its methodologies are applicable for use in PWR reactor analyses for cores containing **[ ]^{a,c}**

4.2 NUCLEAR DESIGN METHODOLOGY

The current design methodology approved by the NRC and employed by Westinghouse is the Westinghouse RSE Methodology (Bordelon et al., 1985). Similarly, the PAC methodology for Combustion Engineering is used and approved on a plant specific basis. RSE and the PAC methodology outline the evaluation of nuclear safety by defining a bounding, conservative safety analysis for all AOOs and DBAs. This analysis performs, on a per-cycle basis, a systematic evaluation to determine whether the reload parameters are bounded by the values used in the reference safety analysis and determines the effects on the reference safety analysis when a reload parameter is not bounded to ensure that specified design bases are met. The results of reload safety analysis are concluded to the NRC through the 50.59 evaluation process.

The WCAP-9272 reload safety analysis methodology (Bordelon et al., 1985) and the PAC methodology are not affected by an increase **[ ]**[a,c] enrichment because the methodology is independent of maximum fuel enrichment. It is expected that core reload designs utilizing feed assemblies with higher enrichment will result in different hot channel factors for a given plant, however the safety limits are to be confirmed and appropriate safety margin will be maintained through the reload safety evaluation.

Prior to startup of each cycle, low power physics testing is performed, to ensure that the fuel has been built correctly, the reactor is in its expected configuration, and the safety analysis is conservative compared to measurement. This testing includes comparison of measurements with calculated values for several key parameters. Westinghouse has two licensed techniques for evaluating the measured worth of control rods: Dynamic Rod Worth Measurement (DRWM) (Chao et al., 1998) and Spatially Corrected Inverse Count Rate Method (SCICR) (Sebastiani et al., 2019). Also, Westinghouse has licensed methodologies for conditional relaxation or elimination of the moderator temperature coefficient (MTC) surveillance in (Fetterman, et al., 1993). These methodologies were approved by the NRC and are not affected by the increase **[ ]**[a,c] Neither control rod worth measurement methodology contains fuel enrichment specific input parameters, nor have steps which require changes to accommodate the higher fuel enrichment. These methodologies require the nuclear design to be within a set of acceptance criteria that are independent of maximum fuel enrichment. Therefore, these methodologies remain applicable for use **[ ]**[a,c]

4.3 PEAKING FACTOR UNCERTAINTIES FOR HIGH ENRICHMENT APPLICATIONS

Pursuant of high enrichment for applications in Westinghouse and Combustion Engineering core designs, analytical uncertainties are evaluated to ensure safety and accuracy in the modeling and surveillance of reactors with fuel containing a fuel pellet enrichment **[ ]**[a,c] (Spier et al., 1988) prescribes the peaking factor uncertainties for all neutronics applications to account for differences in plant measurement and core predictions. These peaking factor uncertainties are developed by combining various independent effects which impact the accuracy of the neutronic solution. The three primary components identified in WCAP-7308 (Spier et al., 1988) are the reconstruction of pin powers from assembly quadrant (often called the pin-to-box uncertainty), the reproduction of the radial power distribution, and the reproduction of the axial power distribution. These factors are statistically convoluted to provide a single value for the local maximum peaking factor, F_Q, and the enthalpy rise hot channel factor, $F_{\Delta H}$.

One limiting factor in the evaluation of the peaking factor uncertainties is the lack of measurement data to assess the range of applicability for the uncertainties defined in (Spier et al., 1988). The methodology applied in (Spier et al., June 1998) compares code performance against measurement, computes differences, and applies that value conservatively to all neutronics calculations. Considering this, an approach is taken to assess any change in predictive behavior that may occur when using higher enriched fuel and evaluate how the uncertainties outlined in (Spier et al., 1988) remain applicable to Westinghouse and Combustion Engineering neutronics calculations. However, the lack of measurement data will limit the extension to **[** **]**^{a,c} regardless of the code performance shown in (Ouisloumen et al., 2021), (Zhang et al., 2005), and (Zhang, 2020). **[**

]^{a,c}

The pin-to-assembly normalization or pin-to-box uncertainty captures the uncertainty in normalization and homogenization of pin cell powers to a fuel assembly power and/or vice versa. A series of critical experiments are performed where pin cell reaction rates are recorded experimentally through foil activation, converted to pin powers, and compared to prediction. The uncertainty component is established as the variation in normalized pin powers between measurement and prediction.

A component of the uncertainty is the performance of the pin power reconstruction methodology employed by ANC9 to determine the pin power distribution based on a nodal power distribution of the fuel assembly. In particular, the location of fuel pins within the fuel assembly are of particular interest as the pin power distribution depends on the characteristics (enrichment, burnable poison loading, burnup) of the surrounding assemblies. The power distribution accuracy in ANC9 is assured by the advanced treatment of the fluxes and currents at the assembly interfaces via appropriate application of discontinuity factors along with an advanced pin power reconstruction methodology (Zhang, 2020). The accuracy of the nodal methodology to reproduce pin power generated at the lattice code level is important for high power rods which are of interest in safety criteria such as DNB and fuel melt.

The formulation of the pin-to-box uncertainty is determined by **[**

]^{a,c} (Ouisloumen et al., 2021) demonstrates good accuracy with PARAGON2 for fuel assemblies containing **[** **]**^{a,c} under these conditions for comparing to measurement and benchmarking against higher order codes such as MCNP and SERPENT2. Thus, the advanced ANC9 pin power reconstruction methodology propagates the accuracy of PARAGON2 through its application in these representative core arrangements (Zhang et al., 2005) and (Zhang, 2020).

The radial power distribution uncertainty addresses the differences between measurement and prediction on the assembly power distribution. As the pin-to-box uncertainty captures the intra-assembly comparison, the radial power distribution uncertainty captures the uncertainty between plant measurement and core prediction on a core-wide, per-assembly basis. For higher fuel pellet enrichment applications, the interaction between fuel assemblies plays a major role in capturing the power distribution informing the depletion over core life. For fuel assemblies which have **[** **]**^{a,c}

[

][a,c] NEXUS/ANC9 accounts for

[

][a,c] In SI or RR methodologies, [

][a,c] (Zhang et al., 2005) and (Zhang, 2020). This is demonstrated through the NEXUS/ANC9 qualification process and was approved by the NRC.

The axial component of the uncertainty is assumed to be unchanged from the value determined in the PARAGON2 topical report because of the capabilities of the **BEACON™** Core Monitoring System to accurately predict the axial core behavior. Also, [

][a,c]

4.4 DETECTOR SENSITIVITIES

The introduction of nuclear fuel with greater than 5 wt% ^{235}U enrichment is expected to harden the neutron spectrum. Westinghouse and Combustion Engineering reactors use moveable or fixed flux detectors that predominantly use the thermal neutron flux to infer the power distribution. It is expected that a reactor core that utilizes a higher than 5 wt% ^{235}U enriched fuel would see a reduction in the thermal neutron flux of [ ][a,c]

This reduction in the thermal neutron flux will not impact the performance of the incore detector system to measure power distribution provided the thermal flux is above the minimum measurable range of the detectors. Combined with standard calibration procedures, an increase in the maximum fuel enrichment [ ][a,c] does not significantly impact detector sensitivity. This applies to both movable and fixed detector systems and is applicable to all nuclear reactor detector types.

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5 THERMAL-HYDRAULIC DESIGN

This section discusses the applicability of the existing thermal-hydraulic methods to analyze fuel designs containing a higher enrichment limit of [ ]^{a,c}

5.1 THERMAL-HYDRAULIC DESIGN METHODS

The thermal-hydraulic methods include DNB correlations such as WRB-1 (Motley et al., 1984), WRB-2 (Davidson and Kramer, 1985), WRB-2M (Smith et al., 1999), WSSV (Joffre et al., 2007) and WNG-1 (Joffre et al., 2010), a thermal-hydraulic (T/H) subchannel code such as Westinghouse version of the VIPRE-01 code, referred to as the VIPRE-W code (Sung et al., 1999), and a statistical method for determination of a 95/95 DNBR limit, such as the Revised Thermal Design Procedure (RTDP) (Friedland and Ray, 1989) and the Westinghouse Thermal Design Procedure (WTDP) (Sung and Singh, 2020). Thermal-hydraulic analysis can also be performed as part of the integrated non-LOCA analysis methodology described in (Beard et al., 2003) and (Beard et al., 2006).

The higher enrichment limit of [ ]^{a,c} within the current burnup limit does not require modification or update to any previously NRC-approved methods and topical reports for DNB and thermal-hydraulic analyses noted above. The enrichment increase does not change any fuel rod geometric parameters or characteristics that could adversely affect DNB performance, and the existing DNB correlations remain applicable. Effects of fuel pellet enrichment variations continue to be addressed through engineering hot channel peaking factors (e.g., F^E_Q and $F^E_{\Delta H}$) as input to the thermal-hydraulic design analysis. The VIPRE-W code can perform transient DNBR calculations and non-LOCA post-critical heat flux (CHF) fuel rod transient analysis, based on fuel temperature input from a fuel performance code. There is no change in the VIPRE-W transient modeling method as described in (Sung et al., 1999) for its application to a fuel design containing the higher enriched fuel pellets.

The method using the VIPRE-W code for DNB propagation evaluation, applicable to both Westinghouse and CE PWR plants, is described in (Sidener et al., 2006). The cladding burst model and fuel temperature applicable to the higher enrichment fuel design are input to the DNB propagation evaluation. There is no change in the evaluation method and the conditions for its application to the higher enriched fuel pellets as described in (Sidener et al., 2006).

The impact of fuel rod bowing on DNB is evaluated using an NRC-approved evaluation methodologies for Westinghouse and CE reactor fuel designs, such as those in (Skaritka, 1979), (Gresham, 2011), (Thomas, 1982), (Carew, 1983), and (CE, 1984). There is no change in the existing rod-to-rod gap closure correlations and the current rod bow DNBR penalties as a function of fuel burnup within the current burnup limit because of the higher enrichment fuel pellets.

5.2 CONCLUSIONS

The higher enrichment limit of [ ]^{a,c} within the current fuel burnup limit does not require modification or update to any previously NRC-approved topical reports for assessing margins to thermal-hydraulic design criteria, including the DNBR limit. Upon approval of this topical report, the existing thermal-hydraulic methods will be applied to analyze fuel designs containing the higher enriched fuel pellets.

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16. Thomas, C. O., December 1982, "Acceptance for Referencing of Licensing Topical Report WCAP-8691(P)/WCAP-8692(NP)," Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse).

6 FUEL ROD PERFORMANCE

6.1 FUEL PERFORMANCE INPUTS AND MODELS

Fuel performance analyses is performed using an approved fuel performance code that explicitly accounts for thermal conductivity degradation. PAD5 (Bowman et al., 2017) is the most recent NRC-approved Westinghouse Fuel Performance and Design Model. PAD5 is extended to **ADOPT** fuel in (Hallman et al., 2022). In the SE for the PAD5 topical report, the NRC explicitly limited the enrichment to less than **[ ]^{a,c}** However, higher enrichment is explicitly accounted for in the original PAD5 submittal as the PAD5 fuel performance database contains many fuel rods with enrichments above 5 wt% ^{235}U, such as the BR-3 rods and many rods from test reactors. Because enrichment does not impact the microstructure of the fuel, there is no impact on the mechanism of various fuel performance phenomena and the performance models. Several inputs to fuel performance analysis from neutronics can potentially be impacted and are discussed below.

<u>Pellet Radial Power Distribution</u>
Enrichment has an impact on radial power distribution. The PAD5 has built in default radial power distribution **[ ]^{a,c}** (Section 3.6.2 in (Bowman et al., 2017)). Pellet radial power distribution can also be explicitly modeled through input (Section 7.2.1.4.5 in (Bowman et al., 2017)). The impact on pellet radial power distribution from higher enrichment is explicitly accounted in PAD5.

<u>Fast Flux and Fluence</u>
Fast flux and fluence are input to PAD5 from approved neutronics diffusion codes (Section 7.2.1.4.4 in (Bowman et al., 2017)). While higher enrichment can result in lower neutron flux, there is no change to the methods to generate the fast flux/fluence or uncertainties.

<u>IFBA Helium Production</u>
IFBA helium production model has a dependency on fuel enrichment (Equation 4-5 in (Bowman et al., 2017)). This is an indirect link to the neutron spectrum change with enrichment. This relation is confirmed to be valid for higher enrichment.

<u>Radioactive isotopes (gap fraction)</u>
Enrichment does not impact the microstructure of the fuel and therefore has no impact on the release mechanism. There is minor impact on production of nuclides of interest due to differing amounts of Pu buildup. The impact on gap fraction is negligible.

In summary, no changes in fuel performance models are necessary for higher enriched fuel, and any impact on input to fuel performance analysis are either explicitly accounted or negligible.

6.2 FUEL ROD DESIGN CRITERIA AND SAFETY ANALYSIS INTERFACE

Section 7 of the PAD5 topical report (Bowman et al., 2017) has outlined all the fuel rod design criterion and associated evaluation methodology. The **ADOPT** fuel topical report (Hallman et al., 2022) confirmed the applicability of the PAD5 design criteria and methodology for **ADOPT** fuel. The increased enrichment does not impact fuel failure mechanisms, hence does not change the criteria. The methods described in (Bowman et al., 2017) do not need to be modified for fuel enrichments above 5 wt% ^{235}U. The methods defining safety analysis interfaces remain the same as described in Section 7.5 of (Bowman et al., 2017). Definitions for limiting cases and uncertainty development methods are unaffected.

6.3 REFERENCES

1. Bowman, A. B., et al., November 2017, "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1 (NRC Safety Evaluation dated September 28, 2017, ADAMS Accession Number (package) ML17090A443).
2. Hallman, L. H., et al., September 2022, "Westinghouse Advanced Doped Pellet Technology (**ADOPT™**) Fuel," WCAP-18482-P-A (NRC Safety Evaluation dated June 13, 2022, ADAMS Accession Number (package) ML22125A156).

7 SAFETY ANALYSIS

7.1 LOSS-OF-COOLANT ACCIDENT ANALYSIS

7.1.1 Introduction

LOCA analyses for fuel rods with greater than 5 wt% enrichment will utilize the **FULL SPECTRUM™** LOCA (**FSLOCA™**) evaluation model (EM) (Kobelak et al., 2016). The increase in the fuel enrichment impacts the reactor kinetics and decay heat calculations utilized in LOCA analysis. The updates to the **FSLOCA** EM are discussed in Section 7.1.2. The increased fuel rod enrichment can also impact the decay heat assumed in post-LOCA long-term core cooling analyses. The impact of the higher enrichment for post-LOCA analyses is discussed in Section 7.1.3. Conclusions from both discussions are presented in Section 7.1.4.

7.1.2 FULL SPECTRUM LOCA Methodology

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 enrichment 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 and methods used to perform the neutron kinetics and decay heat calculations for the **FSLOCA** EM extension are discussed herein.

7.1.2.1 Nuclear Physics Data

WCOBRA/TRAC-TF2 explicitly models the burnup and initial enrichment dependence of the reactor kinetics and decay heat via the associated nuclear physics data. It is noted that the initial enrichment limit for the licensed **FSLOCA** EM (Kobelak et al., 2016) and for the incremental burnup extension (Kobelak et al., 2020) is 5 wt% ^{235}U. The decay heat and kinetics package submitted in the **FSLOCA** EM was based on the codes discussed in the response to Request for Additional Information (RAI) #25 in (Kobelak et al., 2016). The nuclear physics data supporting the LOCA calculations for the incremental burnup extension (Kobelak et al., 2020) were originally updated based on the PARAGON code (Slagle et al., 2004), but were later revised in the response to RAI 23.2 per (Harper, 2022) to the PARAGON2 code (Ouisloumen et al., 2021). It was observed that the data from PARAGON2 was qualitatively similar to the data from PARAGON. It is noted that the PARAGON2 code was approved by the NRC for an initial enrichment **[ ]^{a,c}**

The supporting physics data for **FSLOCA** methodology are updated based on PARAGON2, which extends the data to higher enrichments **[ ]^{a,c}** The data produced from PARAGON2 were 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 Figure 7.1-1 through Figure 7.1-14 herein for the updated physics data up to an initial enrichment of **[**

]^{a,c}

7.1.2.2 Neutron Capture Correction

The ANSI/ANS 5.1-1979 standard (ANS, 1979) requires a correction to account for neutron capture. There are two different options provided in the standard to determine the neutron capture correction. One option is to utilize equation 11 in the standard, provided the associated limitations of its use have been addressed. There are three limitations associated with the use of equation 11 of the ANS standard, specifically: A shutdown time of less than 10,000 seconds, a maximum operating time of 4 years, and a number of fissions per initial fissile atom less than 3.0.

The other option is to utilize Table 10 of the standard to determine the correction. The WCOBRA/TRAC-TF2 code approved as part of the **FSLOCA** EM (Kobelak et al., 2016) **[**

]^{a,c}

Equation 11 from ANS standard is as follows:

$$G(t) = 1.0 + (3.24 \cdot 10^{-6} + 5.23 \cdot 10^{-10} \cdot t) \cdot T^{0.4} \cdot \psi \tag{7-1}$$

Where:

G(t) = Neutron capture Correction, dimensionless
t = Time after Shutdown (also referred to as the Cooling Time), seconds
T = Total Operating Time, seconds
Ψ = Fissions per Initial Fissile Atom, dimensionless

[

]^{a,c}

$$G(t) = \Big[\qquad \Big]^{a,c} \tag{7-2}$$

[

]^{a,c} As such, the proposed model is considered acceptable over the desired range of fuel rod enrichments and burnups.

7.1.2.3 Normalized Fission Interaction Frequency

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

With the Nuclear Regulatory Commission's approval of PARAGON2 (Ouisloumen et al., 2021), the normalized fission interaction frequency is updated based on data using PARAGON2 for extension to higher enrichment.

Figure 7.1-18 shows the **[**

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

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

]^{a,c}

$$\Bigg[\qquad \Bigg]^{a,c} \tag{7-3}$$

Where,

[

]^{a,c}

Figure 7.1-19 through Figure 7.1-26 show comparisons of the **[**

]^{a,c}

[**]a,c** As such, the proposed model is considered acceptable over the desired range of conditions.

7.1.2.4 Gamma Energy Redistribution

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

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

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

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

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

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

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

The updated GEDM transfer matrix results for the 15x15 fuel design are presented in Table 7.1-3, that replaces Table 9-12 of (Kobelak et al., 2016). Figures 9-21 and 9-22 of (Kobelak et al., 2016) based on current Table 9-12 clearly illustrate the dependence of the heat flux deposition on both coolant density and relative source strengths. Given that the conclusion remains valid for the updated GEDM transfer matrix, Figures 9-21 and 9-22 of (Kobelak et al., 2016) are not re-created.

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

]a,c

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

7.1.3 Post-LOCA Long Term Cooling

Compliance with criterion (b)(5) of Title 10 of the Code of Federal Regulations Part 50.46 (10 CFR 50.46) ensures that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The methods to address long-term cooling for higher enrichment fuel will utilize the decay heat identified in Appendix K of 10 CFR 50 to determine the heat input from the core. The Appendix K decay heat curve is based on the draft ANS 5.1-1971 standard (ANS, 1971) and assumes infinite irradiation with an initial fueling consisting of ^{235}U as the fissile isotope and ^{238}U as the fertile material. The Appendix K decay energy rate is independent of fuel enrichment and conservative compared to more recent decay heat standards, such as ANS/ANSI 5.1-1979 (ANS, 1979), which account for additional isotopes that are dependent on fuel enrichment. Use of the Appendix K decay heat remains bounding for higher enrichment fuel products without degradation of its intended margin and conservatism.

7.1.4 Conclusions

The nuclear physics data within the WCOBRA/TRAC-TF2 code were updated as described in Section 7.1.2 to extend the validity of the kinetics and decay heat model for fuel rods with greater than 5 wt% enrichment. Models of the neutron capture correction and normalized fission interaction frequency are modified for analysis of fuel rods with higher initial enrichment. Also, it is found that the updated gamma energy redistribution model is valid for the analysis of fuel rods with higher initial enrichment.

Finally, for the purpose of Post-LOCA Long Term Cooling Analysis, the Appendix K decay heat is confirmed to be bounding for higher enrichment fuel rods without degradation of its intended margin and conservatism.

Table 7.1-1: Updated Normalized Fission Interaction Frequency Model Coefficients

a,c

Table 7.1-2: Normalized Fission Interaction Frequency Difference Between Enrichment Values at each Moderator Density

a,c

Table 7.1-3: Typical 15x15 GEDM Gamma Transfer Matrix

a,c

Table 7.1-4: Normalized Gamma Photon Energy Based on PARAGON2

a,c

a,c

Figure 7.1-1: ^{235}U Fission Fraction (Updated Figure 9-1 from (Kobelak et al., 2016))

[ ]a,c

Figure 7.1-2: 239Pu Fission Fraction (Updated Figure 9-2 from (Kobelak et al., 2016))

a,c

Figure 7.1-3: 238U Fission Fraction (Updated Figure 9-3 from (Kobelak et al., 2016))

a,c

Figure 7.1-4: $\overline{\beta}$ versus Burnup (Updated Figure 9-5 from (Kobelak et al., 2016))

a,c

Figure 7.1-5: Prompt Neutron Lifetime (Updated Figure 9-6 from (Kobelak et al., 2016))

a,c

Figure 7.1-6: Prompt Energy Release (Updated Figure 9-7 from (Kobelak et al., 2016))

a,c

Figure 7.1-7: Total Energy Release (Updated Figure 9-8 from (Kobelak et al., 2016))

[ ]a,c

Figure 7.1-8: Delayed Group I Lambda (Updated Figure 9-9 from (Kobelak et al., 2016))

a,c

Figure 7.1-9: Delayed Group II Lambda (Updated Figure 9-10 from (Kobelak et al., 2016))

a,c

Figure 7.1-10: Delayed Group III Lambda (Updated Figure 9-11 from (Kobelak et al., 2016))

a,c

Figure 7.1-11: Delayed Group IV Lambda (Updated Figure 9-12 from (Kobelak et al., 2016))

a,c

Figure 7.1-12: Delayed Group V Lambda (Updated Figure 9-13 from (Kobelak et al., 2016))

a,c

Figure 7.1-13: Delayed Group VI Lambda (Updated Figure 9-14 from (Kobelak et al., 2016))

[ ]a,c

Figure 7.1-14: 238U Capture / Fission Ratio as a Function of Initial Enrichment and Burnup (Updated Figure 9-15 from (Kobelak et al., 2016))

a,c

Figure 7.1-15: Comparison of [ ]^{a,c} Factors

[ ] a,c

Figure 7.1-16: Comparison of [ ]a,c Factors

a,c

Figure 7.1-17: Comparison of [ ]a,c Factors

a,c

Figure 7.1-18: PARAGON2 Normalized Fission Interaction Frequency

a,c

Figure 7.1-19: Comparison of [ ][a,c] and Results of Revised Model

a,c

Figure 7.1-20: Comparison of [ ][a,c] and Results of Revised Model

a,c

Figure 7.1-21: Comparison of [ ][a,c] and Results of Revised Model

a,c

Figure 7.1-22: Comparison of [ ][a,c] and Results of Revised Model

a,c

Figure 7.1-23: Comparison of [ ][a,c] and Results of Revised Model

a,c

Figure 7.1-24: Comparison of [ ][a,c] and Results of Revised Model

[ ]a,c

Figure 7.1-25: Comparison of [ ]a,c and Results of Revised Model

[ ]a,c

Figure 7.1-26: Comparison of [ ]a,c and Results of Revised Model

7.2 NON-LOCA TRANSIENT ANALYSIS

This section discusses the effect of the fuel enrichment higher than the current limit of 5 wt% ^{235}U on the non-LOCA transient analyses.

7.2.1 Transient Analysis

Non-LOCA analyses are performed to demonstrate that the acceptance criteria for the fuel rod failure and coolability are met. No new fuel rod failure or accident phenomena are identified for the fuel enrichment higher than the current limit of 5 wt% ^{235}U.

The impacts on non-LOCA safety analyses are due to:

- Input parameters
- Codes & Methods
- Acceptance Criteria

There are no changes to the non-LOCA topical reports for the analysis codes and methodologies for the fuel enrichment higher than the current limit of 5 wt% ^{235}U, including the analysis methodologies using three-dimensional (3-D) kinetics for the control rod ejection analysis (Beard et al., 2003) and for flow reduction and cooldown event analyses (Beard et al., 2006). For non-LOCA transient analyses, there is no new phenomenon identified to be addressed for higher fuel enrichment. Therefore, there are no new methodology changes and acceptance criteria proposed for the higher enrichment fuel product. There are several assumptions used in the models of the analysis codes and methodologies, such as heat generated in the fuel. These parameters are not significantly impacted by the higher fuel enrichment. These parameters will be confirmed to be valid for the implementation of the higher enrichment fuel product.

The impacted parameters (e.g., decay heat, radial pellet power distributions, cross-sections, fuel conductivity and temperature) are considered inputs to the analyses and will be evaluated on a plant and fuel specific basis.

There are two categories of non-LOCA events that need to be considered with respect to the impact of a change in fuel enrichment:

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 RCS, overpressurization of the secondary system, or overfilling of the pressurizer. Implementation of higher fuel enrichment could potentially change the core average parameters used in the analyses, such as decay heat, radial pellet power distributions, and initial stored energy. The current decay heat modeling remains applicable; however, adjustments to the current inputs for decay heat may be required to account for increase in the fuel enrichment. Therefore, any impact to decay heat will be addressed through input to existing methods. Fuel temperature data from an approved fuel rod design model will be utilized for core stored energy and, any impact will be addressed through input to existing methods. The conservative radial pellet power distribution inputs currently used in the non-LOCA analyses were confirmed to be valid for higher enrichment fuel.

Within 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 minimum DNBR, fuel melting, and peak cladding temperature (PCT). Higher fuel enrichment does not impact these acceptance criteria. Applicability of Regulatory Guide 1.236 (US NRC, 2020) to the higher enriched fuel for the control rod ejection analysis is discussed in Section 7.2.2.

Therefore, the existing non-LOCA acceptance criteria remain applicable to the higher fuel enrichment.

7.2.2 Applicability of Regulatory Guide 1.236 for Rod Ejection Accident with Higher Enriched Fuel

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][a,c]

7.2.3 Conclusion

The existing computer codes and methods used in the analysis of the non-LOCA licensing basis events remain applicable for the higher fuel enrichment. The non-LOCA accident acceptance criteria continue to be applicable for the higher fuel enrichment. The impact of fuel enrichment higher than the current limit of 5 wt% ^{235}U will be addressed via analysis inputs.

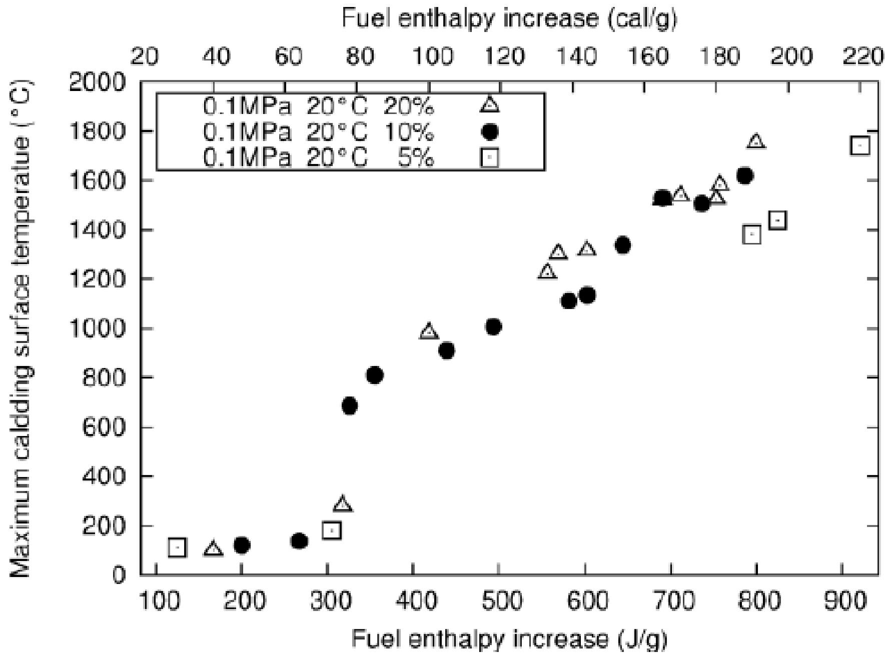

Figure 7.2-1: Maximum Cladding Surface Temperature Measurements for Different Fuel Pellet Enrichments (NEA, 2016)

7.3 CONTAINMENT INTEGRITY ANALYSES

This section discusses the effect of higher enriched fuel on the containment integrity analyses. Any impact would be the result of a change in the M&E 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 SLB event.

7.3.1 Short Term LOCA Mass and Energy Releases

The short-term LOCA M&E release methodology is documented in (Shepard et al., 1975). These LOCA transients are 1 to 3 seconds in 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 corresponding to the 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 releases. Therefore, any change to the fuel enrichment would not impact the short term LOCA M&E releases used for short term subcompartment analyses.

7.3.2 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 for Westinghouse and CE designs. Those licensed methodologies are:

- WCAP-10325-P-A (Westinghouse, 1983)
- WCAP-17721-P-A (Logan, 2015)
- CENPD-132P (CE, 1974 through 2001)

7.3.2.1 WCAP-10325-P-A Methodology

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 releases 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, and the material properties of the cladding material. The licensed LOCA M&E release methodology in (Westinghouse, 1983) does not have any limitations defined with respect to the fuel enrichment. The data that comes from the fuel performance calculations is used as input for the generation of the LOCA M&E releases. It is the fuel performance methodology that has the enrichment limitation. Therefore, the use of approved fuel performance methods at higher enrichments 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, 1983) is created from the ANS/ANSI 5.1-1979 standard (ANS, 1979) plus 2 sigma uncertainty. The standard provides the flexibility to model a range of fuel enrichment. Section 2.4 of (Westinghouse, 1983) lists several assumptions associated with the use of the decay heat standard and these assumptions cover a fuel enrichment **[ ]^{a,c}** The decay heat curve presented in Figure 16 of (Westinghouse, 1983) was created based on maximizing the ^{238}U fission fraction to maximize the decay heat rate and treating the remaining fission fraction as ^{235}U. Increasing the fuel enrichment of ^{235}U will increase the ^{235}U fission fraction and this will reduce the ^{238}U fission fraction. Since the current licensed decay heat curve maximizes the fission fraction from ^{238}U, the current decay heat curve will remain conservative for an increase in enrichment. Therefore, no changes are needed for the (Westinghouse, 1983) licensed methodology for fuel enrichments **[ ]^{a,c}**

7.3.2.2 WCAP-17721-P-A Methodology

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

]^{a,c} Thus, the core is modeled as an average core. The data that comes from the fuel performance calculations is used as input for the generation of the LOCA M&E releases.

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, 2015) is the ANS/ANSI 5.1-1979 (ANS, 1979) standard plus 2-sigma uncertainty. The standard provides the flexibility to model a range of enrichment values. The licensed LOCA M&E release methodology in (Logan, 2015) specifies that **[ ]^{a,c}** should be used to maximize the decay heat curve unless a plant specific enrichment is provided. It is noted that adjustments to the current inputs for decay heat may be required to account for changes in the fuel pellet properties. Therefore, any impact to decay heat will be addressed through input to existing methods.

Thus, no changes are needed for the (Logan, 2015) methodology that models an average core for an increase in fuel enrichment **[ ]^{a,c}**

7.3.2.3 CENPD-132P Methodology

The CE LOCA M&E release methodology is documented in (CE, 1974 through 2001), (CE, 1974 through 1985), (CE, 1988) and (Aerojet Nuclear Company, 1972). 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 approved methodology only references the (ANS, 1971) plus 20% uncertainty without any mention of burnup, enrichment, or plant specific variations or limitations. The decay heat in the containment response code, CONTRANS, documented in (Mitchell, 1976) includes 20% margin. Additionally, in CONTRANS, several decay heat options that include the ANS/ANSI 5.1-1979 (ANS, 1979) standard without any uncertainty are available. The 10 CFR 50 Appendix K decay heat curve also assumes that the reactor has been operating for an infinite amount of time and is initially fueled with ^{235}U as the fissile material and ^{238}U as the fertile material. Further review of both (ANS, 1971) and (ANS, 1979) indicates that assuming ^{235}U as the only fissile material is conservative. The rate of production of fission products at any time after the reactor has been shut down is proportional to the reactor power. Therefore, as long as the licensed thermal power does not increase as a result of the higher enrichment, the 10 CFR 50 Appendix K decay heat is not impacted by increases in the fuel enrichment **[ ]^{a,c}**

Due to the conservatism in the methodology, no methodology changes will be needed for a full core of fuel enriched **[ ]^{a,c}**

7.3.3 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 enrichment do not influence the short-term SLB M&E releases. Therefore, higher enrichment fuel does not impact the short-term SLB M&E releases used for short-term subcompartment analyses.

The long-term SLB M&E release analyses use methods and models that are similar to those discussed for the non-LOCA analyses in Section 7.2. These analyses may be affected by a change in the long-term decay heat parameters that include higher enriched fuel as well as the higher stored energy in the fuel. However, the approved computer codes and analysis methods used to calculate the long-term SLB M&E releases will remain applicable. The impact of higher enrichment 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:

- LOFTRAN (Land, 1976), (Thomas, 1983), and (Osborne and Love, 1986)
- RETRAN (Huegel et al., 1999)
- SGNIII (CE, 1974) and (US NRC, 1975)

7.3.3.1 LOFTRAN and RETRAN Methodologies

The long-term SLB M&E release safety analyses licensed codes and methods are not tied directly to any specific fuel performance limit or specific fuel design. Therefore, the codes and methods related to long-term SLB M&E release safety analyses are not specifically affected by a higher enrichment limit **[ ]**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 a higher enrichment limit on the long-term SLB M&E release safety analyses:

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

[

]a,c

7.3.3.2 SGNIII Methodology

The higher enrichment effects fuel performance parameters such as core stored energy and decay heat which 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.

The approved methodology only references (ANS, 1971) for decay heat without any mention of burnup, enrichment, or plant specific variations or limitations. 10 CFR 50 Appendix K decay heat curve also assumes that the reactor has been operating for an infinite amount of time and is initially fueled with ^{235}U as the fissile material and ^{238}U as the fertile material. Further review of (ANS, 1971) indicates that assuming ^{235}U as the only fissile material is conservative. The rate of production of fission products at any time after the reactor has been shut down is proportional to the reactor power. Therefore, as long as the licensed thermal power does not increase as a result of the higher enrichment, the 10 CFR 50 Appendix K decay heat is not impacted by increases in the fuel enrichment **[ ]**^{a,c}

Due to the overall conservatism in the SGNIII methodology, there will not need to be any changes to the methodology when modeling a higher enrichment limit **[ ]**^{a,c}

7.3.4 Containment Integrity Response

The long-term SLB containment integrity safety analyses licensed codes and methods are not tied directly to any specific fuel performance limit or specific fuel design, including higher fuel enrichment. Long-term SLB containment analyses use the SLB M&E releases as input to determine the resulting containment pressure and temperature to ensure the containment design limits are met. Inputs related to the fuel product and specific aspects of the fuel performance are accounted for within the M&E releases. Therefore, the impact of higher enrichment will be addressed through input changes to existing M&E release analyses.

7.3.5 Conclusions

The short term LOCA M&E releases and subcompartment analyses are generated for 1 to 3 seconds. The short term LOCA M&E release methodology is not impacted by the fuel enrichment level and the subcompartment methodology does not model the fuel so both methods do not require any changes for an increase in fuel enrichment from 5 wt% ^{235}U **[ ]**^{a,c}

There are three separate approved methodologies for generating long term LOCA M&E releases for a containment integrity 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 is modeled as an average core. The core in the CE method is based on a hot rod model. The limitations on enrichment are within the methodology that generates and provides the fuel performance data that is used as input for the LOCA M&E release methodologies and also the generation of the decay heat curves. The decay heat curves in the three licensed LOCA M&E release methodologies are not limited by an increase in the fuel enrichment and therefore do not require any modifications to the computer codes or methodologies if the maximum allowed fuel enrichment were to be increased from 5 wt% ^{235}U **[ ]**^{a,c}

The long term LOCA M&E 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, however there can be a long-term boil-off steam release that is modeled in the containment codes for equipment qualification purposes via a decay heat curve as an input. Since the decay heat curve used to generate the LOCA M&E releases does not require any modifications for a higher enrichment, the containment integrity analysis codes do not require any changes, and they remain valid for an increase in fuel enrichment from 5 wt% ^{235}U **[ ]**^{a,c}

The analyses related to short-term SLB M&E releases do not model fuel related parameters; thus, there are no required changes to account for the higher enrichment limit **[ ]**^{a,c}

There are three separate approved methodologies for generating long-term SLB M&E releases for containment integrity pressure and temperature analyses. 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 and are independent of enrichment. Therefore, no modifications are required to the computer codes or methodologies to support higher enrichment limit **[ ]**^{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 and methods do not require any changes and they remain valid for a higher enrichment limit **[ ]**^{a,c}

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8 RADIATION ANALYSIS

This section discusses the effect of the higher enrichment fuel on the radiation analysis methods which provide core sources and evaluate reactor pressure vessel fluences and reactor internals heating rates for radiological consequences analyses.

8.1 CORE SOURCES

The ORIGEN-ARP cross section libraries were updated for applicability to **[** **]**[a,c] The process used to generate the libraries has been verified, and there are no known deficiencies in the method used relative to higher enrichment and burnup. With the updated libraries, ORIGEN can be used to provide the core fission product inventories needed by the radiological consequence analyses.

8.2 REACTOR PRESSURE VESSEL NEUTRON FLUENCE

Reactor pressure vessel neutron fluence is calculated using the approved methodology in (Fischer and Chen, 2018) or (Andrachek et al., 2004). These methods are consistent with the regulatory guidance for determining pressure vessel neutron fluence in (US NRC, 2001). These methods are not impacted by the presence of fuel enriched beyond 5 wt% ^{235}U, and no modifications to the existing codes or methods are needed.

8.3 REACTOR INTERNALS HEATING RATES

There is no specific regulatory guidance for the methods to be applied for calculating reactor vessel internals heating rates. The methods used have historically been the same as those used for calculating reactor pressure vessel neutron fluence. Reactor pressure vessel neutron fluence is calculated using the approved methodology in (Fischer and Chen, 2018) or (Andrachek et al., 2004). These methods are consistent with the regulatory guidance for determining pressure vessel neutron fluence in (US NRC, 2001). These methods are not impacted by the presence of fuel enriched beyond 5 wt% ^{235}U, and no modifications to the existing codes or methods are needed.

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9 SUMMARY AND LIMITATIONS

9.1 SUMMARY

This topical report has described the impact of increasing the fuel enrichment **[ ]**[a,c] on design criteria, codes and methods, and NRC-approved topical reports. The limitations constraining the application of this topical report are summarized in Section 9.2.

Higher enrichment fuel products may be used in high energy applications, including incremental or extended burnup. While increased burnup applications are outside the scope of this topical report, Westinghouse intends to use Accident Tolerant Fuel (ATF), including chromium-coated cladding for core designs with higher enriched fuel. Applicability and accuracy of Westinghouse codes and methods or the performance of these ATF products are not expected to be sensitive to fuel enrichment.

9.2 LIMITS OF APPLICABILITY OF HIGHER ENRICHMENT

The limitations for the application of the methods described in this topical report are as follows:

Limitation #1: The maximum nominal fuel enrichment permitted with this topical report is **[ ]**[a,c] for NRC-approved Westinghouse and Combustion Engineering fuel designs.

Limitation #2: This topical report is applicable to Westinghouse-designed 2-loop, 3-loop, and 4-loop PWRs and Combustion Engineering-designed PWRs.

Any required changes in the fuel evaluation methods described herein, and/or justifications provided for applicability of existing NRC-approved Westinghouse methods to a higher enrichment are applicable to any of the fuel assembly designs, cladding materials, and fuel pellets that are covered by the NRC-approved codes/methods and topical reports referenced in this topical report.