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LTR-NRC-20-33

May 8, 2020

Subject: Submittal of Topical Report WCAP-18482-P / WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (**ADOPT™**[1]) Fuel" (Proprietary / Non-Proprietary)

Reference: LTR-NRC-20-23 dated March 30, 2020, "Transmittal of the Pre-Submittal Meeting Slides for Topical Report WCAP-18482-P, "Westinghouse Advanced Doped Pellet Technology (**ADOPT™**) Fuel," ADAMS Accession Number ML20090G200

Enclosed are proprietary and non-proprietary versions of Topical Report WCAP-18482-P / WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (**ADOPT™**) Fuel," dated May 2020, submitted for review and approval under the NRC's licensing topical report program for referencing in licensing actions. Approval of this topical report is requested by February 2022, as discussed during the Pre-Submittal Meeting conducted on April 7, 2020 (see the above reference).

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

Correspondence with respect to the proprietary aspects of this submittal or the Westinghouse Affidavit should reference AW-20-5044 and should be addressed to Korey L Hosack, Manager, Licensing, Analysis, and Testing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Suite 133, Cranberry Township, PA 16066.

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

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

[1] **ADOPT™** is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

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

(2) I am requesting that the WCAP-18482-P enclosure to LTR-NRC-20-33 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

AFFIDAVIT

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

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

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

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

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

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

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

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

AFFIDAVIT

refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

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

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

Executed on: 2020 05 07

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

WCAP-18482-NP
Revision 0

May 2020

Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel

Westinghouse

WCAP-18482-NP
Revision 0

Advanced Doped Pellet Technology (ADOPT™) Fuel

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May 2020

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EXECUTIVE SUMMARY

ADOPT™ fuel is a direct replacement for standard uranium dioxide (UO_2) fuel and provides enhanced fuel pellet properties to enable higher burnup and improved accident tolerance. This topical report does not seek to take full advantage of all the benefits of the **ADOPT** fuel. Subsequent licensing submittals will further expand upon the approval of **ADOPT** fuel to more fully exercise all the benefits of the fuel material.

ADOPT fuel is a modified UO_2 pellet doped with small amounts of chromia (Cr_2O_3) and alumina (Al_2O_3). The additives facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2. **ADOPT** fuel is characterized by the nominal inclusion of **[ ]**^{a,c} chromium oxide and **[ ]**^{a,c} of aluminum oxide. It exhibits an increased nominal density of **[ ]**^{a,c} which corresponds to an approximate theoretical density (TD) of **[ ]**^{a,c}, and an average grain size of **[ ]**^{a,c}.

As a result of the higher density and larger grain size, **ADOPT** fuel exhibits:

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

These performance characteristics result in higher burnup capability and enhanced accident tolerance when compared to standard UO_2 pellets; however, as stated above, the specific credit to be taken for these benefits will be the subject of a future supplement to the topical report.

Westinghouse has obtained extensive operating experience with **ADOPT** fuel through its use as a commercial fuel product in Europe. This operating experience is discussed in the topical report and used to characterize the **ADOPT** fuel material properties and performance. The topical report describes in detail how the properties and performance of **ADOPT** fuel is incorporated into existing NRC-approved analytical methods for use in plant-specific safety analyses. The topical report contains a regulatory roadmap in Section 2.1.

The inclusion of dopants does not introduce any new failure modes or phenomena that require new or revised SAFDLs. The topical report establishes generic qualification of the **ADOPT** fuel material, its properties and performance, and the approach for modeling **ADOPT** fuel in safety analysis methods.

ADOPT fuel may be used with all current NRC-approved Westinghouse and Combustion Engineering pressurized water reactor mechanical fuel designs and will be manufactured to the pellet dimensions reflected in the approved fuel design descriptions. **ADOPT** fuel may be used with NRC-approved zirconium-based cladding materials and fuel enrichments.

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ACRONYMNS, ABBREVIATIONS, AND TRADEMARKS

Acronyms and Abbreviations:

ADOPT	Advanced Doped Pellet Technology
ALPS	Advanced LWR Fuel Performance and Safety Research
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BOCA	boiling capsule
BOL	beginning of life
BPVC	Boiler and Pressure Vessel Code
BWR	boiling water reactor
CEA	Commissariat à l'Energie Atomique
CE	Combustion Engineering
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CLB	current licensing basis
COLR	Core Operating Limits Report
DNB	Departure from Nucleate Boiling
DNBR	DNB Ratio
DSC	differential scanning calorimeter
ECCS	emergency core cooling system
EDC	expansion due to compression
EDS	energy dispersive spectroscopy
EM	evaluation model
EOL	end of life
EPMA	electron probe microanalysis
EPRI	Electric Power Research Institute
FA	fuel assembly
FCEP	fuel criteria evaluation process
FFRD	fuel fragmentation, relocation, dispersal
FGR	fission gas release
FHA	fuel handling accident
FRD	fuel rod design
FSLOCA™	**FULL SPECTRUM™** Loss-of-Coolant Accident
FSRM	Fuel Safety Research Meeting
GDC	general design criteria
HBS	high burnup structure
HBWR	Halden Boiling Heavy Water Reactor
HD – UO_2	Undoped UO_2 pellets with density of 10.60 g/cm^3 (96.7% of TD)
IFA	instrumented fuel assemblies
IFBA	integral fuel burnable absorber
ITU	Institute for Transuranium Elements
JAEA	Japan Atomic Energy Agency
[ ]^{d}	**[ ]^{d}**

LBLOCA	large break loss-of-coolant accident
LOCA	loss of coolant accident
LWR	light water reactor
M&E	mass and energy
MOX	mixed oxide fuel
MSLB	main steamline break
NFIR	Nuclear Fuel Industry Research
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research reactor
O/M	oxygen-to-metal ratio
O3	Oskarshamn 3 Reactor
PCI	pellet-clad interaction
PAD	Performance Analysis and Design
PCMI	pellet-clad mechanical interacation
PIE	post irradiation examination
PIRT	phenomena identification and ranking table
pRXA	partially recrystallized annealed
PWR	pressurized water reactor
RAI	Requests for Additional Information
RCS	reactor coolant system
RIA	reactivity-initiated accident
RTDP	Revised Thermal Design Procedure
SAFDL	specified acceptable fuel design limit
SBLOCA	small break loss-of-coolant accident
SCIP	Studsvik Cladding Integrity Program
SE	Safety Evaluation
SEM	scanning electron microscope
SGTR	steam generator tube rupture
SLB	steamline break
SRP	Standard Review Plan
STA	simultaneous thermal analyzer
STS	Standard Technical Specifications
Std pellet	Conventional UO_2 pellet with density of 10.52 g/cm^3 (96.0% of TD)
TD	theoretical density
WDS	wavelength-dispersive spectroscopy
WSE	Westinghouse Electric Sweden
WTDP	Westinghouse Thermal Design Procedure

Trademark Notes:

1 INTRODUCTION

Westinghouse Electric Company has developed Advanced Doped Pellet Technology (**ADOPT™**) fuel to improve performance and enhance the accident tolerance of uranium dioxide (UO_2) fuel pellets. **ADOPT** fuel is a standard UO_2 pellet doped with small amounts of chromia (Cr_2O_3) and alumina (Al_2O_3). The additives facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2. As a result of the higher density and larger grain size, **ADOPT** fuel exhibits: **[**

][a,c] These performance characteristics result in enhanced accident tolerance when compared to standard UO_2 pellets.

1.1 ADOPT FUEL DEFINITION

ADOPT fuel is a modified UO_2 pellet nominally doped with **[ ]**[a,c] chromia (i.e. chromium oxide) and **[ ]**[a,c] of alumina (i.e. aluminum oxide). It is characterized by its increased density of **[ ]**[a,c] which corresponds to an approximate theoretical density (TD) of **[ ]**[a,c]. By comparison, Westinghouse Electric in Västerås, Sweden produces standard UO_2 that is nominally **[ ]**[a,c] or approximately **[ ]**[a,c], and Westinghouse in Columbia, SC produces **[ ]**[a,c] nominally. Additionally, the average grain size, as defined by American Society for Testing and Materials (ASTM) E112, is **[ ]**[a,c] compared to an average grain size of **[ ]**[a,c] for standard UO_2.

1.2 OPERATING EXPERIENCE

Westinghouse Electric Sweden AB in Västerås, Sweden has a long and varied operating experience employing **ADOPT** fuel in the European market. As shown in Table 1-1, Westinghouse has over 20 years of irradiation experience **[ ]**[b,c] burnup and has been delivering reloads for more than 15 years. To date, Westinghouse has delivered more than 600 metric tons of **ADOPT** pellets.

Table 1-1: Deliveries of ADOPT Fuel in the European Market [d,e]

1.3 PURPOSE & CONSTRAINTS

The purpose of this licensing topical report is to provide a detailed description of the **ADOPT** fuel pellets and to describe and characterize the material properties through a review of past operating history and qualification data. The topical report will also review the performance enhancements that **ADOPT** pellets enable and will identify how these enhancements will be incorporated into analytical codes and methods. While the performance improvements inherent to the **ADOPT** fuel are applicable to operation in both pressurized water reactors (PWRs) and boiling water reactors (BWRs), this topical report will focus on application of the **ADOPT** fuel for use with PWRs.

Because of the similar material performance characteristics between **ADOPT** fuel and standard UO_2 fuel, most of the existing analytical methods may be used to analyze **ADOPT** fuel without major modifications. Westinghouse will utilize the most recent Nuclear Regulatory Commission (NRC) approved fuel performance methodology as documented in WCAP-17642-P-A (PAD5) to model the mechanical performance of **ADOPT** fuel. Modifications will be made to existing fuel performance parameters to model the increased density and reduced densification behavior of **ADOPT** fuel.

This topical report focuses specifically on the methods associated with Westinghouse and Combustion Engineering (CE) PWR fuel types. Westinghouse will utilize **ADOPT** fuel within the following constraints:

Reactor & Fuel Assembly Design Constraints

- For use with NRC-approved PWR reactor designs
- For use with NRC-approved Westinghouse and CE fuel designs with corresponding pellet dimensions
- For use with NRC-approved zirconium-based cladding materials

Fuel Design Constraints

- Fuel burnup up to **[** **]**^{a,c} under the following provisions:
  - No rod burst is predicted to occur using an NRC-approved methodology.
  - Additional information is submitted to the NRC and approved for performance of **ADOPT** fuel at higher burnups prior to exceeding a peak rod average burnup of 62 MWd/kgU.
- With or without annular pellets and application of ZrB_2 integral fuel burnable absorber (IFBA) coating consistent with the defined IFBA parameters in applicable NRC-approved fuel performance or product topical reports.
- Nominal pellet density ranging from **[** **]**^{a,c}
- Fuel grain sizes ranging from **[** **]**^{a,c}, as measured according to ASTM E112 as linear intercept without correction factor, which corresponds to **[** **]**^{a,c} with correction
- Inclusion of Cr ranging from **[** **]**^{a,c} which corresponds to inclusion of Cr_2O_3 ranging from **[** **]**^{a,c}
- Inclusion of Al ranging from **[** **]**^{a,c} which corresponds to inclusion of Al_2O_3 ranging from **[** **]**^{a,c}

2 TOPICAL REPORT OVERVIEW AND REGULATORY ROADMAP

This report will demonstrate that **ADOPT** fuel may be used in commercial nuclear reactors in compliance with all applicable regulations. This section provides a regulatory roadmap of the topical report. Section 2.1 maps the content of the topical report to applicable regulatory guidance. Section 2.2 provides a list of current NRC-approved topical reports which will extend their applicability to include the fuel material described herein.

2.1 ROADMAP TO APPLICABLE REGULATORY GUIDANCE

General Design Criteria (GDC) 10, "Reactor Design" in Appendix A of Title 10 of the Code of Federal Regulations (CFR) Part 50 (Ref. 1) states the following:

> *"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, including the effects of anticipated operational occurrences."*

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" (Ref. 2) is followed. Section 4.2 "Fuel System Design," Section 4.3 "Nuclear Design," and Section 4.4 "Thermal and Hydraulic Design" are the sections most pertinent to the performance of fuel rods. Analyses that are performed in accordance with the guidance in SRP Section 6.2.1 and SRP Chapter 15 may also be indirectly impacted by the properties of **ADOPT** pellets.

Of these, SRP Section 4.2 is of primary importance to this topical report and is discussed in more detail in the following subsection. Application of the guidance in SRP Section 4.3 to **ADOPT** fuel is no different than for standard UO_2 pellets since there are no changes required to the neutronics codes and methods for **ADOPT** fuel pellets as discussed in subsection 2.2.4 and Section 6.3. Similarly, use of the guidance in SRP Section 4.4 is no different for **ADOPT** fuel than for standard UO_2 fuel as discussed in subsection 2.2.6 and Section 6.4.

2.1.1 SRP Section 4.2

The guidance of SRP Section 4.2 is established to provide assurance of the following:

(1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs)
(2) fuel system damage during postulated accidents is never so severe as to prevent full control and shutdown rod insertion within the assumed rod drop time when it is required
(3) the number of fuel rod failures is not underestimated for postulated accidents
(4) core coolability is always maintained.

Since this topical report does not describe a change to fuel skeleton, structural materials, or cladding materials, only those acceptance criteria directly impacted by the **ADOPT** pellets are discussed herein. The following sections in this topical report address the acceptance criteria delineated in SRP Section 4.2.

II.1.A – Fuel System Damage – See Section 6.1 of this topical report.
II.1.B – Fuel Rod Failure – See Section 6.1 of this topical report.
II.1.C – Fuel Coolability – See Section 6.1 and Section 6.2 of this topical report.
II.2 – Description and Design Drawings – See Section 1 of this topical report.
II.3.A/B – Operating Experience / Prototype Testing – See Section 1 of this topical report.
II.3.C – Analytical Predictions – See Section 6.2 of this topical report.
II.4 – Testing Inspection and Surveillance Plans – No different than standard UO_2.

2.1.2 SRP Chapter 6.2.1

Section 6.2.1 present information related to containment integrity following postulated loss of coolant accident (LOCA), steam line, or feedline break accidents. The impact of **ADOPT** fuel on the analyses to demonstrate compliance to SRP Section 6.2.1 is discussed in Section 6.2 of this report.

2.1.3 SRP Chapter 15

Section 15.0 identifies high-level acceptance criteria applicable to AOOs and postulated accidents. The remainder of SRP Chapter 15 is split into individual event sections which include more detailed acceptance criteria for each AOO or postulated accident. The impact of **ADOPT** fuel on the acceptance criteria throughout SRP Chapter 15 is addressed in Section 6.2 of this report.

2.2 APPLICATION TO EXISTING NRC-APPROVED TOPICAL REPORTS

Once approved, **ADOPT** fuel will be considered appropriate for use in place of standard UO_2 fuel. This topical report will expand the limits of applicability for existing NRC-approved topical reports to include **ADOPT** fuel. In many cases, existing NRC-approved topical reports do not have specific descriptions of the fuel material composition but were written with the implicit assumption of standard UO_2 fuel. In many cases, no changes are needed to the existing topical reports to ensure compatibility with the **ADOPT** fuel material. In cases where some modification is necessary to incorporate **ADOPT** fuel, this topical report describes the updates necessary to model **ADOPT** fuel and demonstrates why **ADOPT** fuel may be safely utilized in place of standard UO_2 fuel. In all cases, no revisions will be made to the existing NRC-approved topical reports to specifically list **ADOPT** fuel as an approved fuel material. This approach is appropriate since any operating plant that would implement **ADOPT** fuel would need to incorporate this topical report into their licensing basis using an appropriate licensing change process and since the following subsections define the applicability of **ADOPT** fuel to existing NRC-approved topical reports.

2.2.1 Reload Methodology

WCAP-9272-P-A (Ref. 3) defines the methodology which is used for plants that have contractual arrangements with Westinghouse for reload designs. WCAP-16500-P-A (Ref. 8) discusses the reload methodology used for CE-NSSS plants. The reload safety evaluation 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 specific fuel materials. Upon NRC approval of this topical report, these reload methodologies may be used to evaluate reloads containing **ADOPT** fuel using analytical methodologies approved by the NRC.

2.2.2 Westinghouse Fuel Criteria Evaluation Process

WCAP-12488-A and WCAP-12488-A, Addendum 1-A (Ref. 4) defines the NRC-approved fuel criteria evaluation process (FCEP) for Westinghouse NSSS plants. FCEP defines a systematic approach for assessing fuel design changes to determine if prior NRC review and approval is needed before implementing the design change. The NRC approval of the FCEP process notes that it cannot be used to extend applicability of fuel performance models to new materials. This topical report defines all necessary updates to fuel performance and safety analysis analytical modeling for **ADOPT** fuel. Once these updates are approved for use by the NRC, the updated modeling and impact of **ADOPT** fuel will become part of the analysis methods used within the NRC-approved FCEP process. Therefore, upon approval of this topical report, WCAP-12488-A and WCAP-12488-A, Addendum 1-A will be applicable to fuel designs containing **ADOPT** fuel.

2.2.3 Fuel Assembly Designs and Cladding Materials

The current Westinghouse and CE fuel designs are based on standard UO_2 fuel. The fuel assembly topical reports do not include detailed definition of the fuel material composition appropriate for use in the fuel design aside from specifying it as UO_2. In some cases, the fuel design reference reports include definition of the nominal initial fuel pellet density, enrichment, or minimum grain size. Similarly, the NRC-approved **Optimized ZIRLO™** fuel rod cladding topical report (Ref. 5) does not include limitations on the allowable fuel material compositions acceptable for use with the cladding. Supported by the material properties and modeling discussed in this topical report, Westinghouse can appropriately analyze the use of **ADOPT** fuel with all currently approved fuel designs and cladding materials. Upon NRC approval of this topical report, all approved Westinghouse and CE fuel assembly designs and approved cladding materials composed of zirconium-based alloys will include **ADOPT** fuel as an acceptable fuel material within the current enrichment limitations (limited by 10 CFR 50.68(b)(7) and various shipping / transport regulations) and NRC-approved burnup levels.

2.2.4 Nuclear Design Methods

Implementation of **ADOPT** fuel does not require modification or update to any previous NRC-approved topical reports assessing neutronics and nuclear design since the applicable codes already include the dopant materials within the cross-section libraries. The existing topical reports support inclusion of **ADOPT** fuel into the analytical methods without updates to the NRC Safety Evaluations or content of those approved topical reports. Additional discussion of the nuclear design methods is provided in Section 6.3. Upon approval of this topical report, the existing Westinghouse and CE nuclear design methods will include **ADOPT** fuel as an acceptable fuel material.

2.2.5 Fuel Performance Methods

Implementation of **ADOPT** fuel will be performed using PAD5, the most recent NRC-approved Westinghouse fuel performance and design model, documented in WCAP-17642-P-A, Rev. 1 (Ref. 6). In the Safety Evaluation (SE) for the PAD5 topical report, the NRC included the following Conditions and Limitations that would require modification to accommodate the **ADOPT** fuel:

- Fuel grain sizes ranging from **[ ]**^{a,c}

Upon submittal of this topical report, the Condition and Limitation noted above will be requested as follows for **ADOPT** fuel.

- Fuel grain sizes ranging from **[ ]**^{a,c} for **ADOPT** fuel

As noted previously, it is not necessary to revise the PAD5 topical report to incorporate this revised limitation and condition since this limitation and condition will only be applicable for analyses performed on **ADOPT** fuel and will be reflected in this topical report, which will be incorporated into the plant's licensing basis with implementation of **ADOPT** fuel. Additional details of updates to the fuel performance and design model (PAD5) are discussed in further detail in Section 6.1.1. This discussion does not impose a limitation or restriction on the use of future NRC-approved fuel performance methods in place of PAD5 or supplements to PAD5.

2.2.6 Thermal-Hydraulic Design Methods

Implementation of **ADOPT** fuel does not require modification or update to any existing NRC-approved topical reports assessing thermal-hydraulic performance. Although the existing methods have been developed based on standard UO_2 pellets, the similarities between standard UO_2 fuel and **ADOPT** fuel will support inclusion of **ADOPT** fuel into those analytical methods without updates to the NRC Safety Evaluations or content of those approved topical reports. Applicability of the existing evaluation methods is discussed in Section 6.4. Upon approval of this topical report, the existing thermal-hydraulic design methods remain applicable to **ADOPT** fuel as an acceptable fuel material. Applicability of the existing thermal-hydraulic design methods is further discussed in Section 6.4.

2.2.7 Safety Analysis Methods

Westinghouse maintains many different NRC-approved methods for performing safety analyses in support of Chapter 15 of a plant's updated final safety analysis report. These methodologies are separated into different categories as presented below.

2.2.7.1 LOCA

Implementation of **ADOPT** fuel does not require modification or update to any previously NRC-approved methodology used to analyze LOCA behavior. Although the existing methods have been developed based on standard UO_2 pellets, the similarities between standard UO_2 fuel and **ADOPT** fuel will support inclusion of **ADOPT** fuel into those analytical methods without updates to the NRC Safety Evaluations or content of those approved topical reports. Applicability of the existing evaluation methods is discussed in Section 6.2.1. Upon approval of this topical report, the existing Westinghouse LOCA methods described herein will include **ADOPT** fuel as an acceptable fuel material.

2.2.7.2 Non-LOCA Transient Analyses

The implementation of **ADOPT** fuel pellets 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 increased fuel density, and its associated effect on fuel thermal conductivity. Additional discussion of the non-LOCA transient analyses is provided in Section 6.2.2. Upon approval of this topical report, the existing Westinghouse and CE non-LOCA transient analysis methods will include **ADOPT** fuel as an acceptable fuel material.

2.2.7.3 Containment Integrity Analyses

Implementation of **ADOPT** fuel does not require modification or update to any previously NRC-approved topical reports used to analyze LOCA and steamline break mass and energy releases and containment response. In general, the applied methodologies are insensitive to the fuel material and the similarities between **ADOPT** fuel and standard UO_2 fuel will allow incorporation of **ADOPT** fuel into the methods without making updates to the NRC Safety Evaluations or content of those approved topical reports. Additional discussion of the containment integrity analysis methods is provided in subsection 6.2.3. Upon approval of this topical report, the existing Westinghouse containment integrity analysis methods will include **ADOPT** fuel as an acceptable fuel material.

2.2.7.4 Radiological Consequences Analyses

Westinghouse does not maintain any NRC-approved methodologies for performing radiological consequences analyses as the analyses are performed in accordance with published Regulatory Guidance. Applicability of **ADOPT** fuel to the assumptions identified in NRC Regulatory Guidance is discussed further in subsection 6.2.4.

2.3 ANTICIPATED LICENSEE IMPLEMENTATION ACTIONS

A license amendment request is required. The exact nature of the required Technical Specification markups will depend upon the current licensing basis (CLB) of the plant. The following list applies to plants that have converted to Improved Technical Specifications based on the Standard Technical Specifications (STS) of NUREG-1431:

- Implementation of PAD5 (Reference 6) requires a change to the burnup-dependent Safety Limit on peak fuel centerline temperature (STS 2.1.1.2) which shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWd/kgU of burnup. If this burnup dependency is discussed in the plant Technical Specification Bases for Safety Limit 2.1.1, corresponding Bases changes will also be required. Markups to the UFSAR Reactor chapter will be required under the plant process for updates pursuant to 10 CFR 50.71(e).

- The COLR list of references should be revised to add WCAP-18482-P since this topical report will be a standalone document and not result in revisions to any other topical reports.

- Implementation of **FSLOCA** EM (Reference 7) requires a change to the COLR list of references (STS 5.6.3.b) to add WCAP-16996-P-A Revision 1 and delete outdated methods (such as BASH,

BELOCA, and ASTRUM). Technical Specification Bases changes and UFSAR updates are also required for **FSLOCA** EM.

- A License Condition may be imposed such that the peak rod average burnup for **ADOPT** fuel will not exceed 62 MWd/kgU until additional information has been reviewed and approved by the NRC.

- Spent and new fuel criticality analyses are not addressed in this topical report, but may require licensee implementation action, as applicable, based on their CLB.

2.4 CHAPTER 2 REFERENCES

1. U.S. NRC. 10 CFR Part 50: Appendix A. "General Design Criteria for Nuclear Power Plants."
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3 CHARACTERIZATION OF ADOPT FUEL PROPERTIES

3.1 MICROSTRUCTURE

3.1.1 Additives

Doping UO_2 with chromia enhances the mobility of UO_2 at high temperatures and adding small amounts of alumina intensifies the effect. Although dopants play a significant role in promoting densification and grain growth, it should be noted that there are several other manufacturing characteristics affecting densification and growth, such as the sintering time, sintering temperature, and the oxygen potential of the sintering atmosphere.

Alumina

In stoichiometric and hyperstoichiometric UO_2, Al_2O_3 is largely insoluble and should inhibit grain growth, since the oxide is present as an intergranular precipitate. However, above the 70 ppm solubility limit, it is shown to act as a grain growth promoter. The excess Al_2O_3 precipitates to form secondary phase inclusions within the UO_2 matrix. At high temperatures, like those experienced during sintering, these inclusions coalesce by an Ostwald ripening phenomenon. An AlO eutectic is formed enhancing diffusion of uranium; **[**

]^{b,c}

Chromia

[

]^{b,c} Cooper et al. have performed several atomistic simulations to determine the defect chemistry of chromia in UO_2. Their findings suggest a preference of the interstitial site in UO_2 for low valence cations, like Cr, which can readily access the 1+ and 2+ charge state. As a result, there is an increased concentration of negatively charged uranium vacancies promoting uranium diffusivity (Ref. 3).

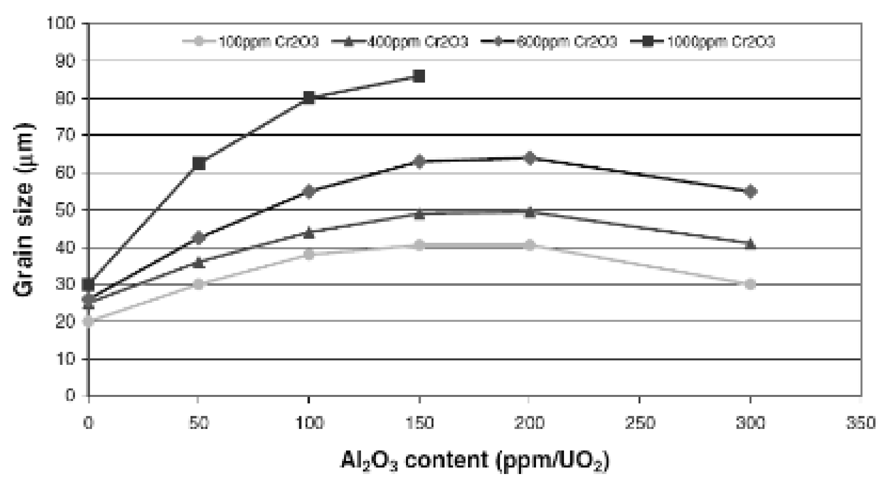


Figure 3-1 The variation of grain size with alumina and chromia content (Ref. 1)

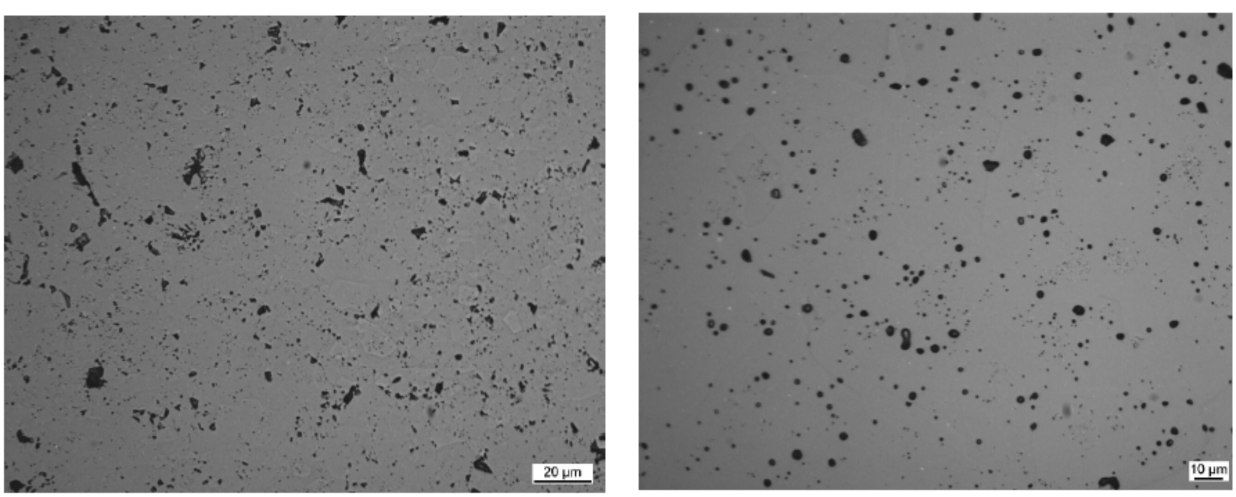
3.1.2 Microstructure

As discussed in Section 3.1.1, chromia and alumina enhance grain growth during sintering, resulting in a significantly larger grain size, approximately 3-5 times larger than conventional UO_2 (see Figure 3-2). As a result, **ADOPT** fuel is characterized by a stable microstructure, experiencing minimal in-pile grain growth and densification. More information on thermal stability testing and densification is discussed in Section 5.2.

b,c

*Figure 3-2 Grain size comparison of standard UO_2 (left) and **ADOPT** pellets (right)*

The additives also have an influence on the pore shape. The pores of undoped UO_2 are irregular (elliptical and triangular) as consequence of typically being located at the grain boundaries and being easily influenced by grain boundary energy; **ADOPT** fuel has a round pore shape, as a result of typically being trapped within the grain, as shown in Figure 3-3 below (Ref. 2). Note that the scales are not the same in the two images.



UO_2 pellet **ADOPT pellet**

Figure 3-3 Pore shape at center of as-manufactured pellet (Ref. 2)

3.1.3 Alumina and Chromia Residence in Pellet

3.1.3.1 Wavelength dispersive spectrometry

The radial concentration of aluminum and chromium, in an unirradiated pellet, has been measured using wavelength dispersive spectrometry (WDS). The examinations reveal a large statistical variation, which is typical for this method; **[**

]b,c

[ ]b,c

Figure 3-4 Line scan of the first 500 microns into the ***ADOPT*** *pellet from the surface.*

3.1.3.2 Micrograph and Energy Dispersive (X-ray) Spectrometry (EDS)

ADOPT material has been studied at the center (mid-axial position) and in the outer surface position (mid-axial position), as shown in Figure 3-5 and Figure 3-6. These examinations [

]b,c

[ ]b,c

Figure 3-5 Micrograph of ***ADOPT*** *pellet at center position (1000x and 5000x)*

[ ]b,c

Figure 3-6 Micrograph of ***ADOPT*** *pellet at periphery position (1000x and 5000x)*

The elemental composition of the inclusions has been measured using Energy Dispersive (X-ray) Spectrometry (EDS) at the center and periphery positions. These particles [

]b,c

3.1.4 Theoretical Density

Uranium dioxide, an ionic solid composed of a positively charged U^{4+} cation and negatively charged O^{2-} anion, crystallizes in the fluorite structure. Within each unit cell, four uranium and eight oxygen atoms are present. The theoretical density can be calculated from the crystal structure and lattice parameter, as demonstrated below. This value is calculated to be approximately 10.96 g/cm^3.

$$mass\ of\ unit\ cell\ (g) = \frac{4 \times m_{UO_2}}{N_A} = 1.79 \times 10^{-21}$$

$$theoretical\ density\ \left(\frac{g}{cm^3}\right) = \frac{1.79 \times 10^{-21}}{{a_o}^3} = 10.96$$

Where:

N_A = Avogadro's constant
a_o = lattice constant
m_{UO_2} = molecular mass of UO_2

However, the addition of **[ ]a,c** chromia and **[ ]a,c** alumina (by mass) will slightly alter the lattice parameter, as well as the theoretical density. As such, determination of the theoretical density from the unit cell size is not directly possible.

It is acknowledged that **ADOPT** fuel is not a composite material, and the rule of mixtures is not directly applicable. However, at these minimal dopant levels, it can be used to estimate the theoretical density:

$$\rho_{mixture}\left(\frac{g}{cm^3}\right) = \frac{M_{UO_2} + M_{Al_2O_3} + M_{Cr_2O_3}}{\frac{M_{UO_2}}{\rho_{UO_2}} + \frac{M_{Al_2O_3}}{\rho_{Al_2O_3}} + \frac{M_{Cr_2O_3}}{\rho_{Cr_2O_3}}} = [\qquad]^{a,c}$$

Where,

M = mass fraction of UO_2, Al_2O_3, or Cr_2O_3
ρ = density of UO_2, Al_2O_3, or Cr_2O_3

Doing so, gives a density of **[ ]a,c** In other words, the additives have minimal effect on the overall density of **ADOPT** fuel. Therefore, it is reasonable to assume that the 100% TD of **ADOPT** fuel is that of UO_2.

3.2 THERMAL PROPERTIES

3.2.1 Specific Heat

The specific heat capacity of the fuel is fundamental to determining the stored energy in the fuel rod, which is especially important in transient analysis. Additionally, the specific heat is used to calculate the thermal conductivity.

The specific heat was determined using the differential scanning calorimeter (DSC) technique at the Institute for Transuranium Elements (ITU) in Karlsruhe, Germany using a Netzsch simultaneous thermal analyzer (STA) 409C. DSC measures the difference in heat input into a sample and a reference material, as a function of temperature. The tests were performed with flowing argon with a rate of 0.1 l/min and a temperature ramp rate of 25 K/min. In each study, sapphire was used as the reference material. This technique is most accurate in the temperature range of 400 – 1400 K.

Two unirradiated **ADOPT** samples were analyzed and compared with two pure, unirradiated std UO_2 samples. The results are presented in Figure 3-7. The measurements revealed small differences between the **ADOPT** and reference UO_2 pellets for the temperature range up to 1200°C. In particular, the largest variability is shown at the beginning and end of the test, where the DSC technique is least accurate; however, the results are within the precision of the technique. Therefore, it is reasonable to conclude that there is no appreciable difference in the specific heat of UO_2 and **ADOPT** fuel (Refs. 1, 2).

b,c

*Figure 3-7 Specific heat capacity measurements for **ADOPT** and reference UO_2 fuel*

3.2.2 Thermal Diffusivity (and Thermal Conductivity)

Thermal conductivity is an important material property that is used to determine the temperature distribution within the fuel rod. It is typically determined indirectly by measuring the thermal diffusivity of a material using the laser flash technique. This method involves irradiating the top surface of a sample material with a laser, providing an instantaneous energy pulse. Simultaneously, a photovoltaic infra-red detector monitors the temperature rise of the bottom surface of the sample. The thermal diffusivity can then be calculated by using the following equation:

$$\alpha = \frac{k \cdot L^2}{t_{1/2}}$$

where: k is a constant; L is thickness of the specimen in cm; and $t_{1/2}$ is the time for the back face of the sample to reach half of its maximum temperature rise in seconds.

If the specific heat (C_p) and density (ρ) of a material at a particular temperature are known, the thermal conductivity (λ) of that material can be calculated from thermal diffusivity (α) using the following equation:

$$\lambda = \alpha \cdot \rho \cdot C_p$$

The thermal diffusivity of **ADOPT** fuel was measured at the KTH Royal Institute of Technology in 1999 using a ULVAC Sinku-Riko TC-7000H/MELT. Measurements were made on two separate unirradiated **ADOPT** samples and compared with an unirradiated standard UO_2 sample. Measurements were taken between 20°C and 1400°C in approximately 100°C increments during heating; up to four measurements were performed during cooling. The sample thickness was corrected for thermal expansion; a maximum correction of approximately 2.5% was made for values at 1400°C (Refs. 1, 2). The results are presented in Figure 3-8 below.

b,c

Figure 3-8 Thermal diffusivity measurements of ***ADOPT*** *and reference UO_2 fuel*

The thermal diffusivity analysis reveals an insignificant difference between **ADOPT** and standard UO_2 fuel. It is reasonable to conclude that there is no appreciable difference in the thermal diffusivity of conventional UO_2 and **ADOPT** fuel.

3.2.3 Melting Temperature

The melting temperature of **ADOPT** fuel was determined using a laser-pulse melting technique at ITU in Karlsruhe, Germany.

The technique involves laser radiation heating a sample to above-melting followed by a controlled cool. As detailed in Ref. 5, this method relies on the simultaneous use of a pyrometer and a spectrometer, providing both the true temperature and the spectral emissivity function of a specimen. The calibration of the pyrometer was performed using certified tungsten strips, with a temperature reproducibility of 2500K ± 7 K. An emissivity value of 0.83 was used to convert temperature brightness into the true value. All measurements were performed in the high-purity argon at 2 bars. The reliability of the procedure was confirmed by the melting point measurement on stoichiometric reactor-grade UO_2, which was shown to be within 0.5% of the recommended value.

Two pellet variants (**ADOPT** fuel and reference UO_2) were laser-pulse melted three times each. No measurable difference in melting temperature was observed between the two pellet variants, as reproduced in Figure 3-9 below. The measured melting temperature, 3122 ± 7 K, agrees well with the measured value of the reference UO_2. For these reasons, it is reasonable to conclude that there is no appreciable difference in the melting point of UO_2 and **ADOPT** fuel (Refs. 1, 2).

b,c

*Figure 3-9 Thermogram of **ADOPT** and reference UO_2 fuel*

3.2.4 Thermal Expansion

Thermal expansion is a material property affecting the physical change in a fuel's density or volume at a given temperature relative to a standard temperature. In particular, thermal expansion is an important consideration in pellet-clad mechanical interaction (PCMI).

The thermal expansion was determined using a BÄHR 802 S differential dilatometer at ITU in Karlsruhe, Germany. All tests adhered to ASTM E 831-86, the 'Standard Test Method for Linear Thermal Expansion of Solid Materials by Thermomechanical Analysis' (Ref. 6). The tests were performed under a continuous flow of argon gas with 5% H_2 at 50 l/h in order to keep the stoichiometry of the oxygen-to-metal ratio (O/M) = 2. Two unirradiated **ADOPT** samples were analyzed and compared with three pure unirradiated UO_2 samples. Data was collected over a temperature range from 20°C - 1490°C with a heating rate of 5°C/min. The results are shown in Figure 3-10 below (Refs. 1, 2).

b,c

*Figure 3-10 Thermal expansion measurements of **ADOPT** and reference UO_2 fuel*

As expected, the difference in thermal expansion between standard UO_2 and **ADOPT** fuel is negligible; the measurements are within the error range associated with the differential dilatometer measurement technique, and it is reasonable to conclude that there is no difference between the thermal expansion of **ADOPT** and UO_2 fuel.

3.3 MECHANICAL PROPERTIES

3.3.1 Modulus of Elasticity

It is expected that minor additions of chromia and alumina will have minimal effect on the elastic property variation as compared to standard UO_2 fuel. The fuel temperatures will have a much more significant impact on the elastic moduli. Furthermore, the rule of mixtures, as used for the calculation of theoretical density in Section 3.1.4, can likewise be used to show that the impact of the added chromia and alumina will result in minimal impact to the elastic moduli.

3.3.2 Creep

A series of fresh fuel creep tests were performed to investigate how **ADOPT** fuel compares to conventional UO_2. The tests were performed by the Commissariat à l'Energie Atomique (CEA), in Cadarache, France. Two types of tests, one creep and one hardening, were performed in a compressive mode utilizing a compressive screw-type Instron device equipped in an adapted furnace. In the creep test, deformation versus time was measured at constant load and temperature. In the hardening test, load versus deformation was measured at constant strain rate and temperature (Ref. 7,8).

Each specimen was prepared at room temperature, and a stress of 5 MPa was applied. This load was maintained during the initial heat up process, until the testing temperature stabilized; once achieved, a constant stress (creep testing) or a constant strain rate (hardening test) was performed by an automatic control system. A reducing 95% Ar – 5% H_2 atmosphere was utilized throughout the experiment in order to maintain a constant O/M ratio of the samples. Tests lasted for a maximum of five hours, or less if the axial strain exceeded 10%, so as to prevent destruction of pellet, while being long enough to determine steady-state creep. Duplicate experiments were performed under identical conditions and demonstrated very good reproducibility.

A complete summary of the test material can be found in Table 3-1 below:

Table 3-1: Summary of ADOPT and standard fuel properties in creep test (Ref. 7)

Fuel Type	Hydrostatic density (%TD)	Linear Intercept[1] (microns)	Dopants
ADOPT	97.2	36.9	Cr 300-650 μg/gU Al 70-150 μg/gU
UO_2	96.7	8.2	None

[1] As determined by ASTM E112 lineal intercept method

In the creep tests**, ADOPT** fuel and reference UO_2 were tested at three different temperatures (1300°C, 1500°C, and 1700°C) and three compressive stresses (30 MPa, 45 MPa, and 60 MPa). The measurements revealed a classical creep curve with a strong temperature dependency and sensitivity to applied stress. Figure 3-11 illustrates that measured strain increases dramatically with rising temperature. As shown in Figure 3-12, at temperatures greater than 1500°C, the **ADOPT** fuel exhibited higher viscoplasticity as

compared to the reference UO_2 pellets. However, at temperatures lower than 1300°C, there appears to be no creep benefit (Ref.7).

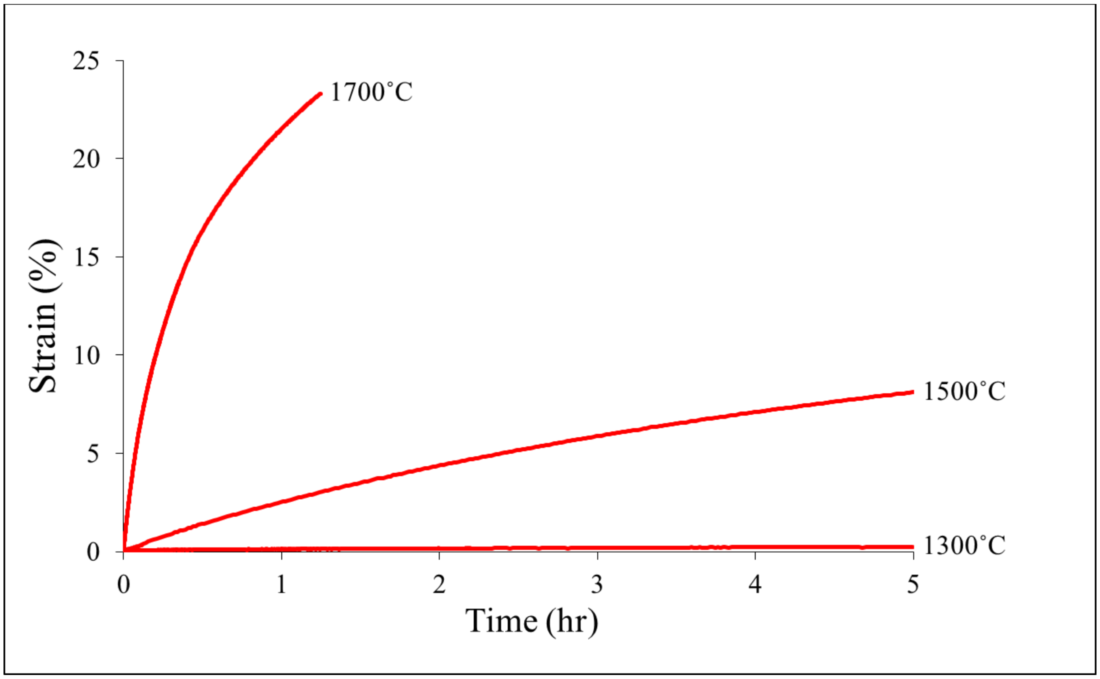


*Figure 3-11 Measured strain for **ADOPT** pellets under constant 60 MPa stress at 1300°C, 1500°C, and 1700°C*

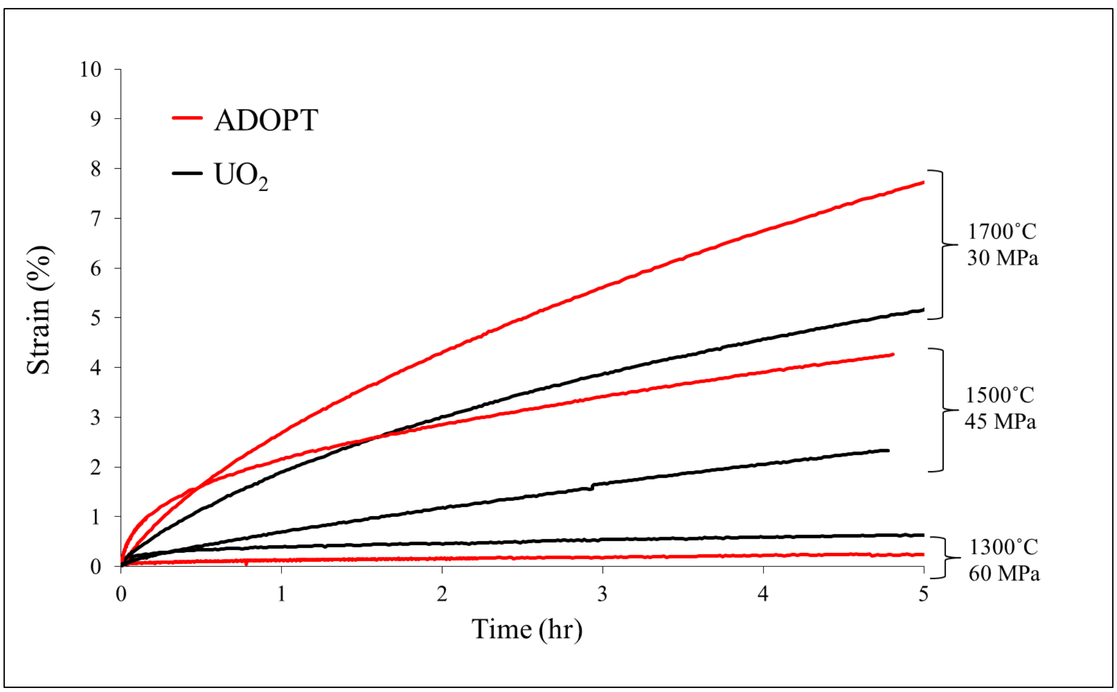


*Figure 3-12 Creep testing of **ADOPT** and reference UO_2 fuel under identical test conditions*

It is clear that at temperatures in excess of 1500°C, the creep rate difference between **ADOPT** and standard UO_2 increases. This can be attributed to the enlarged grain size**.** In this temperature regime, the viscoplastic behavior of **ADOPT** fuel should provide a pellet-clad interaction (PCI) benefit, as the pellet deforms under its own internal stresses and fills in as-manufactured dimples. However, in steady state operations, there is no appreciable difference in the creep behavior of conventional UO_2 and **ADOPT** fuel.

In the hardening tests, **ADOPT** fuel and a reference UO_2 pellet were tested at temperatures ranging from 1100°C to 1700°C. At each temperature, a constant strain rate of 10%/hr or 50%/hr was applied to the specimen.

Again, the hardening tests showed a strong temperature dependency and sensitivity to the applied strain rate. However, in contrast to the creep test, **ADOPT** and standard UO_2 fuel showed similar viscoplasticity in the hardening test, except for 1300ºC where the flow stress value is higher for **ADOPT** fuel. However, at strains that are of interest for PCI, i.e. lower than about 1%, **ADOPT** fuel is more ductile; this is to say that **ADOPT** fuel requires less stress than standard UO_2 to achieve a given strain rate (Figure 3-13). This indicates a more viscoplastic capability for **ADOPT** fuel in the strain levels of interest for PCI (Refs. 7,8).

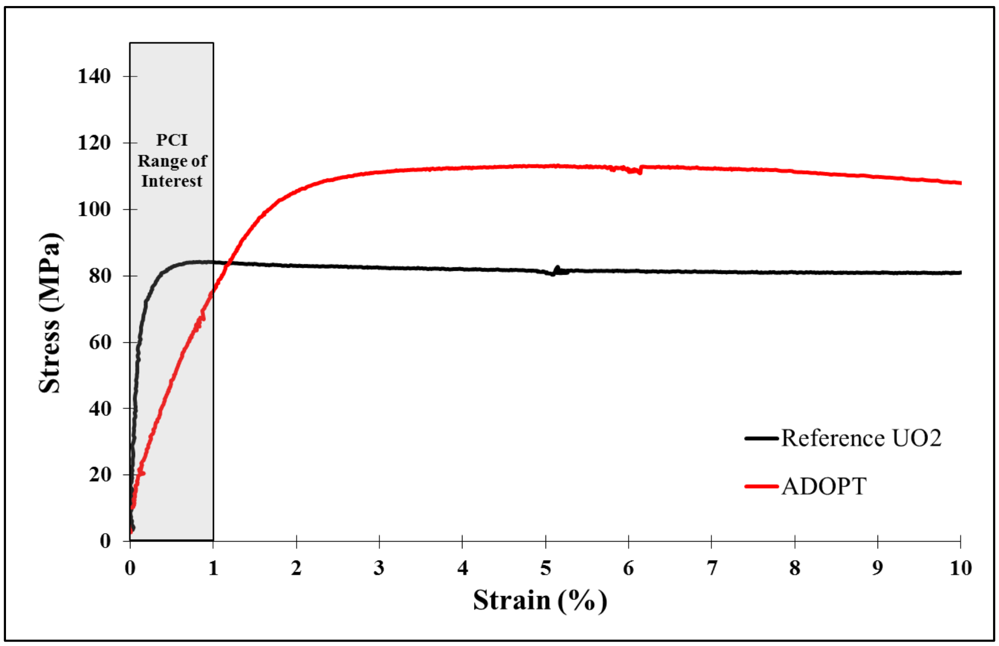


*Figure 3-13 Hardening test of **ADOPT** and reference UO_2 fuel at 1500°C (50%/h)*

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4 IRRADIATION PROGRAMS AND EXPERIENCE

4.1 BILATERAL BÄRSEBACK-STUDSVIK RAMP AND BUMP TESTING

The following sections detail the operating history at Barsebäck 2, as well as the subsequent ramp testing performed at Studsvik.

4.1.1 Barsebäck 2 Base Irradiation

Two segmented liner rods with doped UO_2 and reference UO_2 were base irradiated at the Barsebäck 2 reactor in Sweden up to a segment burnup average of 30 to 33.5 MWd/kgU under normal BWR conditions; Table 4-1 presents the nominal as-manufactured properties of the relevant pellet variants tested, and Figure 4-1 details the full power history (presented as Table 2 and Fig. 4 in Ref. 1.)

Note that the doped D1 pellets don't have alumina as an additive, and the conventional UO_2 D3 pellets have slightly larger than nominal grain size. However, both pellet types are similar in density, composition, and grain size to the **ADOPT** pellet that comparable behavior is expected (Refs. 1, 2).

Table 4-1: Nominal Properties of Pellets in Segmented Rods Irradiated in Bärseback 2

Pellet designation	Test segment	Enrich	Additives [ppm]	Nominal Density [%]	Nominal Density [g/cm³]	Nominal 3D grain size [μm]	Volume change after irradiation at Barsebäck*
HD UO_2	D0	4.2%	-	96.7	10.60	10-12	- 0.2%
Doped UO_2	D1	4.2%	1000 Cr_2O_3	97.3	10.66	44	0.8 to 1.4%
Doped UO_2	D3	4.2%	200 Al_2O_3 + 500 Cr_2O_3	97.4	10.68	52	0.8 to 1.4%

*As determined using both profilometry and pellet-cladding gap measurements

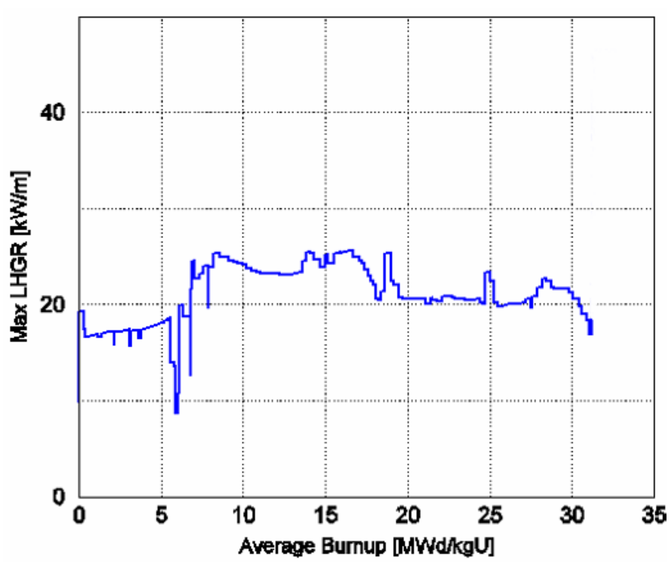


Figure 4-1 Power history of base irradiated segmented rods at Barsebäck 2 Reactor

Following base irradiation, profilometry and pellet-cladding gap measurements were taken, which is used to determine the volume change in the pellet types. The conventional UO_2 pellet experienced a slight volume reduction of approximately -0.2%, whereas the doped UO_2 segments had volume changes ranging from 0.8 to 1.4%. As expected, due to the lack of in-pile densification, the dimensional changes of the doped variants were slightly higher than that of standard UO_2.

4.1.2 Ramp and Bump Testing at the Studsvik R2 Research Reactor

Two segments, standard UO_2 (D0) and doped UO_2 (D1), were refabricated into rodlets and ramp tested in the R2 research reactor at Studsvik. The ramp test initiated by conditioning the rods at 22 kW/m for 12 hours, followed by a 5 kW/m stepped power increase; each step held 1 hour before proceeding, achieving a maximum power of 56.7 and 57.7 kW/m for the conventional and doped UO_2, respectively. Unfortunately, a malfunction occurred, preventing both segments to be held for the same amount of time; the standard UO_2 segment had a hold time of 12 hours, whereas the doped segment only held for 7.7 hours (Refs. 1, 2).

Two other segments, standard UO_2 (D0) and doped UO_2 (D3), were also refabricated into rodlets and bump tested at the R2 reactor. The segments were conditioned at a power of 22 kW/m for 9 hours, followed by a steady power increase to a maximum of 46.4 and 45.1 kW/m for the D0 and the D3 segment, respectively. The rodlets were held at these powers for 17.5 days before the test was terminated (Refs. 1, 2).

The full power history for ramp and bump testing is depicted in Figure 4-2 (shown as Fig. 5 in Ref. 2).

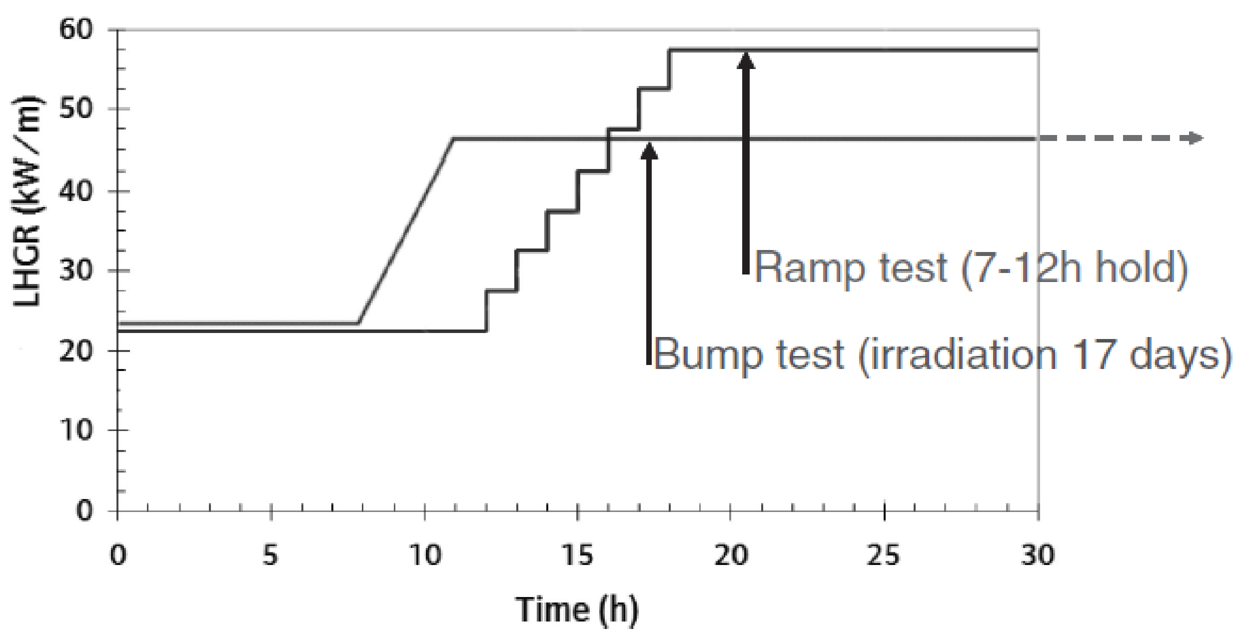
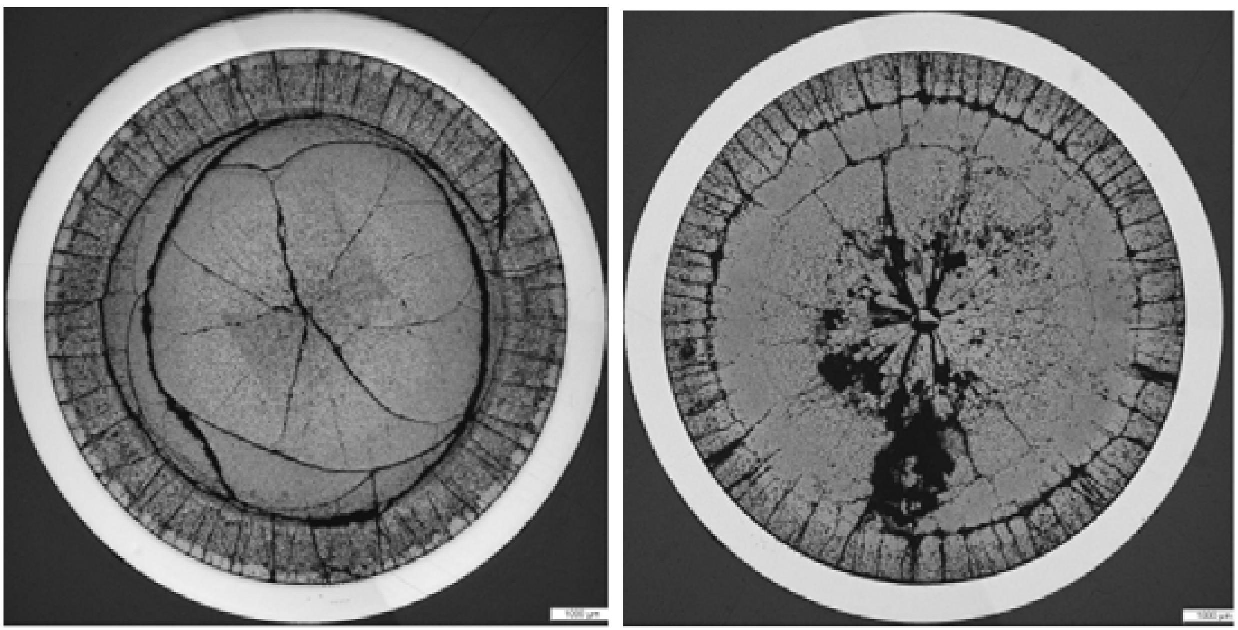


Figure 4-2 Power history of ramp and bump test segments at R2 reactor

4.1.3 Ramp and Bump Test PIE

<u>Ceramography</u>

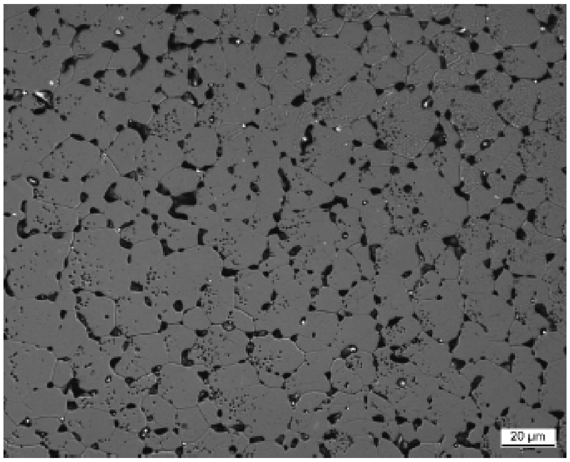
Ceramography of the ramp tested segment (D1) with 1000 ppm Cr_2O_3 additive revealed a central void, unlike the standard pellet, see Figure 4-3. The central void appeared only in the region of the segment which had been subjected to the highest powers. As demonstrated in Section 3.3.2, the additives enhance the pellet's viscoplasticity at these high temperatures, filling the pellet's dished area during power ramp test. Small cracks are observed at the periphery in both pellet types, however less prominent cracking is observed in the central region of the doped pellet (Refs. 1, 2).

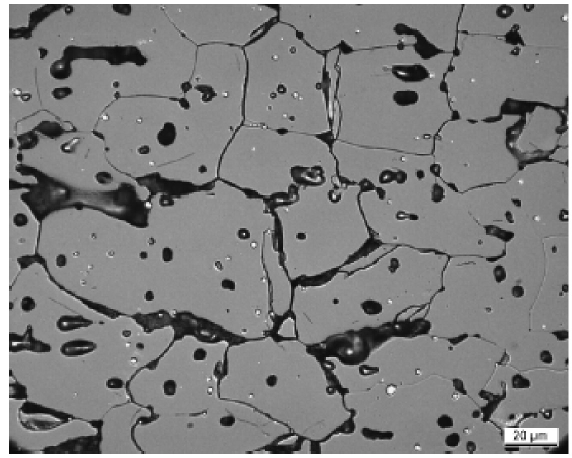


D0 (Reference UO_2) **D1 (Doped UO_2)**

Figure 4-3 Post-ramp ceramographic cross section at mid-pellet region

An increase in porosity following ramp testing was observed in both pellet types. In the standard UO_2 material, fine porosity was observed inside the grains at mid-radius position; the bigger pores were predominately precipitated at the grain boundaries. In the doped (D1) material, pores were primarily observed within the grain. Increased porosity within the grain is indicative of enhanced fission gas retention, see Figure 4-4.





Reference UO_2 (r = 2.2mm from center) **D1 pellets (r = 2.0mm from center)**

Figure 4-4 Pore precipitation at mid radius positions after power ramp test

Fission Gas Measurements

Following ramp testing, the rodlets were punctured to measure the FGR of the two pellet types. A direct comparison was not possible due to the early malfunction of the equipment and resulting variance in hold times. The reference UO_2 segment had a hold time of 12 hours, whereas the doped segment only held for 7.7 hours. However, for reference, the FGR of standard UO_2(D0) was found to be 30.2%, whereas the doped segment (D1) was found to be 17.2%.

Following bump testing, the rodlets were also punctured to examine the amount of FGR of the pellet types. The standard pellet (D0) was measured to have 29.7% FGR, whereas the doped segment (D3) was found to have 20.5% FGR. Thus, the **ADOPT** pellets have about 30% less FGR than the standard UO_2 pellets.

The enlarged grain size of the **ADOPT** pellets gives an improved FGR retention as compared to the standard UO_2 pellets. The FGR behavior is a combination of two competing effects. Firstly, the enlarged grains of the **ADOPT** pellets creates longer diffusion paths for fission products precipitated within the grains. This is beneficial to the FGR retention of the pellets. Secondly, as a result of the additives, the gas diffusion rate is enhanced, which is negative to the FGR behavior. During the relatively short hold times investigated, the first beneficial effect considerably exceeds the second negative, as can be seen in Table 4-2 from the 30% lower FGR for the **ADOPT** pellets as compared to standard UO_2 (Refs. 1, 2).

Table 4-2: Summary of FGR measurements in Studsvik R2 Reactor

Test	Standard UO_2	ADOPT
Ramp test (57 kW/m)	30.2% (12 hour)	17.2 (7.7 hour)
Bump test (45 kW/m)	29.7% (17 day)	20.5% (17 day)

<u>Volume change</u>

The total volume change from base irradiation to the end of ramp testing, calculated from profilometry and gap measurements, are [ ]^{b,c}, respectively, see Figure 4-5. Subtracting the volume change before the ramp test (Section 4.1.1) from the total volume change, the volume change generated during the ramp test is obtained. Taking only the ramp irradiation into account, the volume change determined from the profilometry and gap measurement is larger for the standard UO_2 than the D1 segments, with a difference of about [ ]^{b,c} (Refs. 1, 2) see Figure 4-5 and Figure 4-6.

[ ]^{b,c}

*Figure 4-5 Total volume change of **ADOPT** and UO_2 fuel after Studsvik ramp testing*

[ ]^{b,c}

*Figure 4-6 Volume change of **ADOPT** and UO_2 fuel during Studsvik ramp testing*

4.2 OSKARSHAMN 3 NUCLEAR POWER PLANT

4.2.1 Operating history

Rods with **ADOPT** pellets, placed in four fuel elements, were irradiated in Oskarshamn 3 between 2000 and 2008, achieving a rod average burnup up to 60 MWd/kgU. Afterwards, two rods with **ADOPT** fuel and two standard rods were transported to the Studsvik hot cell laboratory for post irradiation examinations including FGR analysis (Section 5.3), metallographic and ceramographic examinations, and SEM and EPMA. Nominal pellet properties are presented in Table 4-3, and the power histories of the investigated rods in presented in Figure 4-7.

This material has also been subjected to ramp testing within the SCIP II program (Section 4.2.2), FGR testing within [$]^{a,c}$ and RIA testing at JAEA (Section 5.5).

Table 4-3: Nominal Properties of Pellets in Oskarshamn 3 Reactor

b,c

b,c

Figure 4-7 Power histories for the investigated O3 rods

4.2.2 Fuel Pellet Cracking

The fuel pellet crack pattern, which is developed during operation, is an important factor for the PCMI. Cracking in the pellet is induced by radial temperature gradients during operation and begins during power escalation, as the pellet reaches operating temperature.

Hot-cell examination of **ADOPT** and reference UO_2 pellets with very similar operating histories has been performed at a fuel rod average burnup of 60 MWd/kgU. Ceramographic examinations of cross sections at 2000 mm elevation are shown in Figure 4-8. From these examinations, it can be concluded [

][b,c]

b,c

Figure 4-8 Crack pattern of ***ADOPT*** *(left) and reference* UO_2 *(right) pellet at base irradiation*

4.2.3 Ceramography of High Burnup Structure

At very high burnup levels and relatively low temperatures, a high burnup structure (HBS) can begin to form at the rim of the pellet; the grains at the pellet periphery restructure into sub-micron grains, and the fission gases relocate into these newly developed intergranular pores. (Ref. 4)

Hot-cell examination of 24565/C1 (**ADOPT**) and 24565/A2 (standard UO_2) clearly shows an onset of the high burnup rim structure at burnups by 60 MWd/kgU (see Figure 4-9 and Figure 4-10). The variation of the high burnup rim structure around the circumference can be quite large, but as seen in the figures, there is no significant difference between these two pellets types with respect to formation of a high burnup rim structure. At a radius of 3mm, needle shape features are sparsely present but steadily increase toward the periphery of the pellet, where the HBS is beginning to form. As discussed in Ref. 5, these planar defects in the crystal structure are seen prior to formation of the high burnup structure.

From this investigation, **[**

]^{b,c}

4.2.4 Cladding Metallography

Formation of radial hydrides within the cladding are of primary concern, as they reduce ductility and affect cladding integrity. Mainly, the fear is that the hydrides will precipitate radially throughout the cladding, thereby reducing the failure limit of the tube.

One of the primary contributors to hydride reorientation is excessive pressure on the interior wall of the cladding. Given **ADOPT** fuel's increased thermal stability, there is concern that the earlier pellet-cladding contact would cause excessive hydride reorientation. However, as demonstrated in Section 5.2, the swelling rates of UO_2 and **ADOPT** fuel are similar. Beyond earlier contact, no additional pressure is brought to the cladding.

Metallography at 0°, 90°, 180°, and 270° angles of 24565/A2 (standard UO_2) and 24565/C1 (**ADOPT**) is presented in Figure 4-11 and Figure 4-12, respectively. A visual comparison reveals that both rods are very similar. The **ADOPT** segment contained 245 ppm H whereas the standard UO_2 rod contained 289 ppm H; furthermore, and more importantly, both rods appear to have a consistent ratio of radial to tangential hydrides around the rod. **ADOPT** fuel is concluded to have no impact on clad hydride reorientation.

[ ]b,c

Figure 4-9 Ceramography of ***ADOPT*** *fuel pellet at 3mm, 3.8mm, and at the periphery under 500x magnification*

[ ]b,c

Figure 4-10 Ceramography of reference UO_2 fuel pellet at 3mm, 3.8mm, and at the periphery under 500x magnification

[ ]b,c

Figure 4-11 Metallography of hydride structure in Zr-2 cladding with liner at 0°, 90°, 180°, and 270° in standard UO_2 rod

[ ]b,c

*Figure 4-12 Metallography of hydride structure in Zr-2 cladding with liner at 0°, 90°, 180°, and 270° in **ADOPT** rod*

4.2.5 EPMA

Electron probe microanalysis (EPMA) was also performed on the **ADOPT** rods to investigate the dopant migration behavior under steady state conditions. A line scan across the 59.1 MWd/kgU **ADOPT** rod, shows **[**

]^{b,c}

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

*Figure 4-13 EPMA line scan analyses of Al across 59.1 MWd/kgU **ADOPT** rod*

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

*Figure 4-14 EPMA line scan analyses of Cr across 59.1 MWd/kgU **ADOPT** rod*

4.3 STUDSVIK SCIP II RAMP TESTING

FGR measurements

Within the Studsvik SCIP II program, two **ADOPT** and two UO_2 rods manufactured by Westinghouse were ramp tested; the segments were fabricated from rods base irradiated in the Oskarshamn 3 NPP, described in the previous section. [

]^{b,c}

[ ]^{b,c}

Table 4-4: Trend of FGR with hold time for both UO_2 and ADOPT fuel

PIE

PIE was then performed on **ADOPT** pellets after ramp testing. An isotope spectrometry technique was able to more clearly demonstrate [

]^{b,c}

[ ]^{b,c}

*Figure 4-15 Spectrometry techniques on the irradiated **ADOPT** pellets*

4.4 [ ][d]

Four assemblies containing 28 **ADOPT** rods each (in total 112 rods) have been irradiated in **[ ]**[d] since 2006. They reached an assembly burnup of about 50 MWd/kgU in 2010. Visual examination, length measurements, and FGR gamma scanning were performed. Twelve **ADOPT** rods and four rods containing undoped pellets were selected, and loaded in two assemblies with burnup 26 MWd/kgU. They were irradiated up to a final fuel rod average burnup of approximately **[ ]**[b,c] The total number of cycles for these rods is then 7.

These assemblies have been inspected with respect to visual performance and length measurements during 2-life irradiation. After the end of irradiation, the fuel rods were measured for fission gas release (Section 5.3), as well as rod growth (Section 5.2.2).

4.5 HALDEN IFA-677 – THE HIGH INITIAL RATING EXPERIMENT

IFA-677.1 was an experiment designed to investigate the effect on fuels subjected to high initial ratings. A total of six rods, three of which were fabricated by Westinghouse (Rods 1, 5, and 6), were irradiated in the Halden Boiling Water Reactor (HBWR) from 2004 to 2007; two rods contained doped fuel pellets, while the other four rods contained standard UO_2 pellets. Relevant fuel characteristics for all of the fuel rods is presented are presented in Table 4-5 below (Refs. 3, 6, 7).

Table 4-5: IFA-677 Fuel Parameter Characteristics

[ ][b,c]

The Halden rig operated at very high linear heat rates, approximately 35-45 kW/m, achieving an average burnup of 26.3 MWd/kgU. Throughout irradiation, the operating temperatures ranged from 1000-1500°C. The local power and temperature profile for rod 5 (**ADOPT** fuel) is presented in Figure 4-16 below (Ref. 6, 7).

^{b,c}

[

]

Figure 4-16 Local Power and Temperature Profile of Rod 5

4.5.1 Dimensional Changes

As expected, the **ADOPT** rods experienced almost no densification; Rod 1 densified [$]^{b,c}$, which is typical of UO_2 fuel (see Section 5.2.1 for thermal stability testing). This is to be expected, as the additives facilitate greater diffusion during sintering, thereby resulting in a higher as-fabricated density and increased thermal stability.

Throughout irradiation, the two **ADOPT** fuel rods [$]^{b,c}$ This difference is attributed to the negligible densification of the **ADOPT** rods at the beginning of life.

Accounting for initial densification, gives a more accurate representation of the dimensional changes. The non-doped rods swelled [$]^{b,c}$ The data demonstrate that following the densification period, the swelling rates of **ADOPT** and UO_2 are comparable. A complete summary is provided in Table 4-6 (Refs. 6, 7).

Table 4-6: IFA-677 swelling rates

[ ]b,c

4.5.2 Fission Gas Release

Data from IFA-677 confirmed that the dominant effect driving steady state FGR is the fuel centerline temperature. Any effect of the larger grains of doped pellets is a much smaller, second order effect. However, the operational history has been shown to have a significant impact on the results. Higher powers, and therefore temperatures, are known to promote FGR. In particular, a 'burst' of gas is released once the centerline temperature jumps above the Vitanza fission gas release threshold.

The Vitanza threshold is shown as a pink dotted line in Figure 4-17. The threshold is derived from many data points as a line of best fit through them and should not be viewed as a hard threshold line but rather the approximate peak temperature region where significant, defined as in the order of a few percent, FGR is likely to occur.

The size of the burst is affected by the power transient and the incubation period below the threshold where gas is building up inside the fuel. In particular, rod 1 had a significant power increase, (and a centerline temperature increase) at 15 MWd/kgU which promoted a large burst of gas release.

It can be concluded from this experiment that **[**

]b,c

[ ][b,c]

Figure 4-17 Summary results from the IFA-677 experiment containing ***ADOPT*** *pellets.*

4.6 [ ][a,c] – FISSION GAS RELEASE

[ ][a,c] was an experiment developed to compare FGR, PCMI differences, and thermal behaviors at high burnups in doped UO_2. The irradiation occurred in the [ ][a,c] The test rods were refabricated with **ADOPT** fuel and UO_2, which had previously been irradiated in the same father assembly that had achieved approximately 58 MWd/kgU burnup in the Oskarshamn 3 reactor, see Section 4-6. [ ][a,b,c]

Power was incrementally increased with a 24 hour hold time at each level; each stepped power increased corresponded to approximately 35 °C increase. After the maximum power was achieved, the power was maintained until completion of the experiment, which lasted approximately 80 days.

Figure 4-18 depicts the centerline temperatures of rods [ ][a,c] as well as the corresponding fission gas release for each rod. (Ref. 8).

b,c

*Figure 4-18 Comparison of **ADOPT** and Standard UO_2 fuel FGR and centerline temperatures in [ ]^{a,c}*

[

]^{a,b,c}

4.7 CHAPTER 4 REFERENCES

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8. **[**

   ]^{a,c}

5 CHARACTERIZATION OF ADOPT FUEL BEHAVIOR

5.1 CORROSION AND WASHOUT CHARACTERISTICS

5.1.1 Thermo-balance Test

A thermal-microbalance test was performed to quantify **ADOPT** fuel's resistance to corrosion. Samples of conventional UO_2 and **ADOPT** fuel were tested under highly oxidizing conditions while measuring their on-line weight increase.

After sintering, each pellet type was ground to fit the testing equipment. The dimensions and surface roughness were carefully controlled to minimize differences between the samples. Two of each pellet type were tested; a specimen was placed in the furnace, and the temperature was raised to 400°C at a rate of 10°C/min. The temperature was maintained for 20 hours and then cooled to room temperature at a rate of 10°C/min. The oxidizing atmosphere was produced by flowing moisturized argon (100 ml/min) at room temperature. The weight change was measured during the entirety of the experiment. The fuel parameters are depicted in Table 5-1 below.

[**Table 5-1: Thermo-balance Fuel Parameter Characteristics**] b,c

In Figure 5-1, it can be seen that the addition of chromium and aluminum decreased the rate of oxidation by more than 50% after 20 hours, which can be attributed to the enlarged grain size. Oxidation occurs along the grain boundaries. Due to the enlarged grain size of **ADOPT** pellets, the ratio of grain boundaries to grains plus grain boundaries is decreased. The oxidation attack is therefore more difficult (Ref. 2).

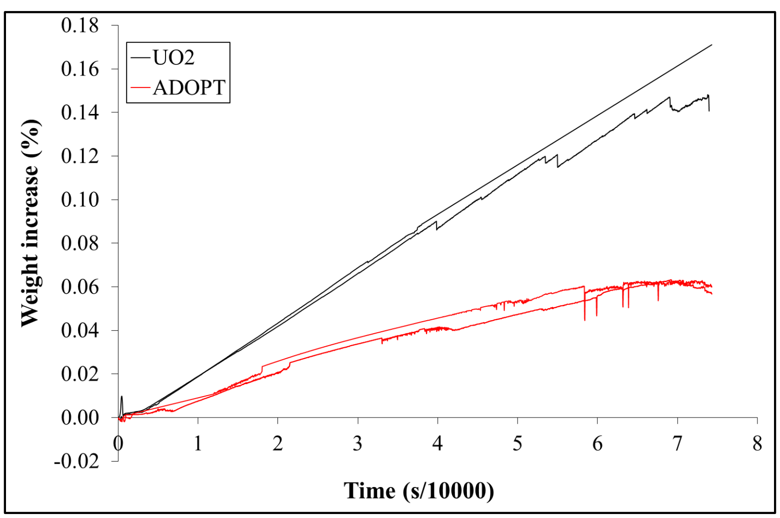

*Figure 5-1 Weight gain in 400°C steam of standard UO_2 and **ADOPT** fuel.*

5.1.2 Studsvik Pellet Erosion Test

In the event of cladding failure, such as that caused by grid-to-rod fretting or PCI, leaking fuel can result in increased coolant activity, contamination within the core, and increased risk to personnel. Furthermore, it can cause premature shutdown of a power plant, resulting in costly repairs.

The Studsvik pellet erosion project was an experiment performed in the R2 reactor with the purpose to measure the resistance against post-failure degradation of several pellet types including Std UO_2, HD UO_2, **ADOPT** pellets, as well as several doped variants.

Test rodlets were irradiated for approximately 70 days over four cycles in the boiling capsule (BOCA) loop of the R2 reactor. During each cycle, the rod powers were held between 25 and 35 kW/m to ensure that sub-cooled boiling conditions were sustained. Each test section contained three pellets and had a 20 mm long, 2 mm wide open slot to simulate a breach within the cladding (Ref. 2).

Following irradiation, the rodlets were subjected to PIE and gamma scanning, which could be used to estimate the fuel loss. The activity of an isotope with high-energy gamma emission corresponds to the volume of fuel, and the activity of an isotope with low-energy gamma emission corresponds to the fuel's surface area. By taking the ratio between the activities of such isotopes, the amount of fuel remaining in the rod could be estimated.

Analysis of the data suggests that fuel loss, i.e. erosion, decreased with density, see Figure 5-2. It can be reasoned that **ADOPT** fuel has enhanced resistance to fuel washout due to its increased density over conventional UO_2 (Ref. 2).

b,c

Figure 5-2 Correlation of density and average fuel loss in Studsvik erosion test

5.2 SWELLING BEHAVIOR

5.2.1 Densification Behavior (Resintering Testing)

During the manufacturing process, the pellets are checked to ensure they are compliant with the material specification. A resintering test is performed for 24 hours at 1700°C to check the thermal stability, a measurement of the pellet's expected densification behavior during irradiation.

A manufacturing analysis was performed on all **ADOPT** pellet lots fabricated at the Westinghouse fuel facility over a two-year period. **[** **]**^{b} Normally and non-normally distributed data was analyzed, and the upper and lower one sided 95/95 tolerance limits were calculated in accordance with the methods specified in US NRC Regulatory Guide 1.126 (Ref. 3). The results for **ADOPT** and reference UO_2 fuel are tabulated in the table below:

Table 5-2: Relative densification of ADOPT and standard UO_2 fuel

a,b,c

There is a clear difference in the densification of standard UO_2 and **ADOPT** fuel, which will certainly have an impact on the allowable effective plastic strain criteria of the cladding. The reduction in densification is due to the material already achieving high density during the sintering process – a clear advantage in other aspects.

5.2.2 Rod Growth Data

As a result of **ADOPT** fuel's reduced in-reactor densification, the pellet-to-clad gap will close earlier than standard UO_2. After gap closure, pellet swelling will initiate and control the rod growth behavior. Several factors influence rod growth; of primary importance are pellet swelling, the as-fabricated gap size, and cladding type.

Westinghouse has a diverse rod growth database on varying **ADOPT** fuel types up to a **[** **]**^{b,c} In the figures presented below, the fuel rod growth is presented by the initial manufactured gap, as well as reactor type (BWR or PWR).

Figure 5-3 and Figure 5-4 depict rod growth data for the Westinghouse BWR design with LK3 Zircaloy-2 cladding, with the primary difference being the as-fabricated gap. The measurements illustrate that rods with **ADOPT** pellets have an increased axial length growth. **[**

]^{b,c}

Figure 5-5 depicts the rod growth data from the [ ][d] As expected, the rods containing **ADOPT** fuel had a higher relative elongation compared to the undoped fuel rods.

[ ][b,c]

Figure 5-3 Fuel rod growth data for BWR fuel designs with diametrical gap of 0.17 mm

[ ][b,c]

Figure 5-4 Fuel rod growth data for BWR fuel designs with diametrical gap of 0.15 mm

[ ][b,c]

Figure 5-5 Fuel rod growth data for PWR fuel design

The fuel rod growth data clearly demonstrates that the earlier pellet-cladding contact of the **ADOPT** rods results in increased growth. Since the cladding material is the same for both pellet types, any differences in the rate of rod growth should give an indication of the relative pellet swelling rates. At an intermediate burnup when the standard pellet is in contact with the cladding, the growth rate for standard and **ADOPT** fuel rods is similar, which indicates that the swelling rate for **ADOPT** and standard pellets is also very similar.

5.3 STEADY STATE FGR DATABASE

Fission gas release measurements have been performed on fuel rods with **ADOPT** pellets and standard UO_2 pellets in two programs at O3 and [ ][d] reactors. Each site was selected because it operates at a different power levels, offering a range of FGR data.

The majority of these measurements were performed using poolside gamma scanning, a technique that has now been qualified and its accuracy confirmed by sending some of the measured rods to hot-cells for rod puncturing. In both programs, the **ADOPT** and reference rods have been irradiated with very similar power histories to make a comparison of the different pellet types possible; they are symmetric rods within the same assembly. As fission gas release is significantly impacted by local conditions and power ramps, it is extra important that rods with different fuel types have experienced identical power histories.

In Figure 5-6, the steady state fission gas release measurements for standard and **ADOPT** fuel with similar operating histories is presented; data is further broken down by irradiation program. [

][b,c]

[ ]b,c

*Figure 5-6 Steady state fission gas release measurements for standard and **ADOPT** fuel with similar operating histories*

5.4 FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL (FFRD)

During a LOCA event, the temperature rise results in significantly increased cladding ductility. This, in combination with a rod internal overpressure, may lead to what is described as 'cladding balloon and burst.' The possible subsequent fuel fragmentation, relocation and dispersal (FFRD) then becomes a safety concern. The existence of a burnup threshold with regards to FFRD has been the topic of extensive research is recent years, with significant research efforts taking place within the Electric Power Research Institute (EPRI) and USNRC sponsored Studsvik tests, the Halden reactor program, within the SCIP program and elsewhere.

Studies indicate that fine fragmentation, which can lead to significant fuel exiting the rod, does not readily occur for fuels below the currently approved burnup limit. Beyond this limit, higher burnup itself appears to be one of many factors affecting FFRD; many hypotheses proposed also take account of last cycle power history and fission gas rearrangement inside the pellet. Nevertheless, high burnup and formation of significant high burnup structure in the pellet rim region beyond the currently approved burnup limit appears to be one of the primary causes of significant FFRD (Ref. 4).

[

]b,c

5.5 REACTIVITY INITIATED ACCIDENTS (RIA)

A reactivity-initiated accident (RIA), which is triggered by a control rod ejection in PWRs or a control rod drop in BWRs, results in an immediate, undesirable surge in fission rate and reactor power. Such accidents cause an immediate rise in fuel temperature and power, with the main safety concern being loss of long term coolable fuel geometry plus damage to the core and vessel from pressure wave generation.

Test reactor data suggests that the failure threshold is strongly dependent on the hydrogen content of the cladding. In particular, high burnup rods are prone to failure due to a combination of tightly bonded fuel-clad gap and high hydrogen content in the cladding towards the end of life. During the initial RIA phase when the pellet is heated adiabatically, the power and local pellet expansion is a function of the fissile atom distribution. With high burnup rods, featuring significant breeding of Pu-239 in the rim region, it is the outermost parts of the pellet which expand and cause cladding stress. In addition, the hydrogen orientation, and not solely its total content in the cladding, determines the failure threshold. For this reason, at the same total hydrogen content, fully re-crystalized annealed cladding (RXA) with its propensity to precipitate radial hydrides has a lower failure threshold compared to stress relieved annealed (SRA) cladding. Tests in cold conditions are also prone to failures due to lack of metallic clad ductility plus dissolved hydrogen being a function of temperature (Ref. 7).

Pellet thermal expansion, fission-gas induced swelling, fuel fragmentation, melting point, heat-up rate and transient fission gas release are all important factors related to the pellet in determining the applied stress on the cladding. Diffusion driven processes do not have time to occur during the short RIA pulse, and so the greater high temperature creep of the **ADOPT** pellet (Section 3.3.2) does not have time to reduce the

clad stress. The characteristics of the RIA power pulse, in particular the pulse width and amplitude, affect the susceptibility of the rod to fail. Short high amplitude tests are more difficult for the cladding to survive than longer duration lower amplitude tests.

[

$]^{a,c}$

5.5.1 RIA Simulation by Expansion Due to Compression Testing

An **ADOPT** rod, 24565/C1, and its sibling reference UO_2 rod, 24565/A2, were base irradiated in the same assembly at Oskarshamn NPP [Section 4.2] and subjected to expansion due to compression (EDC) tests at Studsvik. The EDC method is a high strain test intended to simulate the PCMI behavior during RIAs. A polymer pellet is placed within a cladding specimen and rapidly compressed, resulting in radial expansion of the pellet. The rapid compression occurs in 30-100 milliseconds and is capable of achieving hoop strain rates of $100\%s^{-1}$ to $1000\%s^{-1}$.

Prior to testing, Hot Vacuum Extraction (HVE) of hydrogen from two of the samples was performed. [

$]^{a,c}$ The difference in the hydrogen content between the two specimen is not large; furthermore, the local concentration may vary.

Four cladding specimens were tested, two segments fueled with conventional UO_2 and two fueled with **ADOPT**. All tests employed the same loading conditions, a compression of the polymer pellet of 2.9 mm in 80 ms. A low temperature test was performed at 60°C, and a high temperature test was performed at 300°C. [

$]^{a,c}$ The results suggests that temperature is a dominate factor for cladding failure. A complete summary of the results is presented in Table 5-3 (Ref. 5).

Table 5-3: Expansion due to compression testing on ADOPT and reference UO_2 fuel

b,c

5.5.2 JAEA ALPS-II RIA test

One **ADOPT** rod, with a segment burnup of 64 MWd/kgU, was subjected to a pulse irradiation test at the Nuclear Safety Research reactor (NSRR) by the Japan Atomic Energy Agency (JAEA) in the Advanced LWR Fuel Performance and Safety Research II (ALPS-II) program. The segment was taken from parent rod 24565-C1, which was base irradiated in the Oskarshamn 3 reactor (Sections 4.2 and 5.5.1). The full specifications of the test are detailed below (presented as Table I in Ref. 6)

Table 5-4: Specifications of test fuel rods and test conditions in JAEA RIA test (Ref. 6)

Fuel Type	Cladding	Fuel Burnup [MWd/kgU]	Cladding Oxide Thickness [μm]	Cladding Hydrogen Content [ppm]	Coolant Temperature	Coolant Pressure [MPa]	Max increase in fuel enthalpy [J/g]
BWR 10x10	Zircaloy-2	64	24	245	Room temperature	~0.1	287

The RIA test in Japan using **ADOPT** fuel was performed under extremely challenging conditions where a low failure threshold would naturally be expected. It was performed at room temperature on a high burnup segment with Zr-2 RXA cladding and subjected to a challenging power pulse. Nevertheless, even after accounting for the conditions, the failure was observed at a lower than expected fuel enthalpy, 160 J/g, Figure 5-7.

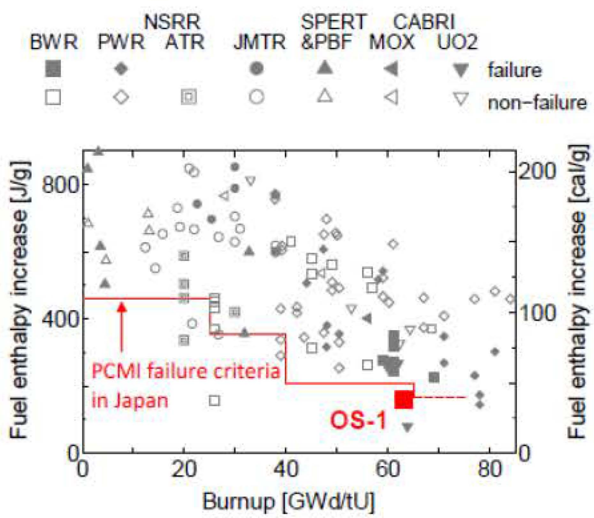

Figure 5-7 Existing RIA-simulated-experiment data and the current Japanese PCMI-failure criteria shown as a function of fuel burnup. (presented as Fig. 7 in Ref. 6)

Post-test examinations of the OS-1 rod revealed an axial crack in the cladding, which is indicative of a PCMI failure. JAEA attributed preliminary indications of the failure to a larger than normal fraction of radial hydrides and those also being in-homogeneously distributed, i.e., a high concentration of radial hydrides was observed at a specific rod orientation. The **ADOPT** rod contained 245 ppm hydrogen in the cladding, which was less than a UO_2 rod tested under similar conditions, and survived, 69 MWd/kgU and 300 ppm hydrogen, Figure 5-8.

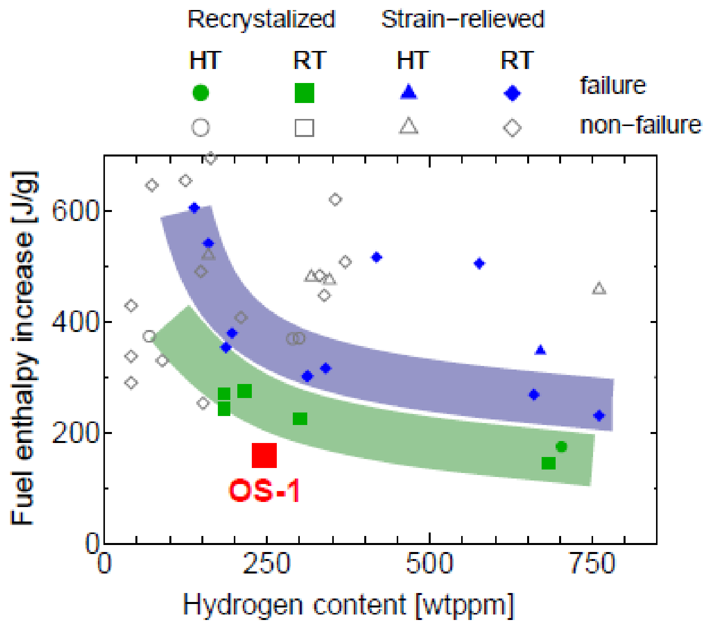

Figure 5-8 Existing RIA-simulated-experiment data plotted as a function of cladding hydrogen content. (presented as Fig. 8 in Ref. 6)

A review of PIE data from other **ADOPT** rods and similar UO_2 rods, did not show any difference in either total hydrogen pickup or from visual examination of the metallography images any difference in the ratio of radial hydrides, See Section 4.2.

In summary, the OS-1 test was very aggressive, and a low failure threshold could be expected. The use of very high burnup fuel (closed pellet-clad gap), low temperature (low clad ductility and precipitated hydrides), and the use of RXA cladding (radial hydrides) all played a significant role in the test.

There is no reason to suspect that the early failure is linked to **ADOPT** fuel. Additionally, the failure threshold using partially re-crystalized annealed (pRXA) **ZIRLO**® or **Optimized ZIRLO** cladding in the PWR environment will be considerably higher than in the BWR Zr-2 RXA cladding due to its much lower overall hydrogen pickup.

5.6 CHAPTER 5 REFERENCES

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6 LICENSING CRITERIA ASSESSMENT

6.1 STEADY STATE AND AOO ANALYSES (THERMO-MECHANICAL EVALUATION)

PAD5 (Ref. 1) is the primary Westinghouse fuel rod design (FRD) tool. It incorporates all relevant fuel performance phenomena, including fuel thermal conductivity degradation with burnup and enhanced fission gas bubble swelling at high burnup, as an integrated set of interrelated performance models. Using appropriate input describing fuel rod design and operating conditions, PAD5 calculates key fuel performance parameters such as cladding stress, strain, oxidation and hydriding, fuel temperature and volume changes, and rod internal pressure.

6.1.1 Fuel Performance Models and Methods

The additives of the **ADOPT** pellets facilitate pellet densification during sintering and enlarge the pellet grain size. Differences in physical properties of **ADOPT** and UO_2 pellets are negligible, as shown in Section 3. The available corrosion and creep data suggest beneficial rod performance properties such as improved resistance against post failure degradation and increased PCI margins for **ADOPT** pellets in comparison to undoped UO_2 pellets. It has also been shown through power ramp tests that there is significantly less gas release from the **ADOPT** fuel during transients. The effective plastic strain at discharge burnups is slightly higher for the **ADOPT** fuel rods in comparison with high density UO_2 fuel rods. Key differences from regular UO_2 considered in PAD modeling are higher density and lower fuel densification. Both can be modelled via modifications to existing PAD5 input variables. In other words, PAD5 can simulate the behavior of **ADOPT** fuel within the existing NRC-approved performance models (Ref. 1). **[**

]^{a,c} Applicability of these models is justified below.

Densification Model:

Section 5.7.1 of the PAD5 topical report (Ref. 1) describes **[**

]^{a,c} The lower densification behavior of **ADOPT** fuel, described in Section 5.2.1, can be explicitly modeled with the PAD5 densification model.

Thermal Conductivity Model:

It is acknowledged that the dopant can have an impact on the phonon term of the thermal conductivity model, and the thermal conductivity is expected to decrease with the increase in dopant content. However, the amount of dopant in Westinghouse **ADOPT** fuel is small, and the measured difference in diffusivity and thermal conductivity is negligible, as shown in Section 3.2.

The negligible impact of dopants in **ADOPT** fuel is also confirmed in PAD5 predictions for **[**

]^{a,c} in Ref. 1. The data is taken from **[** **]**^{a,c}

The NRC has requested temperature results at early burnup for Rods 2 through 6 of IFA-677.1 in Request for Additional Information (RAI) 7a of the 2nd set of PAD5 RAIs (Ref. 1). The PAD5 prediction of the

ADOPT rod is similar to the other UO_2 rods, as shown in Figure 6-1 and Figure 6-2 (copied below from Figures 7-4 and 7-5 in Ref. 4).

a,c

Figure 6-1 Measured and Predicted Centerline Temperatures versus Burnup [ ]^{a,c}

a,c

Figure 6-2 Measured and Predicted Centerline Temperatures versus Burnup [ ]^{a,c}

The temperature measurement [ ][a,c] has shown very consistent burnup degradation between UO_2 and **ADOPT** fuel. The temperature increase as a function of power for **ADOPT** fuel ([ ][a,c] as shown in Figure 46 and Figure 50 of Ref. 2) follows the same trend as UO_2 fuel ([ ][a,c] as shown in Figure 51 of Ref. 2) at later ramps. [ ][a,c]

Relocation Model

As discussed in Section 4.2.2, the crack pattern between **ADOPT** and UO_2 pellets are very similar for steady state operation. The difference is mainly in ramp tests, where **ADOPT** fuel shows fewer cracks in the middle of the pellet and more small cracks in the periphery. The impact on fuel temperature and clad strain calculation for ramps or transients are negligible, as the pellet and cladding are likely in hard contact at high power. Therefore, the PAD5 relocation model is applicable to **ADOPT** fuel.

Fission Gas Release Model

The PAD5 steady state thermal fission gas release model is a saturation model, which does not depend on diffusion coefficient and grain size. Predicted fission gas release from [ ][a,c] (Westinghouse **ADOPT** fuel with [ ][a,c] Cr_2O_3 and [ ][a,c] Al_2O_3) and [ ][a,c] (Westinghouse Reference UO_2 fuel) [ ][a,c] from Ref. 3 were provided in Table 8-1 and Figure 8-1 in the response to RAI-8a (Refs. 1 and 5). Both rods are well predicted, with [ ][a,c] measured FGR of [ ][a,c] vs. predicted [ ][a,c], and [ ][a,c] measured FGR of [ ][a,c] vs predicted [ ][a,c]. The PAD5 steady state fission gas release model is confirmed to be applicable to **ADOPT** fuel.

From ramp test data (Sections 4.1.3 and 4.3), **ADOPT** fuel has less transient FGR for the same test conditions relative to UO_2 fuel. To conservatively model **ADOPT** transient fission gas release, the same transient FGR model for UO_2 fuel is also used for **ADOPT** fuel.

Fractional Release of Volatile Fission Products

Calculation of the fractional release of volatile fission products for UO_2 fuel is based on the ANS5.4 standards (Refs. 6 and 32). The standards correlate the release of volatile fission gas product to the release of stable fission gas. These standards can be conservatively applied to **ADOPT** fuel since **ADOPT** fuel has similar steady state FGR and less transient FGR than UO_2 fuel.

Fission Gas Swelling Model

Fission gas swelling is important to predict the cladding diameter change as a result fuel thermal expansion and gaseous swelling from ramp test data. Volume changes can be calculated from compressed gaps from the ramp test. Figure 4-6 shows that the volume change of **ADOPT** rods from the ramp test in Studsvik is smaller than the UO_2 rod in the same test. The volume change is the combined effects of fission gas swelling and dish filling. [ ][a,c] Applying the PAD5 fission gas swelling

model for **ADOPT** fuel will predict slightly larger pellet deformation and therefore is conservative to the calculated cladding diameter change for transient strain analysis.

<u>Rod Growth Model</u>

ADOPT rods show slightly higher rod growth due to earlier pellet-cladding contact. The slightly under-prediction of rod growth with PAD5 model is conservative for rod internal pressure calculation, as the rod internal volume is under-predicted. The PWR rod growth data in Figure 5-5 are replotted along with PAD5 upper bound rod growth model and shown in Figure 6-3. The PWR **ADOPT** data is still bounded by PAD5 upper bound model at higher fluence. Axial clearance between the fuel rods and the fuel assembly structure is still ensured with PAD5 upper bound rod growth model.

a,c

Figure 6-3 Fuel Rod Growth for PWR Fuel Design

6.1.2 Fuel Rod Design Criteria

The purpose of the fuel rod design criteria is to ensure the fuel rod can perform its intended function throughout lifetime of the fuel. The key criteria that impact the Westinghouse fuel rod performance are provided in Section 7.4 of the PAD5 topical report (Ref. 1). Among them, clad oxidation, clad hydrogen pickup, and clad free standing do not depend on pellet properties/models. The impacts on the affected criteria which are applicable for this topical report are described below.

[

]a,c

[ ]^{a,c} All other fuel performance models do not require a change for the **ADOPT** fuel when using PAD5 for evaluating the fuel rod design criteria described in Section 7.4 of Ref. 1.

The primary effects of **ADOPT** pellets on the fuel rod design criteria can be explained by the impacts on fuel temperature and hot gap size, as shown in Figures 6-4 and 6-5, respectively. These plots are for a high burnup, twice-burned assembly of an uprated 3-loop plant with 15x15 Upgrade fuel. There is negligible difference in fuel temperature, with **ADOPT** fuel having consistently lower fuel temperatures relative to UO_2 fuel. This is due to the fact that the **ADOPT** fuel will close the gap slightly early due to low densification (relative to UO_2 fuel). Earlier pellet-clad contact improves conductive heat transfer of the fuel, thereby lowering the fuel temperatures. The impact on various design criteria are summarized in Table 6-1 at the end of this section.

[ ] **a,c**

Figure 6-4 ***ADOPT*** *and UO_2 Fuel Centerline Temperature*

a,c

Figure 6-5 ***ADOPT*** *and* UO_2 *Fuel Hot Gap Size*

6.1.2.1 Clad Stress

Per Section 7.4.1 of Ref. 1, the fuel rod shall not be damaged due to excessive fuel clad stress. The maximum cladding stress intensities, excluding PCI-induced stress, shall be evaluated based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) guidelines. The only parameter in the ASME-bases PAD5 stress methods described in Ref. 1 affected by the presence of **ADOPT** fuel pellets is the pressure differential across the cladding. As discussed in Section 6.1.2.3, the rod internal pressure of the **ADOPT** fuel can be slightly higher than that of standard UO_2 fuel rods due to reduced void volume associated with lower fuel densification. However, the presence of **ADOPT** fuel pellets and subsequent increase in pressure differential across the cladding can be accommodated for clad stress calculations with available margin.

6.1.2.2 Clad Strain

Per Section 7.4.2 of Ref. 1, the fuel rod will not fail due to excessive fuel clad strain. The design limit for the fuel rod clad strain is that the total tensile strain (elastic plus plastic deformation) due to uniform cylindrical fuel pellet deformation during any single Condition I or II transient shall be less than 1% from the pre-transient value.

Transient clad strain is caused by a rapid thermal expansion and fission gas swelling (when power is high enough) of the fuel pellet during a short-term overpower event. Section 5.2 states that thermal expansion rates between **ADOPT** and UO_2 fuel is comparable. Section 6.1.1 has justified no change in the fission gas swelling model. Therefore, the clad strain during a transient event is expected to also be similar. Figure 6-6 shows that the transient clad strain values for **ADOPT** fuel are generally comparable to that of UO_2 fuel (ZrB_2 IFBA and non-IFBA) under the same conditions except at beginning of life (BOL). This plot is for a high burnup, twice-burned assembly of an uprated 3-loop plant with 15x15 Upgrade fuel. Condition II transient events are modeled at each time step. Early in life before pellet-clad contact is made, the UO_2 pellets have a significantly increased gap size relative to **ADOPT** fuel because of the reduced rate of fuel

densification. However, the differences in transient clad strain for **ADOPT** fuel are negligible, generally in the conservative direction, and any slight increase can be accommodated.

a,c

Figure 6-6 ***ADOPT*** *and* UO_2 *Transient Clad Strain*

6.1.2.3 Rod Internal Pressure

Per Section 7.4.3 of Ref. 1, the fuel system will not be damaged due to excessive fuel rod internal pressure. The internal pressure of the lead fuel rod in the reactor will be limited to a value below which would:

- Cause the diametral gap to increase (cladding liftoff) due to outward cladding creep during normal (Condition I) operation;
- Result in cladding hydride reorientation in the radial direction; and
- Preclude extensive departure from nucleate boiling (DNB) propagation.

<u>No Clad Liftoff</u>

The reduction in void volume within the rod, which is driven by the smaller pellet-clad gap size, subsequently increases the rod internal pressure of **ADOPT** fuel. Although there may be a slight reduction in fission gas release for **ADOPT** fuel because of the slightly lower fuel temperatures relative to UO_2 pellets, the reduction in void volume is more significant.

Figure 6-7 shows a comparison between **ADOPT** fuel and UO_2 fuel (with and without ZrB_2 IFBA). **[ ]**[a,c]

[

]a,c As seen in Figure 6-7, the **ADOPT** fuel rod internal pressure is higher than UO_2 fuel at end of life (EOL).

[ ]a,c

*Figure 6-7 **ADOPT** and UO_2 (IFBA and Non-IFBA) Rod Internal Pressure*

In general, the rod internal pressure of **ADOPT** fuel is expected to be consistently higher than UO_2 fuel under the same conditions. The differences between **ADOPT** and UO_2 rods are not significant enough to invalidate the rod internal pressure – no clad liftoff design criteria and can be accommodated.

<u>No Extensive DNB Propagation</u>

DNB propagation is addressed on a mechanistic basis to meet fuel rod burst and ballooning limits using the NRC-approved code and method as described in Refs. 27 and 33. Any increase in rod internal pressure for **ADOPT** fuel is evaluated using the same code and method as for evaluating the standard UO_2 fuel. No other features of the **ADOPT** fuel pellets will affect the rod burst or ballooning calculations, or the DNB propagation evaluation.

Clad Hydride Reorientation

The formation of radial hydrides can reduce the cladding ductility and increase the potential for brittle failure due to subsequent fuel rod handling. However, rod internal pressure analyses performed for the no liftoff criterion confirm that the threshold pressures for hydride reorientation are not exceeded (Ref. 1). As rod internal pressure – no clad liftoff is confirmed to remain met on a cycle-specific basis, the **ADOPT** fuel has no impact on clad hydride reorientation.

6.1.2.4 Clad Fatigue

Per Section 7.4.4 of Ref. 1, the fuel system will not be damaged due to fatigue. The fatigue life usage factor is restricted in order to prevent reaching the material fatigue limit, considering a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles, whichever is more limiting. Fatigue is driven by the accumulated effects of cyclic strains associated with daily load follow and normal plant shutdowns and returns to full power. The fuel-cladding gap for **ADOPT** pellets is closed earlier than undoped UO_2 fuel due to the reduced densification, which results in additional cyclic loading. However, the amplitude of the cyclic stresses is not expected to be significantly different between UO_2 and **ADOPT** fuel, as discussed in Section 6.1.2.2. Therefore, the increase of cladding fatigue for **ADOPT** fuel can be accommodated.

6.1.2.5 Fuel Rod Axial Growth

Per Section 7.4.7 of Ref. 1, the fuel system will not be damaged due to excessive axial interference between the fuel rods and the fuel assembly structure. Fuel rods are designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the assembly without interference.

Fuel rod growth data in Section 5.2.2 demonstrates that the earlier pellet-cladding contact for rods containing **ADOPT** pellets results in increased axial growth. Due to the reduced in-pile densification, the fuel rod growth initiates earlier in life. However, at an intermediate burnup when the standard pellets are in contact with the cladding, the growth rate for standard and **ADOPT** fuel rods is very similar. There is sufficient clearance between the fuel rod and the top and bottom nozzles to accommodate the increase caused by the presence of **ADOPT** fuel.

6.1.2.6 Clad Flattening

Per Section 7.4.8 of Ref. 1, the fuel rod design shall preclude clad flattening during projected exposure. Westinghouse fabricated fuel is sufficiently stable with respect to fuel densification such that the axial column gaps that can form as a result of fuel densification and axial shrinkage are too small to allow clad flattening to occur. Axial column gaps that could occur are sufficiently small such that no densification power spike factor is required. Westinghouse fabrication processes are well controlled with respect to the parameters that impact fuel densification such that adverse fuel performance issues associated with clad flattening do not occur.

Section 5.2.1 shows that the densification of **ADOPT** fuel is significantly less than that of standard UO_2 pellets, making it less likely to create axial gaps large enough to allow clad flattening to occur. It will also be fabricated using well controlled processes such that no axial gaps large enough to allow clad flattening are expected to form. **ADOPT** fuel is therefore acceptable with respect to clad flattening.

6.1.2.7 Fuel Pellet Overheating (Power-to-Melt)

Per Section 7.4.10 of Ref. 1, the fuel rods will not fail due to the fuel centerline melting for Condition I and II events. The fuel rod centerline temperature shall not exceed the fuel melt temperature when accounting for degradation of the melt temperature due to burnup and the addition of integral burnable absorbers.

Section 3.2.3 concludes that there is no difference in the melting point of standard UO_2 and **ADOPT** fuel. Figure 6-4 shows that the fuel centerline temperature for **ADOPT** fuel is slightly lower relative to standard UO_2 fuel. As the design limit is the same, but the calculated centerline temperatures are lower, the power-to-melt limit for UO_2 fuel is conservative for **ADOPT** fuel.

6.1.2.8 Pellet-Clad Interaction

Per Section 7.4.11 of Ref. 1, the fuel rod will not fail due to pellet-clad interaction. There is no specific design criterion for PCI so long as the clad strain and fuel overheating limits are met. **ADOPT** fuel is capable of meeting these design limits, as described in Sections 6.1.2.2 and 6.1.2.7, respectively, so no additional PCI calculations are required.

6.1.2.9 Fuel Rod Design Criteria Conclusions

All fuel rod design criteria identified in Ref. 1 can be accommodated for **ADOPT** fuel pellets. Table 6-1 contains a description of how the design margins are affected by the presence of **ADOPT** fuel (compared to those of standard UO_2 pellets). Margin in this case is defined as the difference between the design limit and the upper or lower bound calculated value with consideration for all relevant uncertainties.

Table 6-1: ADOPT Fuel Rod Design Criteria Assessment

a,c

6.2 SAFETY ANALYSES

6.2.1 LOCA

Design basis Loss-of-Coolant Accident (LOCA) analyses seek to demonstrate that the emergency core cooling system (ECCS) meets the requirements of 10 CFR 50.46. Introduction of **ADOPT** pellets to the fuel design does not affect the overall goal of the LOCA analysis, but does introduce potentially different physical effects which can change the results. This section describes the manner in which **ADOPT** pellets will be addressed for the **FULL SPECTRUM**™ Loss-of-Coolant Accident (**FSLOCA**™) Evaluation Model (EM), described in WCAP-16996-P-A, Revision 1 (Ref. 7), and the NOTRUMP EM, described in WCAP-10054-P-A (Ref. 8) and WCAP-10079-P-A (Ref. 9).

As part of the development process for the **FSLOCA** EM, a phenomena identification and ranking table (PIRT) was constructed to assess the relative importance of various phenomena to both small-break LOCA (SBLOCA) and large-break LOCA (LBLOCA) results. Section 2.3.2.1 of Ref. 7 discusses fuel-related phenomena; those are the phenomena which could be affected by the introduction of **ADOPT** pellets. The potentially affected phenomena are briefly discussed below, with more specific aspects of the **FSLOCA** EM and the NOTRUMP EM described in the following subsections.

Stored Energy

For small breaks, the core remains covered during the early periods of the transient, and reactor trip occurs early. During this period the heat transfer is good, and there is only a small temperature difference between the fuel centerline temperature and the coolant. This removes much of the initial stored energy of the fuel.
[

]^{a,c}

Decay Heat

Decay heat is the main driver of the heatup transient for small breaks and **[**

]^{a,c}

[

]^{a,c}

Clad Deformation (Burst Strain, Relocation)

[

]^{a,c}

Following burst, fuel pellet fragments can relocate into the ballooned section of the clad at the burst location, thereby increasing the local heat generation rate. As discussed in Section 6.2.1.1.2, [

]^{a,c}

6.2.1.1 FULL SPECTRUM LOCA Evaluation Model

This section describes the aspects of the **FSLOCA** EM, described in Ref. 7, which could be affected by **ADOPT** fuel pellets. The models and correlations developed for use in the **FSLOCA** EM [

]^{a,c} as discussed in the following subsections.

6.2.1.1.1 Thermal Properties

The thermal properties of uranium dioxide (UO_2) modeled in the **FSLOCA** EM is described in Section 11.4.1 of Ref. 7.

Density

In the **FSLOCA** EM, the (cold) density for uranium-dioxide is assumed to be 684.86 lbm/ft^3 (10.97 g/cm^3), multiplied by the fraction of theoretical density which is a user input. Section 3.1.4 of this report notes that the chromia and alumina additives adjust the theoretical density of **ADOPT** fuel downward from approximately 10.96 g/cm^3 to [ ]^{a,c} of the theoretical density of UO_2 The theoretical density of 684.86 lbm/ft^3 as assumed in the **FSLOCA** EM therefore remains applicable for **ADOPT** pellets. The increased fraction of theoretical density is modeled through adjustment to the user input. To model the [ ]^{a,c} density for **ADOPT** pellets, the user input for fraction of theoretical density is [ ]^{a,c}.

Thermal Conductivity

WCOBRA/TRAC-TF2 as used in the **FSLOCA** EM uses the modified Nuclear Fuels Industries (NFI) model to account for the effects of fuel burnup on pellet thermal conductivity. As discussed in Section 11.4.1 of Ref. 7, the modified NFI model represents the thermal conductivity for as-fabricated density of

95% of theoretical density, and an adjustment is made to account for as-fabricated fractions other than 95%. The range of applicability of the modified NFI correlation is provided by volume 4 of NUREG/CR-6534 (Ref. 10) includes 92%-97% fraction of theoretical density. As noted in Section 1.1, **ADOPT** pellets have density consistent with **[ ]**^{a,c} of theoretical.

Requests for Additional Information (RAI) 36-39 associated with the **FSLOCA** EM covered pellet thermal conductivity and related burnup effects (see pg. A-261 of Ref.7). Specifically, part 4 of RAI 36 requested information of the Performance Analysis and Design (PAD) fuel rod design code. As discussed in that response, **[**

]^{a,c} Figure RAI 36-1 shows a comparison of the modified NFI conductivity model used in WCOBRA/TRAC-TF2 and the model in PAD5. **[**

]^{a,c}

Specific Heat

As discussed in Section 3.2.1, no change is necessary to the models used for UO_2 specific heat when modeling **ADOPT** fuel pellets.

6.2.1.1.2 Material Behavior

Thermal Expansion

Section 8.4.1 of Ref. 7 discusses the modeling of fuel pellet thermal expansion. As discussed in Section 3.2.4, there is negligible difference in thermal expansion between UO_2 and **ADOPT** pellets. As such, the model described in Section 8.4.1 of Ref. 7 remains applicable for **ADOPT** pellets.

Thermal Conductivity of Relocated Fuel

Section 8.6.2 of Ref. 7 describes the model used to represent relocated fuel (fuel fragments axially relocated within the location of a rupture). **[**

]^{a,c}

[

]^{a,c}

6.2.1.2 NOTRUMP Evaluation Model

This section addresses the aspects of the NOTRUMP EM, described in Refs. 8 and 9, which could be affected by **ADOPT** fuel pellets. Per Section 3-1-2 of Ref. 8, the fuel model in the NOTRUMP code core model is based on the LOCTA-IV code (Ref. 11) as described in Ref. 9. Section 3-1-14 of Ref. 8 indicates the LOCTA-IV code, modified per Section 3-13-1 of Ref. 8, is approved for use in the NOTRUMP EM to calculate the peak cladding temperature in the core during a small break LOCA transient. The models and correlations used in the NOTRUMP EM **[** **]**^{a,c} as discussed in the following subsections.

6.2.1.2.1 Material Properties

The material properties of uranium dioxide (UO_2) modeled in the NOTRUMP EM are described in Appendix T of Ref. 9, Section T-4.

<u>Density</u>

The room temperature theoretical density of UO_2 is assumed to be 684 lbm/ft^3 (10.96 g/cm^3), and is adjusted to account for the user input percent of theoretical density. Section 3.1.4 of this report notes that the chromia and alumina additives adjust the theoretical density of **ADOPT** fuel downward from approximately 10.96 g/cm^3 to **[** **]**^{a,c} of the theoretical density of UO_2. Therefore, the theoretical density of 684 lbm/ft^3 assumed in the NOTRUMP EM remains applicable for **ADOPT** pellets. The increased percent of theoretical density is modeled through adjustment to the user input. To model the **[** **]**^{a,c} density for **ADOPT** pellets, the user input for percent of theoretical density is **[** **]**^{a,c}.

<u>Thermal Conductivity</u>

Section 3.2.2 of this report indicates that the standard Westinghouse methodology for UO_2 can be used to calculate the thermal conductivity for **ADOPT** fuel. The modified Nuclear Fuels Industries (NFI) model is used in the NOTRUMP EM version of the LOCTA-IV code to account for the effects of fuel burnup on pellet thermal conductivity predicted by the PAD fuel performance code. See the Thermal Conductivity discussion in Section 6.2.1.1.1 for applicability of the modified NFI model for **ADOPT** fuel.

<u>Specific Heat</u>

As discussed in Section 3.2.1 of this report, no change is necessary to the models used for UO_2 specific heat when modeling **ADOPT** fuel pellets.

6.2.1.2.2 Material Behavior

<u>Thermal Expansion</u>

Appendix T of Ref. 9, Section T-2, describes the thermal expansion of the fuel pellet in the NOTRUMP EM. Section 3.2.4 herein indicates there is negligible difference in thermal expansion between UO_2 and **ADOPT** pellets. Therefore, the model described in Section T-2 of Ref. 9 remains applicable for **ADOPT** pellets.

6.2.2 Non-LOCA Transient Analyses

This section discusses the effect of the **ADOPT** fuel pellet design on the non-LOCA transient analyses.

6.2.2.1 Effect of ADOPT Fuel Pellets on Non-LOCA Analysis Models

An investigation has been made to determine the material properties of the **ADOPT** fuel pellets, with the objective to provide data for a comparison of the standard UO_2-based fuel and **ADOPT** fuel pellet properties. Data and subsequent evaluations have concluded that the primary differences between standard UO_2-based fuel and **ADOPT** fuel pellets are a higher density and an enlarged grain size as compared to undoped UO_2. Therefore, with the exception of the increased density, and its associated effect on fuel thermal conductivity, the existing standard UO_2 parameters are negligibly impacted with respect to the non-LOCA analyses. The impact of the exceptions will be addressed through input to existing methods, including potential updates to built-in code input and functions, and the approved non-LOCA codes and methods will remain applicable.

6.2.2.2 Acceptance Criteria

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 **ADOPT** fuel pellets.

With respect to the impact of a change in fuel pellet, there are two categories of non-LOCA events that need to be considered:

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

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

]^{a,c}

[**]**^{a,c} The aforementioned existing gross non-LOCA acceptance criteria remain applicable to **ADOPT** fuel pellets.

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 fuel enthalpy (cal/g), minimum DNBR, fuel melting, and peak cladding temperature (PCT). **[**

]^{a,c} Section 6.4 states **ADOPT** fuel pellets do not affect the fuel cladding DNB performance as determined from DNB experiments, or the method for DNBR calculations using a DNB correlation. Additionally, Section 3.2.3 concluded that there is no appreciable difference in the melting point of standard UO_2 and **ADOPT** fuel pellets. Furthermore, the fuel centerline temperature for **ADOPT** fuel pellets is slightly lower relative to UO_2 fuel. As the design limit is the same, but the calculated centerline temperatures are lower, the power-to-melt limit of UO_2 fuel is conservative for **ADOPT** fuel pellets. Finally, since the **ADOPT** fuel pellets do not impact the properties of the fuel rod cladding, there is no impact on the PCT limits. As a result, the aforementioned existing non-LOCA acceptance criteria to address local effects in the fuel rods remain applicable to **ADOPT** fuel pellets.

Therefore, the existing non-LOCA acceptance criteria remain applicable to **ADOPT** fuel pellets.

6.2.2.3 Non-LOCA Conclusions

The computer codes and methods used in the analysis of the non-LOCA licensing basis events remain applicable for the **ADOPT** fuel pellet design. The non-LOCA accident acceptance criteria continue to be applicable for the **ADOPT** fuel pellet design.

6.2.3 Containment Integrity Analyses

This section discusses the effect of the **ADOPT** fuel pellet design on the containment integrity analyses. Any impact would be the result of change in the mass and energy released to containment due to a pipe rupture accident because the containment integrity analyses themselves do not model the fuel. Containment integrity analyses consider the mass and energy released to containment from a loss-of-coolant accident (LOCA) or a steamline break (SLB) event.

6.2.3.1 Short Term LOCA Mass and Energy (M&E) Releases

The short term LOCA mass and energy (M&E) releases are used to determine the maximum differential pressure for structural analyses within sub-compartments inside the containment building resulting from postulated pipe ruptures in the primary system piping. These transients are typically performed for 1 to 3 seconds 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, the corresponding temperature of the fluid in the postulated ruptured pipe, the size of the break, and the initial reactor coolant system pressure. The fuel product and specific aspects of the fuel performance do not influence the short term LOCA M&E. Therefore, any change to the fuel pellet materials would not impact the short term LOCA M&E releases used for short term sub-compartment analyses.

6.2.3.2 Long Term LOCA Mass and Energy (M&E) 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 Combustion Engineering (CE) designs. Those licensed methodologies are:

- WCAP-10325-P-A (Reference 12)
- WCAP-17721-P-A (Reference 13)
- CENPD-132P (Reference 14)

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 to maximize the 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. The specific fuel product is modeled with respect to rod inside and outside diameter, flow area through the core, proposed peaking factors, rod initial gas fractions, rod initial internal pressure, theoretical density of the pellet, the material properties of the pellet, the material properties of the cladding material, and the burnup where the highest fuel temperature during the proposed cycle would occur. These fuel performance characteristics are controlled by a parameter known as the core stored energy that is provided by the Fuel Rod designers (whether Westinghouse, the utility, or a competitor fuel vendor). An iterative process is used to adjust fuel temperature to arrive at the value provided for the core stored energy.

Information provided in Section 3 states that the material properties of the **ADOPT** pellets can be represented by the thermal material properties of UO_2 pellets, including the theoretical density, the specific heat capacity, the thermal diffusivity, and the thermal expansion. Please note that the mechanical material properties of the fuel pellet are not pertinent to the long term LOCA M&E releases. **[** **]**^{a,c} Based on the information in Section 3 that the thermal material properties for the **ADOPT** pellets can be represented by the standard Westinghouse methodology established for UO_2 pellets, no changes are needed for the WCAP-10325-P-A methodology that models an average core or the current plant specific analysis of record values for the core stored energy for a full core with **ADOPT** fuel pellets.

WCAP-17721-P-A Methodology

The methodology approved in WCAP-17721-P-A uses the WCOBRA/TRAC code. This code and methodology model the core as an average core. 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, and material properties that can vary over a temperature range.

Information provided in Section 3 states that the material properties of the **ADOPT** pellets can be represented by the thermal material properties of UO_2 pellets, including the theoretical density, the specific heat capacity, the thermal diffusivity, and the thermal expansion. Please note that the mechanical material properties of the fuel pellet are not pertinent to the long term LOCA M&E releases. **[** **]**^{a,c}

[

]^{a,c} Based on the information in Section 3 that the thermal material properties for the **ADOPT** pellets can be represented by the standard Westinghouse methodology established for UO_2 pellets, no changes are needed for the WCAP-17721-P-A methodology that models an average core for a full core with **ADOPT** fuel pellets.

CENPD-132 Methodology

The CE methodology is documented in CENPD-132. The CEFLASH-4A computer code is used for the blowdown portion of the transient for both the ECCS and LOCA M&E calculations. This code and methodology are based on a hot rod model. Nominal, cold conditions are the foundation for the fuel dimensions. The fuel temperatures that are used are based on a bounding fuel centerline temperature versus linear heat rate over the entire fuel cycle. 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 fuel material properties are also an input into the code. Due to the conservatism in the methodology **[**

]^{a,c} no methodology changes will be needed for a full core with **ADOPT** fuel pellets.

6.2.3.3 Short-Term Steamline Break M&E 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 do not influence the short-term SLB M&E releases. Therefore, any change to the fuel pellet materials do not impact the short-term SLB M&E releases used for short-term subcompartment analyses.

6.2.3.4 Long-Term Steamline Break M&E Releases

The long-term SLB M&E releases analyses use methods and models similar to those discussed for the non-LOCA analyses in Section 6.2.2 and remain valid for the **ADOPT** fuel pellet design. The **ADOPT** fuel pellets are equivalent to standard UO_2 pellets with the addition of chromium oxide and aluminum oxide. The pellet design is characterized by its increased density and enlarged grain size.

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 (References 15 through 17)
- RETRAN (Reference 18)
- SGNIII (Reference 19)

LOFTRAN and RETRAN Methodologies

The long-term SLB M&E releases safety analyses licensed codes and methods are not tied directly to any specific fuel design. Therefore, the safety analyses of the long-term SLB M&E releases are not specifically dependent on the materials that comprise the fuel pellet design. The SLB safety analyses assume bounding reactivity feedback modeling within the licensed computer models to conservatively bound plant operation at the end of core life. Related to the effect of the **ADOPT** fuel pellet design on the long-term SLB M&E releases safety analyses,

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

SGNIII Methodology

The heat effects in the reactor coolant system such as core stored energy, core to coolant heat transfer and decay heat tend to maintain the temperature in the reactor coolant system following a steam line break. A wide variation in these parameters, however, has little effect on the rate of energy release from the steam generators. Due to the overall conservatism in the SGNIII methodology, there will not need to be any changes to the methodology when modeling a full core with **ADOPT** fuel pellets.

6.2.3.5 Conclusions

The computer codes and methods currently used in the analyses of the LOCA and SLB M&E releases used in containment integrity analyses, and therefore the containment integrity analyses themselves, are valid for the **ADOPT** fuel pellet design.

6.2.4 Radiological Consequences Analyses

This section discusses the effect of the **ADOPT** fuel pellet design on the radiological consequences analyses for design basis accidents.

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

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

6.2.4.1 Calculation of Input from Transients

The radiological consequences analyses of the MSLB, Locked Rotor and Rod Ejection rely on transient analyses whose potential effects from the **ADOPT** fuel pellet design have already been discussed in Section 6.2.2 and for which the associated code and methods remain valid for the **ADOPT** fuel pellet

design. The SGTR transient analyses use methods and models similar to those discussed for the non-LOCA analyses in Section 6.2.2 and these too remain valid for the **ADOPT** fuel pellet design. For these accidents, although introduction of a new fuel pellet type could potentially change the calculated results, no codes or methods would have to be changed in order to perform plant specific analysis updates to reflect the change.

The LOCA radiological consequences analyses follow regulatory guidance such as that provided in Regulatory Guide (RG) 1.195 (Reference 20) and RG 1.183 (Reference 21) which specify fuel damage assumptions intended to satisfy a footnote to 10 CFR 100.11 which states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. As such the LOCA radiological consequences analyses are not affected by the **ADOPT** fuel pellet design and this is not dependent on the LOCA analysis discussed in Section 6.2.1. **[**

]^{a,c}

6.2.4.2 Gap Fractions

The radiological consequences analyses for design basis accidents that include consideration of fuel cladding damage follow regulatory guidance such as that provided in RG 1.195 (Reference 20) and RG 1.183 (Reference 21) which specify the fractions of the core inventory assumed to be in the gap for the various radionuclides. **[**

]^{a,c}

6.2.4.3 Fuel Nuclide Inventory

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

6.2.4.4 Conclusions

The computer codes and methods currently used in the analyses of radiological consequence of design basis accidents are valid for the **ADOPT** fuel pellet design.

6.3 IMPACT ON NUCLEAR DESIGN REQUIREMENTS

The **ADOPT** fuel characteristics of density, doping materials, and fuel temperature are inputs into the design and within the capability of current neutronic code package based on previously NRC-approved topical reports assessing neutronics and nuclear design. The **ADOPT** fuel characteristics will be explicitly modeled through design input utilizing standard reload design processes. The low concentration of Cr_2O_3 and Al_2O_3 doping materials have minimal impact on the core reactivity due to the small absorption cross section. The fuel temperature of **ADOPT** and non-doped UO_2 fuel are comparable and therefore fuel temperature changes will have minimal neutronic behavior. The pertinent **ADOPT** fuel characteristic which benefits nuclear design is the higher nominal density of **[** **]**^{a,c} in comparison to the current nominal density of **[** **]**^{a,c}

[

]a,c Therefore, the implementation of **ADOPT** fuel will have no impact on the current design requirements or safety limits. Furthermore, the **ADOPT** fuel characteristics will be explicitly modeled using the standard design processes with the neutronic behaviors of **ADOPT** fuel implicitly captured and confirmed to meet all safety limit requirements as part of the Westinghouse standard reload process.

6.4 APPLICABILITY OF THERMAL-HYDRAULIC DESIGN METHODS

The thermal-hydraulic methods applied to PWR Departure from Nucleate Boiling (DNB) consist of a DNB correlation such as WRB-1 (Reference 22), WRB-2 (Reference 23), WRB-2M (Reference 24), WSSV (Reference 25) and WNG-1 (Reference 26), a thermal-hydraulic (T/H) subchannel code such as Westinghouse version of the VIPRE-01 code, referred to as the VIPRE-W code (Reference 27), and a statistical method for determination of a 95/95 DNB Ratio (DNBR) limit, such as the Revised Thermal Design Procedure (RTDP) (Reference 28) and the Westinghouse Thermal Design Procedure (Reference 29). Thermal-hydraulic analysis can also be performed as part of the integrated non-LOCA analysis methodology described in References 30 and 31.

Implementation of the **ADOPT** fuel does not require modification or update to any previously NRC-approved methods and topical reports for DNB and thermal-hydraulic analyses, such as References 22 through 31. The **ADOPT** fuel does not affect the fuel cladding DNB performance as determined from DNB experiments, or the method for DNBR calculations using a DNB correlation. The VIPRE-W code can perform steady-state and transient DNBR calculations and non-LOCA post-Critical Heat Flux (CHF) fuel rod transient analysis. There is no change in the VIPRE-W transient modeling method as described in Reference 27 for its application to the **ADOPT** fuel based on the similarities between standard UO_2 fuel and **ADOPT** fuel. The existing Westinghouse thermal-hydraulic design methods remain applicable to **ADOPT** fuel as an acceptable fuel material.

6.5 LICENSING CRITERIA CONCLUSION

As discussed in the previous sections, due to similarities in the performance between them, **ADOPT** fuel can be used in place of standard UO_2 fuel. The existing NRC-approved analytical methods and models are appropriate to analyze the performance of **ADOPT** fuel with either minimal or no modifications and the acceptance criteria previously used to demonstrate safety for standard UO_2 fuel are appropriate to confirm safe operation with **ADOPT** fuel. From this, **ADOPT** fuel can be fully characterized and analyzed to show that it may be safely implemented for use in commercial operating nuclear reactors within the constraints identified in Section 1.3.

6.6 CHAPTER 6 REFERENCES

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7 SUMMARY

ADOPT fuel is a standard, commercial product for the European market with extensive operating experience and superior performance; it is characterized by a higher pellet density and larger grain size. Westinghouse has over 20 years of irradiation experience up to **[ ]**[b,c] and has been delivering reloads for 15 years. The current efforts are focused on bringing **ADOPT** fuel technology to the U.S. PWR market.

The strategy for the licensing of **ADOPT** fuel in the US includes two topical reports: (1) A near-term topical report submittal seeking approval for the use of **ADOPT** fuel while crediting minimal material performance enhancements, and (2) a longer-term topical report submittal that will seek to fully credit all the performance enhancements enabled by **ADOPT** fuel.

A brief description of the planned performance enhancements for each topical report is provided below:

Near-Term Topical Report:

[ ][d]

Long-Term Topical Report (for informational purposes only):

[ ][d]

As demonstrated within the topical report, **ADOPT** fuel can be accommodated within existing safety analysis methods through input; implementation of **ADOPT** fuel does not require modification or update to any previously NRC-approved methods. Inclusion of alumina and chromia does not introduce any new failure modes or phenomena that require new or revised SAFDLs.

Upon approval of this topical report, **ADOPT** fuel may be used with all NRC-approved Westinghouse and Combustion Engineering pressurized water reactor mechanical fuel designs including all NRC-approved zirconium-based cladding materials and fuel enrichments.