WCAP-18482-NP-A
Revision 0

September 2022

Westinghouse Advanced Doped Pellet Technology (ADOPT^{TM}) Fuel

Westinghouse

WCAP-18482-NP-A
Revision 0

Advanced Doped Pellet Technology (ADOPT™) Fuel

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September 2022

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Acknowledgement

The authors wish to acknowledge the following individuals for their contributions in development of the topical report: Dave Mitchell, John Reck, Bert Yates, and Andrew Bowman.

Thank you all for your hard work and determination.

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Section A

Final Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 13, 2022

Ms. Camille Zozula, Manager
Infrastructure & Facilities Licensing
Westinghouse Electric Company
1000 Westinghouse Drive
Building 1, Suite 165
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR WCAP-18482-P/WCAP-18482-NP, REVISION 0, "WESTINGHOUSE ADVANCED DOPED PELLET TECHNOLOGY (ADOPT™) FUEL" (EPID: L-2020-TOP-0025)

Dear Ms. Zozula,

By letter dated May 8, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20132A014), Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-18482-P/WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel." By email dated March 8, 2022, the NRC staff issued its draft safety evaluation (SE) (ADAMS Accession No. ML22005A063).

Westinghouse provided comments on the draft SE by letter dated April 5, 2022 (ADAMS Accession No. ML22098A059). Non-proprietary copies of the comments have been placed in the NRC Public Document Room and are available in ADAMS under Accession No. ML22098A061.

The NRC staff has found the TR acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the NRC final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the accepted material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

NOTICE: Enclosure 2 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish accepted versions of the proprietary and non-proprietary TR within three months of receipt of the date of this email. The accepted proprietary version shall incorporate this letter and the proprietary final SE (Enclosure 2), and the accepted non-proprietary version shall incorporate this email and the non-proprietary final SE (Enclosure 1) after the title page. Also, the accepted versions must contain historical review information, including NRC requests for additional information (RAIs) and your responses after the title page. The accepted version shall include an "-A" (designating accepted) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TRs were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Westinghouse will be expected to revise the TR appropriately. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

If you have any questions, please contact the Project Manager for the review, Ekaterina Lenning at 301-415-3151 or via electronic mail at Ekaterina.Lenning@nrc.gov.

Sincerely,

/RA/

Richard Chang, Branch Chief
Licensing Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: 99902038

Enclosures:
1. Final SE (Non-Proprietary)
2. Final SE (Proprietary)

SUBJECT: FINAL SAFETY EVALUATION FOR FINAL SAFETY EVALUATION FOR WCAP-18482-P/WCAP-18482-NP, REVISION 0, "WESTINGHOUSE ADVANCED DOPED PELLET TECHNOLOGY (ADOPT™) FUEL" (EPID: L-2020-TOP-0025) DATED JUNE 13, 2022

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U. S. NUCLEAR REGULATORY COMMISSION

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT

WCAP-18482-P/WCAP-18482-NP, REVISION 0, "WESTINGHOUSE ADVANCED DOPED

PELLET TECHNOLOGY (ADOPT™) FUEL"

DOCKET NO. 99902038 EPID L-2020-TOP-0025

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 8, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20132A014), Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-18482-P/WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel" (Ref. 1) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. By letters dated March 19, 2021 (Ref. 2), June 20, 2021 (Ref. 3), and November 12, 2021 (Ref. 4), Westinghouse supplemented the TR with responses to the NRC staff's requests for additional information (RAI). Westinghouse proposes ADOPT fuel as a direct replacement for standard uranium dioxide (UO_2) fuel. Westinghouse asserts that ADOPT fuel provides enhanced fuel pellet properties to enable higher burnup and improved accident tolerance. ADOPT fuel is a standard UO_2 pellet doped with small amounts of chromium oxide Cr_2O_3 (chromia) and aluminum oxide Al_2O_3 (alumina). Westinghouse requested the chromia content in the range of **[ ]** and alumina in the range of **[ ]** However, Westinghouse has indicated that the ADOPT fuel has a nominal value of **[ ]** chromia and of **[ ]** alumina as additive content. The additives purportedly facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2.

The purpose of the TR 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. This safety evaluation (SE) reviews generic qualifications of the ADOPT fuel material, its properties and performance, and the modeling approach in safety analysis methods, as presented in the TR. The ADOPT fuel is intended to be used with all current NRC-licensed and approved Westinghouse and Combustion Engineering (CE) pressurized water reactors (PWRs). The ADOPT fuel design is intended to be used with two NRC-approved zirconium-based cladding materials - ZIRLO® and Optimized ZIRLO™ - and fuel enrichments up to 5 percent.

This NRC staff review focused on the manner in which additives affected the following major material properties (Section 3, Ref. 1): microstructure, melting temperature, theoretical density, thermal expansion, thermal diffusivity and conductivity, specific heat, grain size and growth, creep, yield stress, modulus of elasticity, strain hardening coefficient and tangent modulus, plastic Poisson's ratio, and rim structure effects. The NRC staff also reviewed the following in-reactor performance concerns (Section 5, Ref. 1) for the use of additive fuel: impact of additives on fuel oxidation resulting in fuel washout when exposed to primary coolant water in the event of fuel failure; impact of additives on fuel melting limits; impact of the additive fuel on reactivity insertion accident (RIA) thresholds; impact of the additive fuel on in-reactor

Enclosure 1

densification; impact of the additive fuel on rod growth; impact of the additive fuel on fission gas release (FGR); impact of the additive fuel on fuel fragmentation, relocation, and dispersal (FFRD); and impact of additive fuel on accident source terms. The NRC staff further reviewed the in-reactor (irradiation) data and used it to examine the performance of additive fuel (Section 4.0 of the TR). The licensing criteria assessment (Section 6.0 of the TR) describes performance of the additive fuel during steady state and anticipated operating occurrences (AOOs) using fuel rod design criteria, safety analyses requirements, and applicable thermal-hydraulic design requirements.

In the TR, Westinghouse uses the most recent NRC-approved fuel performance methodology, as documented in WCAP-17642-P-A (PAD5), to model the mechanical performance of ADOPT fuel (Ref. 5). Key differences in ADOPT fuel from standard UO_2 fuel are higher density and lower fuel densification, both of which Westinghouse modeled via modification to the existing PAD5 input variables. The acceptability of this modeling is discussed below.

Section 2.0 of the SE describes the regulatory basis for the SE. Section 3.0 and its sub-sections contain a technical evaluation of ADOPT fuel: Section 3.1 of the SE focuses on the ADOPT fuel definition; Section 3.2 describes the characterization of ADOPT fuel properties; Section 3.3 describes ADOPT fuel thermal and mechanical properties; Section 3.4 describes irradiation programs and experience with ADOPT fuel; Section 3.5 discusses characterization of ADOPT fuel behavior; Section 3.6 briefly describes ADOPT fuel licensing criteria assessment. Section 4.0 of this SE lists the limitations and conditions (L&Cs).

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel system designs and adherence to General Design Criteria (GDC)-10, “Reactor design,” GDC-27, “Combined reactivity control systems capability,” GDC-28, “Reactivity limits,” and GDC-35, “Emergency core cooling,” is provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition,” (SRP) (Ref. 6), Section 4.2, “Fuel System Design (Ref. 7).” SRP Section 4.3, “Nuclear Design” (Ref. 8) and Section 4.4, “Thermal and Hydraulic Design” (Ref. 9), are also pertinent to the review of fuel systems.

SRP Section 4.2 acceptance criteria are based on meeting the requirements of GDC-10 in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.

GDC 10 states:

> *The reactor core and associated coolant, control, and protection systems shall be designed with the 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.*

GDC 10 establishes specified acceptable fuel design limits to ensure that the fuel is “not damaged.” That means that fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis.

Requirements for analyzing the design-basis loss-of-coolant accident are provided in 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC-35. The most relevant to this review are:

- Per 10 CFR 50.46(a)(1)(i), each boiling or pressurized light water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO® cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents (LOCA) conforms to the criteria set forth in Section 50.46(b). ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for several postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated.

- 10 CFR Part 50, Appendix K, sets forth the documentation requirements for each evaluation model, and establishes required and acceptable features of evaluation models for heat removal by the ECCS.

- GDC-35 requires abundant core cooling sufficient to (1) prevent fuel and cladding damage that could interfere with effective core cooling and (2) limit the metal-water reaction on the fuel cladding to negligible amounts. GDC-35 further requires suitable redundancy of the ECCS, such that it can accomplish its design functions, assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources.

In accordance with SRP Section 4.2, “Fuel System Design” (Ref. 7), the objectives of the fuel system safety review are to provide assurance that:

a. The fuel system is not damaged as a result of normal operation and AOOs,
b. Fuel system damage is never so severe as to prevent control rod insertion when it is required,
c. The number of fuel rod failures is not underestimated for postulated accidents, and
d. Coolability is always maintained.

SRP Section 6.2.1, “Containment Functional Design” (Ref. 10), presents information related to containment integrity following postulated LOCA, steam line, or feedline break accidents as impacted by the ADOPT fuel on the above analyses.

SRP Chapter 15.0, “Transient and Accident Analyses” (Ref. 11), including acceptance criteria for AOOs and postulated accidents and their impact on ADOPT fuel, is addressed in the TR. The review of this TR is based on the acceptance criteria for each of the events described in SRP Chapter 15.

In Section 2.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse provides a roadmap of the TR contents to applicable regulatory guidance, including SRP 4.2. This information is provided to assist the reader and does not require NRC review.

Section 2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes how Westinghouse will expand the limits of applicability for existing NRC-approved TRs to include ADOPT fuel properties and performance, and then defines how licensees will apply these expanded TRs. Justification for the expansion of the NRC-approved TRs is addressed in Section 3.0 of this SE. Section 2.2.2 of the TR concludes that upon approval of this TR, WCAP-12488-A and WCAP-12488-A, Addendum 1-A, which define the Westinghouse fuel criteria evaluation process (FCEP), will be applicable to the fuel designs containing ADOPT fuel at Westinghouse plants.

Approval for Westinghouse's FCEP is limited to nuclear fuel in Westinghouse plants; however, extending the approval of FCEP to CE plants is beyond the scope of this review, because approval for Westinghouse FCEP is limited to nuclear fuel in Westinghouse plants. Thus, with the exception of the FCEP, the NRC staff finds the process for implementing ADOPT fuel via expanded approval of existing TRs acceptable. As noted above, the NRC staff will determine whether this TR can expand the relevant TRs' applicability to include doped pellets below. Approval for Westinghouse FCEP is limited to nuclear fuel in Westinghouse plants.

Section 2.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes anticipated licensee actions to implement ADOPT fuel. Westinghouse states that a license amendment request (LAR) would be required and lists appropriate content for such a licensing action. The NRC staff agrees that a LAR will be required to implement ADOPT fuel. However, given many variants in plants' licensing bases, it is difficult to accurately define the necessary content for future LARs.

3.0 TECHNICAL EVALUATION

Westinghouse has developed ADOPT fuel technology to improve performance and enhance the accident tolerance of UO_2 fuel pellets. ADOPT fuel is a modified UO_2 pellet doped with small amounts of chromia and alumina. The additives are expected to facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2. This review focused on the impact of the additive dopants on major material properties and in-reactor performance.

3.1 ADOPT FUEL DEFINITION

Section 1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, provides the ADOPT fuel definition. This definition consists of nominal dopant concentrations, nominal pellet density, and a range in grain size. The NRC staff's approval of ADOPT fuel is based upon this definition, allowable ranges in composition, including dopant concentrations, and microstructure (described later in this SE), and documented fuel properties and performance.

Section 1.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes ADOPT's extensive in-reactor operating experience, including full batch implementation in European boiling water reactors (BWRs).

Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, defines the range of applicability for this TR. Although Westinghouse requested these limits of applicability, the NRC staff may not find all these limits acceptable. Based on the assessments described in this document, the NRC staff will either restate the limits or establish new limits in Section 4.0 of this SE.

3.2 CHARACTERIZATION OF ADOPT FUEL PROPERTIES

ADOPT Fuel Additives and Microstructure

Section 3.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the impact of the alumina and chromia dopants on densification and grain growth during fuel sintering and, ultimately, on the pellet microstructure. It also describes dopant solubility and residence within the pellet microstructure. In request for information (RAI) 4 (Ref. 2), the NRC staff requested a clarification regarding the distribution and re-distribution of the dopants in fresh fuel, as well as during irradiation. Westinghouse responded that for fresh ADOPT fuel, it measured the radial

concentration of Aluminum and Chromium in an unirradiated pellet, using wavelength dispersive spectrometry (WDS). [

] The NRC staff agrees that the higher vapor pressure of chromia is the reason for redistribution of chromia in ADOPT fuel.

In order to further investigate the distribution of chromia during power ramp, a section of the ADOPT rod was ramp tested to a terminal power of 30 kilowatts per meter (kW/m). Following the ramp, laser ablation spectrometry was performed on the segment. [

]

Westinghouse performed electron probe microanalysis (EPMA) on two irradiated ADOPT rods to investigate the dopant migration under steady-state conditions and found that [
]

In summary, [

] The NRC staff reviewed the irradiation tests and the results of EPMA, laser ablation, and WDS examinations and found that the results are consistent and acceptable.

Based upon the information provided in the TR and response to RAI 4, the NRC staff finds that Westinghouse understands the impact of alumina and chromia dopants on the pellet's microstructure. Understanding microstructure, and the evolution of microstructure under irradiation, is the first step to characterizing the material properties and performance of ADOPT fuel. These are the topics of the next sections.

3.3. THERMAL AND MECHANICAL PROPERTIES

Thermal Properties

3.3.1 Specific Heat

Specific heat (C_p) capacity is important in determining the stored energy for use in transient analysis and loss-of-coolant accident (LOCA) analysis, and in determining thermal conductivity. Section 3.2.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of specific heats of both ADOPT fuel and standard UO_2 fuel pellets.

Specific heat was determined using a differential scanning calorimeter at the Institute for Transuranic Elements (ITU) in Karlsruhe, Germany. The tests were performed with flowing

argon gas at the rate of 0.1 liters per minute (l/min) and a temperature ramp rate of 25 Kelvin per minute (K/min). The referral material used in this measurement was sapphire. The temperature range was 400 – 1400 K, the range at which this measurement technique is most accurate. Two unirradiated ADOPT fuel samples were analyzed and compared with two pure, unirradiated standard UO_2 samples (Figure 3-7 of Ref. 1). The measurements revealed no appreciable difference between the specific heats of ADOPT fuel and reference UO_2 pellets for the temperature range up to 1200°C as per Figure 3-7 in the TR.

The NRC staff reviewed the specific heat measurement and concluded that there is no appreciable difference between specific heats of standard UO_2 fuel and ADOPT fuel as illustrated in Figure 3-7 of the TR.

3.3.2 Thermal Diffusivity and Thermal Conductivity

Thermal conductivity is an important material property that is used to determine the temperature distribution in the fuel rod. Thermal conductivity is determined indirectly by measuring the thermal diffusivity using the laser flash technique. Section 3.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of thermal diffusivity on both ADOPT fuel and standard UO_2 fuel pellets. Thermal diffusivity is measured using the laser flash technique which consists of irradiating the sample material surface with a laser pulse and monitoring the temperature rise of the material using a photovoltaic infra-red detector.

The thermal diffusivity, α, is calculated from $\alpha = k \times \frac{L^2}{t_{1/2}}$

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

Thermal conductivity, $\lambda = \alpha \times \rho \times C_p$,

where ρ is the density and C_p is its specific heat.

Thermal diffusivity of unirradiated ADOPT fuel was measured at the KTH Royal Institute of Technology in 1999 and was compared with an unirradiated standard UO_2 sample. Measurements were taken between 20°C and 1400°C in approximately 100°C increments during heating while correcting for thermal expansion of the samples. Figure 3-8 of the TR compares thermal diffusivity of ADOPT fuel and UO_2 fuel and it shows no appreciable difference between the thermal diffusivities of ADOPT fuel and UO_2 fuel.

Based upon these measurements, Westinghouse concludes that there is no appreciable difference in the thermal diffusivity of standard UO_2 fuel and ADOPT fuel. The NRC staff reviewed the above information and based on this information; the NRC staff finds this conclusion acceptable.

3.3.3 Melting Temperature

Fuel melting temperature is a safety limit defined in a plant's Technical Specifications, an important property in safety analyses, and evaluated against accident analyses. Section 3.2.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of fuel melting temperature on both ADOPT fuel and standard UO_2 fuel pellets simultaneously using a pyrometer and a spectrometer, providing both the true temperature and the spectral emissivity

function of a specimen. The emissivity was used to convert the temperature brightness into the true value. The reliability of the procedure was confirmed by the melting point measurement on stoichiometric reactor-grade UO_2, which was within 0.5 percent of the recommended value. Figure 3-9 of the TR shows melting temperatures of ADOPT fuel and UO_2 fuel during melting of the two specimens using laser-pulse. This figure shows that there is no appreciable difference between the measured melting temperature, 3122 ± 7 K, and the measured value of the reference UO_2.

Based upon review of these measurements, the NRC staff concludes that there is no appreciable difference in the melting temperature of standard UO_2 fuel and ADOPT fuel.

<u>3.3.4 Thermal Expansion</u>

Thermal expansion changes a fuel’s volume, and, thus, density at a given temperature relative to a standard temperature. Thermal expansion is an important consideration in pellet cladding mechanical interaction (PCMI). Section 3.2.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of thermal expansion on both ADOPT fuel and standard UO_2 fuel pellets.

Thermal expansion of unirradiated ADOPT fuel samples and unirradiated UO_2 were measured at the ITU in Karlsruhe, Germany according to ASTM-E831-86, the “Standard Test Method for Linear Thermal Expansion of Solid Materials by Thermomechanical Analysis” (Ref. 22). Data was collected over a temperature range from 20°C - 1490°C with a heating rate of 5°C/min. Figure 3-10 of TR shows thermal expansion of ADOPT fuel and UO_2 fuel and shows no appreciable difference between thermal expansions of the two specimens.

Based upon the review of these measurements and the results, the NRC staff concludes that there is no appreciable difference in the thermal expansion of standard UO_2 fuel and ADOPT fuel.

Mechanical properties

<u>3.3.5 Modulus of Elasticity</u>

Westinghouse reports that the impact of the discussed dopants on the elastic moduli can be determined using the same procedure employing the rule of mixtures as the calculation of theoretical density (TD) in Section 3.1.4 of the TR. Based on that calculation, the addition of the specified nominal amount chromia **[ ]** and alumina **[ ]** to UO_2 will have no significant effect on the elastic properties of ADOPT fuel compared to standard UO_2 fuel. Fuel temperature has a much more significant impact on the elastic moduli.

The NRC staff reviewed the impact of dopants on the elastic moduli of ADOPT fuel and based on the above-described calculation, determined that the dopant has no significant effect on the elastic moduli.

<u>3.3.6 Creep</u>

Section 3.3.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes creep and hardening testing performed on both ADOPT fuel and standard UO_2 fuel pellets. 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 megapascals (MPa), 45 MPa, and 60 MPa). The

measurements revealed a classical creep curve with a strong temperature dependency, such that measured strain increases dramatically with rising temperature, and sensitivity to applied stress. Figure 3-11 of TR illustrates this fact. At temperatures greater than 1500°C, the ADOPT fuel exhibited higher viscoplasticity as compared to the reference UO_2 pellets, meaning it is more resistant to creep. This is illustrated in figure 3-12 of TR. At temperatures lower than 1300°C, there appears to be no creep benefit relative to the reference UO_2. Thus, in steady-state operation, the creep behavior between ADOPT fuel and standard UO_2 fuel shows no appreciable difference because radial average fuel temperature remains below 1300°C.

Hardening tests were performed for ADOPT fuel and UO_2 fuel 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. Hardening tests showed a strong temperature dependency to the applied strain rate. Figure 3-13 in the TR shows that ADOPT fuel is more ductile than the standard UO_2 fuel, which means that ADOPT fuel requires less stress than standard UO_2 fuel to achieve a given strain rate. This shows more viscoplasticity capability for ADOPT fuel in the strain levels of interest for pellet clad interaction (PCI). Thus, the specimen will show rate-dependent inelastic behavior, meaning material will undergo unrecoverable deformations when a load level is reached.

The NRC staff reviewed the details of the test process and the results of both creep and hardening tests performed on standard UO_2 fuel and ADOPT fuel pellets and the staff determined that the creep behavior and hardening behavior of ADOPT fuel is acceptable because: (1) with regard to creep behavior, ADOPT fuel and standard UO_2 fuel have the same behavior below 1300°C and the ADOPT fuel will operate at temperatures less than 1300°C; and (2) with regard to hardening, ADOPT fuel will require less stress than standard UO_2 fuel to achieve a given strain rate at operating temperatures.

3.4 IRRADIATION PROGRAMS AND EXPERIENCE

Section 4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes long-term irradiation programs, in-pile power ramp testing, and subsequent hot-cell examinations to characterize the irradiated properties and performance of ADOPT fuel. Many of the observations, measurements, and findings of these programs (e.g., steady-state FGR, densification, thermal expansion) were used in subsequent sections of the TR (and corresponding sections of this SE) to justify irradiated material properties, performance, and analytical models.

Section 4.1.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, provides details of the ramp and bump testing conducted at the Studsvik R2 research reactor of irradiated fuel rod segments of ADOPT fuel and standard UO_2 fuel. Following the ramp and bump testing, the rodlets were punctured to measure the FGR of the two pellet types. Based on the measurements, Westinghouse concludes that ADOPT fuel has lower transient FGR compared to standard UO_2 fuel.

Section 4.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes ramp testing conducted as part of Studsvik SCIP II program. Based upon FGR measurements on the two ADOPT fuel and two UO_2 fuel rod segments (parent fuel rods irradiated at Oskarshamn 3 NPP), Westinghouse concludes that the same trend described in the previous paragraph, i.e., lower FGR for ADOPT fuel, is apparent. The NRC staff questions this conclusion because of differences in ramp terminal powers (and fuel temperatures) between the ADOPT fuel and UO_2 fuel segments.

Westinghouse explains the improved FGR retentions as follows:

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.

In Section 6.1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse acknowledges these improved transient FGR characteristics, but maintains the existing UO_2 transient FGR model for ADOPT fuel. In general, FGR measurements exhibit a large variance, even for identical fuel designs operating with similar power histories. Data from the R2 ramp and bump testing suggests improved fission gas retention. The data from the SCIP II program has less pedigree due to ramp power differences. Given the inconsistent data, the NRC staff reaches no conclusion about ADOPT fuel's benefits over standard UO_2 fuel with regard to FGR. That said, because Westinghouse is using the existing UO_2 transient FGR model, it is not attempting to take credit for these purported benefits. As the existing model is more conservative than a new model attempting to credit FGR benefit, it bounds ADOPT fuel. Therefore, the NRC staff finds that the use of the existing UO_2 transient FGR model for ADOPT is acceptable.

3.5 CHARACTERIZATION OF ADOPT FUEL BEHAVIOR

Section 5 of WCAP-18482-P/WCAP-18482-NP, Revision 0 (Ref. 1), describes the empirical database used to characterize the performance of ADOPT fuel pellets during normal operations, AOOs, and postulated design-basis accidents (DBAs).

3.5.1 Corrosion and Washout Characteristics

Section 5.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes: (1) high-temperature furnace tests on unirradiated fuel specimens to characterize fuel oxidation performance, and (2) in-pile, irradiated testing on damaged fuel rod segments to characterize erosion (washout) performance. The data show enhanced fuel corrosion resistance and slightly better washout performance relative to standard UO_2 fuel. Fuel corrosion and washout performance are not modeled as part of any AOO or postulated DBA safety analysis. However, these performance aspects are important when operating a reactor with damaged fuel (i.e., leakers). Based upon these furnace tests and in-pile tests, the NRC staff finds ADOPT's oxidation and washout performance acceptable and well characterized.

3.5.2 Swelling Behavior (Pellet Densification and Rod Growth)

Section 5.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes re-sintering tests performed on unirradiated ADOPT fuel pellets to characterize fuel densification. Re-sintered pellet densification measurements were collected and analyzed on **[ ]** ADOPT fuel pellets. There is a clear difference in the densification of standard UO_2 and ADOPT fuel pellets. ADOPT fuel pellets exhibit less densification compared with standard UO_2. The impact of the change in densification will be explicitly accounted for in fuel performance and safety analyses. Based upon these re-sintering tests, the NRC staff finds ADOPT's densification behavior acceptable and well characterized.

Section 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes Westinghouse's fuel rod growth measurements on both BWR and PWR ADOPT fuel rods. An important impact of ADOPT fuel's reduced in-reactor densification is an earlier closure of the fuel pellet-to-cladding gap. After gap closure, irradiation-induced fuel swelling will influence fuel rod growth. The empirical database clearly shows an increase in fuel rod growth compared to standard UO_2 fuel rods. In response to an RAI regarding the continued applicability of the PAD5 upper bound (UB) growth model (RAI 7a, Ref. 2), **[**

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In a revised response to RAI 7 (Ref. 4), Westinghouse proposed an additive term applied to the standard UO_2 fuel rod growth model to account for the increased growth exhibited by the ADOPT fuel rods. The revised model provides an improved upper and lower bound prediction of rod growth when compared to the data. Based on the supplemental response to RAI 7, the NRC staff finds the augmented fuel rod growth model acceptable.

<u>3.5.3 Steady State FGR Database</u>

Section 5.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes FGR measurements following steady-state operation at two different commercial reactors. Symmetric rods containing both standard UO_2 fuel and ADOPT fuel were selected to minimize uncertainties associated with power history and fuel temperature. This information was not used to provide an absolute measurement to re-calibrate models, but instead, used as a relative comparison of the two fuel types. Based on the FGR measurements, Westinghouse concludes **[**

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No explanation for the FGR observations is provided in Section 5.3 of the TR. However, earlier in the TR, Westinghouse states that the FGR behavior is a combination of two competing effects. First, the enlarged grains of the ADOPT fuel pellets create longer diffusion paths for fission products precipitated within the grains. This is beneficial to the FGR retention of the pellets. Second, as a result of the additives, the gas diffusion rate is enhanced, which tends to increase the rate of FGR. In addition to these competing phenomena, the larger grains may also promote an earlier grain boundary saturation which would tend to increase FGR.

In response to an RAI regarding the qualification of the pool-side gamma scanning technique used to measure FGR (RAI 8, Ref. 2), Westinghouse provided comparisons of pool-side measurements against hot-cell destructive testing performed on the same or symmetrical fuel rods. Comparison of the data reveals good agreement. During the audit of WCAP-18482-P/WCAP-18482-NP, Revision 0 (Ref. 25), the NRC staff reviewed the underlying Westinghouse Electric Sweden AB report documenting the FGR measurements, hot-cell investigation, and quantification of uncertainties. Review of this report provided further confidence in the accuracy of the pool-side technique.

Based upon the above information presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, as supplemented, the NRC staff finds ADOPT's FGR behavior acceptable and well characterized.

3.5.4 Fuel Fragmentation Relocation and Dispersal

Higher burnup fuel pellets with an established high burnup structure (HBS) have been shown to be more susceptible to fine fuel fragmentation. Section 5.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the susceptibility of ADOPT fuel pellets to FFRD. **[**

] As described in WCAP-18482-P/WCAP-18482-NP, Revision 0, fuel becomes more susceptible to fuel fragmentation at elevated burnup. Thus, the susceptibility of ADOPT fuel to FFRD will need to be re-addressed should Westinghouse seek approval for higher fuel burnup beyond current limits.

A delay in the formation of the HBS would be beneficial with respect to FFRD susceptibility. Section 5.1.3.2 of Reference 28 describes the formation of the HBS and compiles data and observations from several investigations. Rim width measurements as a function of local burnup (Figure 14, Ref. 28) show a delayed formation for samples with a larger manufactured grain size (similar to the ADOPT's initial grain size). Thus, ADOPT fuel would likely experience delayed HBS formation, and thus may experience decreased FFDR susceptibility, relative to standard UO_2 fuel. **[**

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Therefore, based on these irradiated fuel studies, the staff finds that ADOPT fuel is not more susceptible to FFRD.

The regulatory framework with respect to fuel's susceptibility to FFRD is the subject of ongoing regulatory initiatives associated with licensing higher fuel burnup limits. Based on the above discussion, which shows that ADOPT fuel is not more susceptible to FFRD, the NRC staff has confidence that the introduction of ADOPT fuel does not pose potential safety concerns associated with FFRD phenomena and is therefore acceptable.

3.5.5 Reactivity Initiated Accidents

Section 5.5 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the anticipated performance of ADOPT fuel pellets under postulated RIA conditions, such as those anticipated during certain postulated control rod ejection (CRE) accident scenarios. Phenomena such as the sensitivity of PCMI cladding failure threshold to cladding hydrogen content and orientation and the potential impact of ADOPT fuel properties on the fuel enthalpy threshold for incipient fuel melting are described. The discussion presents a common, holistic CRE methodology with a consistent regulatory and technical bases employed throughout the Westinghouse and CE nuclear fleet and that this common methodology remains applicable to ADOPT fuel. However, because CRE analytical methods (i.e., models, inputs, assumptions) and acceptance criteria vary significantly among plants which may adopt and implement ADOPT fuel, the NRC staff is unable to reach a safety finding.

Identifying all of the variants in CRE analytical models, methods, and acceptance criteria among all plants which may adopt and implement ADOPT fuel and expanding the approval of these methods and acceptance criteria to ADOPT fuel is beyond the scope of this TR. Furthermore, these legacy analytical models, methods, and acceptance criteria may not account for all relevant fuel burnup and cladding corrosion related phenomena. As such, licensees, as part of their license amendment request for deploying ADOPT fuel, will need to justify that their CRE

methods (i.e., models, inputs, assumptions) and acceptance criteria are applicable to and appropriate for ADOPT fuel. The following L&C captures this requirement:

> L&C 1: Licensees must demonstrate that the CRE analytical models, methods, and acceptance criteria are applicable to fuel designs containing ADOPT pellets and capture all relevant fuel burnup and cladding corrosion related phenomena.

Regulatory Guide (RG) 1.236, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents" (Ref. 30), provides guidance including acceptable inputs and assumptions, fuel rod cladding failure thresholds, and damaged core coolability criteria for analyzing the postulated CRE accident. The staff asked Westinghouse during a clarification call if Westinghouse planned on (1) incorporating RG 1.236 guidance as part of the ADOPT methodology, and (2) justifying the use of this guidance for ADOPT fuel. Westinghouse would not commit to incorporating this guidance as part of a common CRE methodology for ADOPT fuel.

Section C.1.1.1 of RG 1.236 (Ref. 30) states that the applicability of this guidance to future light water reactor (LWR) fuel rods designs (e.g., doped pellets, changes in fuel pellet microstructure or density, changes in zirconium alloy cladding microstructure or composition, coated zirconium alloy cladding) will be addressed on a case-by-case basis. In Section 5.5 of WCAP-18482-P/WCAP-18482-NP, Revision 0, and in response to the RAI 9 (Refs. 3 and 4), Westinghouse provided evidence that the performance of ADOPT fuel under RIA conditions is similar to standard UO_2 fuel. Given its similar performance, Westinghouse claimed that the applicability of the RG 1.236 guidance should be expanded to ADOPT fuel.

Section C.2 of RG 1.236 provides acceptable analytical inputs, assumptions, and methods. With the possible exception of transient FGR (TFGR), none of the acceptable analytical inputs, assumptions, and methods are specific to fuel material properties and anticipated performance. Hence, this guidance would be acceptable for ADOPT fuel. The TFGR correlations described in Appendix B of RG 1.236 are based on a large, diverse empirical database comprised of many different fuel types. Based on the large, diverse empirical database which forms the bases of these conservative correlations, the NRC staff finds the use of the TFGR correlations to ADOPT fuel acceptable.

Section C.3 of RG 1.236 defines the following three distinct fuel rod cladding failure thresholds:

- High-Temperature Cladding Failure Threshold (Section C.3.1 of RG 1.236)
  - Because ductile failure depends on cladding temperature and differential pressure (i.e., RIP minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (MPa).
- PCMI Cladding Failure Threshold (Section C.3.2 of RG 1.236)
  - Because fuel cladding ductility is sensitive to hydrogen content, zirconium hydride orientation, and initial temperature, separate PCMI failure curves are provided for fully recrystallized annealed (RXA) and stress relief annealed (SRA) cladding types at both low initial cladding temperature conditions (i.e., below 500°F down to BWR cold startup) and high initial cladding temperature conditions

(i.e., at or above 500°F). The failure threshold curves are expressed as change in radial average fuel enthalpy (Δcal/g) versus fuel cladding excess hydrogen content.

- Molten Fuel Cladding Failure Threshold (Section C.3.3 of RG 1.236)

    - Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions. Fuel melt calculations are sensitive to burnup characteristics, prompt pulse characteristics, and local fuel melting temperature.

With respect to high-temperature and molten fuel cladding failure, ADOPT fuel properties have a negligible impact on the ability of analytical models to predict local burnup, initial RIP, local transient power, and local transient fuel temperature. Sections 2 and 6 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describe the applicability of existing Westinghouse nuclear design and fuel thermal-mechanical design models to ADOPT fuel. Based on the ability of analytical models to predict these parameters, the NRC staff finds that these cladding failure thresholds are applicable to ADOPT fuel designs.

With respect to PCMI cladding failure, the performance of ADOPT fuel has the potential to impact important initial fuel rod conditions (e.g., fuel-to-cladding gap size) and local transient conditions (e.g., fuel thermal expansion, cladding strain). With respect to ADOPT fuel's enhanced creep performance, Westinghouse states that diffusion driven processes do not have time to occur during the short RIA pulse, and so the greater high-temperature fuel creep does not have time to reduce the clad stress. Based on a review of the information provided, the NRC staff agrees that ADOPT's enhanced fuel creep will not impact fuel performance during the postulated CRE accident.

Sections 3 and 4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describe the impact of the dopant addition on fuel pellet densification and pellet-to-cladding gap size. A smaller initial gap size would have a detrimental impact on RIA PCMI performance since it allows less fuel volumetric expansion prior to cladding contact. But this effect disappears when the fuel pellet contacts the cladding due to irradiation-induced fuel swelling (and cladding creep down) at approximately middle-of-life.

While the implementation of the PCMI cladding failure curves requires analytical models capable of predicting both local fuel enthalpy and cladding hydrogen content, the development of the failure thresholds was based on in-pile prompt pulse testing of irradiated fuel rod segments. To date, only a single in-pile prompt pulse test has been conducted on ADOPT fuel. The NRC staff reviewed **[**

] The paper details two Nuclear Safety Research Reactor (NSRR) prompt power tests conducted on chromia doped fuel and ADOPT fuel. Fuel specimen and testing specifications are shown below (extracted from the technical paper).

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Test rod LS-4 did not fail with a maximum increase in fuel enthalpy of 133 Δcal/g. Test rod OS-1 failed at 38 Δcal/g, which is below all previous test failures for rods close in burnup and hydrogen content. The paper concluded with the following observations:

> *The pre- and post-test examinations suggested that one of the reasons of the lower failure limit may be the effect of the hydrides radially oriented and precipitated more densely in the specific angle range in the cladding tube. However, since the possible contribution of ADOPT™–pellets specific effects cannot be ruled out at the present, further investigation is needed on fuel pellet behavior under both normal-operation and pulse-irradiation conditions.*

Figure 5-1 shows the OS-1 test plotted against the RG 1.236 RXA PCMI cladding failure threshold presented in Reference 1 and the supporting empirical database. Note that OS-1 was BWR Zry-2 cladding with a liner. Examination of the figure reveals that the OS-1 failure enthalpy is below the RG 1.236 failure threshold curve.

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Japan Atomic Energy Agency (JAEA) investigated the failure of OS-1 and made the following observations:

> *The morphology of the hydrides precipitated in the fuel cladding of OS-1 was investigated by metallography and compared with previous results obtained in JAEA in connection focusing fuel failure limit. It was suggested that the observed lower limit of fuel failure was related to the amount and length of the hydride precipitated along the radial direction of cladding.*

Based on the performance of OS-1, the NRC staff had concerns that ADOPT fuel may (1) introduce differences in cladding zirconium hydride morphology, and (2) behave differently under prompt pulse conditions. In response to a request to provide evidence that differences between standard UO_2 fuel and ADOPT fuel performance (i.e., steady-state swelling, PCMI) will not introduce differences in cladding properties and microstructure (RAI 9a, Ref. 3), Westinghouse provided micrographs showing the hydrogen morphology of irradiated fuel rod cladding for both standard UO_2 fuel and ADOPT fuel. Westinghouse states that the images appear to show a consistent ratio of radial to tangential hydrides around the rod. Therefore, ADOPT fuel's increased density, and earlier-in-life pellet-clad contact, does not appear to contribute any excessive radial hydride reorientation. Westinghouse also provided results from expansion due to compression testing on irradiated fuel cladding segments which showed that both the standard UO_2 fuel and ADOPT fuel cladding survived relatively large strains at PWR operating temperatures.

In a response to an RAI to characterize the fuel thermal expansion of ADOPT fuel or similar large grain fuel pellets (RAI 9b, Ref. 3), Westinghouse summarized the tests results from NSRR Tests OI-10 (28-micron grain size), MR-1 (40-micron grain size), and LS-4 (50–60-micron grain size). Based on the measured residual cladding strain in OI-10 and MR-1, Westinghouse claimed that the data suggests that pellet expansion was largely generated by solid thermal expansion, and that the fission gas swelling associated with the enhanced fission gas retention of large-grained fuels (like the ADOPT fuel pellet) was minimal. And while the larger grained LS-4 did experience additional cladding strain, it was not unexpected given the larger power pulse. Based on this limited data set, Westinghouse concluded that there is no information to suggest that additives or large grains have a negative impact on the fuel rod's susceptibility to PCMI failure.

Based on the information described above, the NRC staff accepts the premise that the low failure enthalpy of Test OS-1 was likely caused by the cladding hydride morphology and not an intrinsic effect of the dopant (or large grain structure) on fuel thermal expansion characteristics. Given that the empirical database supporting the PCMI failure curves consists of a wide range of fuel designs and operating experience, including the three tests on large grain specimens identified above, the NRC staff finds that the RG 1.236 PCMI cladding failure thresholds are applicable to ADOPT fuel designs.

Sections 4 and 5 of RG 1.236 are independent of fuel design and hence would be applicable to ADOPT fuel. Section 6 of RG 1.236 provides guidance associated with maintaining a known geometry amenable to continued core cooling. The criteria consist of an empirical-based fuel rod fracture limit on peak fuel enthalpy and an analytical limit on fuel melting. Westinghouse states that the minor doping additions will not affect the fissile isotope consumption of U-235 and production of Pu-239. Thus, there will be no impact on the local, transient power, and fuel temperatures experienced during the prompt pulse. In addition, the minor doping additions have a negligible impact on fuel thermal conductivity and fuel melting temperature. Thus, the margin to fuel melting is not impacted. As described earlier, the Westinghouse core physics, and fuel rod models have been shown to be applicable to ADOPT fuel. Hence, predictions of deposited energy and fuel temperatures will account for ADOPT fuel properties. Based on these considerations, the NRC staff finds the coolable geometry criteria applicable to ADOPT fuel designs.

Based on the discussion above, the NRC staff found that RG 1.236 is applicable to the ADOPT fuel designs.

3.5.6 Comparison of Doped UO_2 Fuel Properties and Performance

In 2018, the NRC reviewed and approved a revised set of Framatome analytical models and methods for their chromia-doped UO_2 fuel pellet design (Ref. 26). While the proprietary information in the Framatome submittal may not be used as a basis for the staff's approval of Westinghouse's ADOPT fuel pellet, it is a valuable source of independent data and performance trends which helped focus the staff's review.

Key fuel properties and performance aspects of the two doped UO_2 fuel designs were compiled and compared in the NRC memorandum, "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment" (Ref. 28). This direct comparison provided either independent confirmation or focused attention on fuel properties and performance where deviations were identified. For the latter, RAIs were developed to better explain different trends.

3.5.7 Evaluation of Fuel Burnup Limit

In Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse defined the following constraint for ADOPT fuel applications:

Fuel burnup up to **[ ]** 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.

The bases for establishing a limitation on fuel burnup lies with the extent of in-reactor operating experience and with the empirical database supporting ADOPT fuel's properties and analytical models. Section 1.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, details the extensive in-reactor operating experience Westinghouse has with ADOPT fuel in European commercial reactors. As discussed below, the NRC staff has reviewed this operating experience and determined that, with significant reload quantities of ADOPT fuel, this operating experience supports a fuel burnup limit of 62 MWd/kgU.

Section 3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the characterization of ADOPT pellets' microstructure and fuel properties. Most of these observations and material property measurements were based on the unirradiated state of the ADOPT fuel pellets. With respect to the empirical database supporting the burnup-dependent irradiated material properties and analytical models, the NRC staff compiled the following summary:

- Barseback 2 fuel rods irradiated up to 33.5 MWd/kgU segment average burnup:
  - Data on rod profilometry used to validate fuel swelling model.
  - Ramp and bump test data used to validate transient FGR and fuel thermal expansion models.
- Oskarshamn 3 fuel rods irradiated up to 60 MWd/kgU rod average burnup:
  - Destructive PIE data used to validate FGR model and characterize pellet cracking, HBS formation, cladding stress (hydride orientation).
- SCIP II Ramp Testing conducted on fuel rods up to 54 MWd/kgU segment burnup:
  - Ramp test data used to validate transient FGR and fuel thermal expansion models.
- **[ ]** Commercial irradiation of fuel rods up to 72 MWd/kgU rod average burnup:
  - Data used to validate rod growth, profilometry, and FGR models.
- Halden IFA-677 irradiation on fuel segments up to 26 MWd/kgUO$_2$ average burnup:
  - Data used to validate fuel temperature model.
- Section 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the empirical database used to validate the fuel rod growth. The data shows that fuel rod growth is well characterized and predictable up to 62 MWd/kgU rod average burnup.
- Section 5.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the empirical database used to validate the FGR model. The data shows that steady-state FGR is well characterized and predictable up to 62 MWd/kgU rod average burnup.

Based upon the extent of the empirical database information supporting the irradiated material properties and analytical models described above, the NRC staff finds that ADOPT fuel is acceptable for use up to a rod average burnup of 62 MWd/kgU. The additional information needed to support higher burnups (mentioned in Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0) was not made available to the staff; therefore, the NRC staff was unable to reach a finding as to the performance of ADOPT fuel beyond 62 MWd/kgU. Consequently, the staff can only approve WCAP-18482-P/WCAP-18482-NP, Revision 0, with a limitation and condition limiting ADOPT Pellet use to 62 MWd/kgU, **[ ]** as Westinghouse originally requested. Although Westinghouse's original proposed limitation and condition did require NRC approval of additional information before the NRC approval permitted the use of ADOPT fuel at burnups higher than 62 MWd/kgU; the NRC staff does not consider the limitation and condition sufficient as originally written because it leaves ambiguity as to exactly when burnups higher than 62 MWd/kgU are permissible.

If Westinghouse decides to pursue higher fuel burnup, additional information will be required to remove or alter this limitation and condition in accordance with the Fuel Design Constraints listed in Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0.

3.6 LICENSING CRITERIA ASSESSMENT

3.6.1 Steady State and AOO Analysis

Fuel Performance Models and Methods

Westinghouse Performance Analysis and Design Model (PAD5) (Ref. 5) is the fuel rod design tool which incorporates relevant fuel performance phenomena such as fuel thermal conductivity degradation with fuel burnup (TCD), and FGR and swelling at high burnup. PAD5 calculates fuel performance parameters such as, cladding stress, strain, oxidation and hydriding, fuel temperatures and volume changes, and RIP.

It has been shown that the differences in ADOPT fuel and standard UO_2 fuel properties (Section 3.3 of SE) are negligible. The corrosion and creep data presented in Section 5.1 of the TR shows that ADOPT fuel has improved resistance against post-failure degradation and increased PCI margins in comparison to undoped UO_2 pellets. PAD5 modeling of ADOPT fuel has shown that it has higher density and lower fuel densification which are analyzed by modifying existing PAD5 input variables. The lower densification of ADOPT fuel can be explicitly modeled with PAD5 densification model as described in Section 5.7.1 of Reference 5.

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3.6.2 Fuel Rod Design Criteria

The fuel rod design criteria ensure the fuel rods perform their intended function throughout the lifetime of the fuel. Section 7.4 of WCAP-17642-P-A TR (Ref. 5) provides key criteria that impact the Westinghouse fuel performance:

- Clad stress
- Clad strain
- Fuel RIP
- Cladding fatigue
- Cladding oxidation
- Cladding hydrogen pickup
- Axial growth
- Cladding free standing
- Pellet overheating
- Pellet-cladding interaction
- Interface to other Safety analysis

Among the above listed fuel rod design criteria, clad oxidation, clad hydrogen pickup, and clad free standing do not depend on pellet properties/models. The impact on the affected criteria is discussed in this section.

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] 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 Reference 5. Figures 6-4 and 6-5 in WCAP-18482-P/WCAP-18482-NP, Revision 0, show fuel centerline temperatures for high burnup and twice-burned assembly of an uprated 3-loop plant with 15x15 fuel. These figures show differences in temperature, and consistently lower temperatures, for ADOPT fuel than that for standard UO_2 fuel. The lower ADOPT fuel temperature is due to the ADOPT fuel closing the gap slightly early due to lower densification relative to standard UO_2 fuel. Early pellet-clad contact improves heat transfer from the fuel thereby lowering fuel temperatures.

3.6.2.1 Clad Stress

Section 7.4.1 of Ref. 5, stipulates the criteria that the fuel rod shall not be damaged due to excessive fuel clad stress. The maximum cladding stress intensities excluding PCI induced stress is evaluated based on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) guidelines. Stresses in the cladding are combined to calculate a maximum stress intensity which is then compared to the criteria set forth in Section 7.4.1 of

Ref. 5. The ASME BPVC limits are designed to protect pressure vessels from unconstrained deformation due to high stresses.

The NRC staff concluded that the use of this approved methodology is acceptable to demonstrate compliance with the acceptance limit for clad stress.

3.6.2.2 Clad Strain

The acceptance criteria limit for clad strain is that the total tensile strain, elastic plus plastic, due to uniform cylindrical fuel pellet deformation during any single Condition I or II transient shall be less than 1 percent from the pre-transient value. Transient clad strain is caused by a rapid thermal expansion and fission gas swelling of the fuel pellet during a short-term overpower event.

Section 4.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes a base irradiation and subsequent ramp testing program conducted on two BWR, barrier lined, segmented rods containing standard UO_2 fuel (D0) and two different variants of doped UO_2 fuel (D1 and D3). Ceramography (D0, D1), FGR measurements (D0, D1, D3), and fuel volume change (D0, D1) are presented. Based on the predicted volume change (D0 versus D1), Section 6.1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, concludes that 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.

Section 4.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes ramp testing performed on ADOPT fuel rod segments as part of the Studsvik SCIP-II program. Two ADOPT fuel rod segments (WaL, WaH) and two standard UO_2 fuel rod segments (WsL, WsH) were ramp tested.
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Based on its review of two previous fuel designs involving large grain UO_2 fuel pellets from other fuel vendors, the NRC staff has acquired knowledge of fuel performance differences between large grain UO_2 fuel pellets and standard UO_2 fuel pellets. Specifically, a large empirical database of ramp tests on irradiated fuel rod segments exists and demonstrates that the larger grains promote an increase in incremental diametral cladding strain relative to standard UO_2 fuel pellets. Due to the limited ramp data presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, the staff requested additional information to demonstrate the relative performance of ADOPT fuel and PAD5's ability to predict its performance. In response to RAI 11 (Ref. 3), Westinghouse provided measured and predicted incremental diametral cladding strain for the ramp tests described in Sections 4.1 and 4.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0. Below are the NRC staff's observations on this information:

D1 Step Ramp Test

- D1 does not fully represent ADOPT fuel pellet composition **[**

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- Difference in ramp profile between D0 and D1. Due to a malfunction at the test facility, D0 rod segment held for 12 hours, whereas D1 rod segment held for 7.7 hours.

- After accounting for initial conditions (i.e., pre-ramp, base irradiation fuel volume), the calculated fuel volume change was larger for D0 relative to D1 (Figure 4-6 of WCAP-18482-P/WCAP-18482-NP, Revision 0).
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<u>D3 Bump Ramp Test</u>

- D3 does not fully represent ADOPT fuel pellets composition **[** **]**
  - D3 contains **[** **]**
- D3 segment burnup average of 30 - 33.5 GWd/MTU, never exceeding 8 KW/ft.
- No cladding profilometry or fuel volume change estimates provided in WCAP-18482-P/ WCAP-18482-NP, Revision 0.
- **[**

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<u>SCIP-II Ramp Test</u>

- Not clear whether WaL and WaH represent ADOPT fuel pellets composition.
- WaL at 47 GWd/MTU with an initial grain size of 31.4 microns.
- WaH at 54 GWd/MTU with an initial grain size of 43.3 microns.
- No cladding profilometry or fuel volume change estimates provided in WCAP-18482-P/ WCAP-18482-NP, Revision 0.
- No relative comparison can be drawn because of variations in the ramp terminal power (WsL at 10.9 KW/ft, WaL at 13.3 KW/ft; WsH at 9.4 KW/ft, WaH at 11.5 KW/ft).
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Westinghouse's response to the RAI 11 also includes a discussion about the effect of time on gas diffusion. Westinghouse concludes that the application of the PAD5 fission gas swelling model for the short transient is also justified based on the necessary time for the gas atom to travel to the large intragranular bubble site and grain boundary and the delayed development of HBS. The NRC staff was not convinced by this line of reasoning since large intragranular fission gas bubbles exist prior to the AOO overpower event and that their presence impacts the overall fuel pellet thermal expansion and cladding strain.

Section A.2.4.3 of WCAP-17642-P-A (Ref. 5) describes the derivation of an UB, additive uncertainty term on the PAD5 cladding diameter change prediction. The empirical database

used to derive this uncertainty term is shown in Figure A.2.4-4 of WCAP-17642-P-A (provided below for convenience). **[**

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The PAD5 predictions of the D1 and D3 ramp tests appear well behaved as they follow trends in measured cladding diameter change along the axial length of the test specimen. **[**

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In a response to the request to clarify Westinghouse position on the ability of PAD5 to predict ramp data (RAI 11, Ref. 4), Westinghouse provided measured versus predicted cladding strain for the entire ramp database, including the doped fuel data. **[**

] Based on the information provided in response to the RAI 11, the NRC staff finds the PAD5 code

acceptable for calculating diametral cladding strain as long as the larger uncertainty term is applied.

3.6.2.3 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 (clad liftoff) due to outward creep during normal operations,
- result in hydride reorientation in radial direction, and
- preclude extensive departure from nucleate boiling (DNB) propagation.

ADOPT fuel has a slight reduction in FGR because of the slightly lower fuel temperatures relative to standard UO_2 fuel pellets, therefore the reduction in volume is more significant. Figure 6-7 of WCAP-18482-P/WCAP-18482-NP, Revision 0, compares RIP against rod average burnup and shows that the ADOPT fuel RIP is higher than the standard UO_2 fuel at the end of life (EOL). The RIP of ADOPT fuel is expected to be consistently higher than the standard UO_2 fuel under the same conditions.

The NRC staff determined that since the differences in ADOPT fuel and standard UO_2 fuel are not significant with respect to RIP, no clad liftoff design criteria can be accommodated.

Departure from Nucleate Boiling (DNB) Propagation

DNB propagation is investigated on a mechanistic basis to meet fuel rod burst and ballooning limits. The analysis is performed using the VIPRE code (Ref. 19) and using the methodology as prescribed in Reference 32. The RIP will be calculated using PAD5. No other features of the ADOPT fuel pellets will affect the rod burst or ballooning calculations, or the DNB propagation evaluation.

The NRC staff determined that the same procedures for DNB propagation are used for standard UO_2 fuel and for ADOPT fuel.

Clad Hydride Reorientation

Hydride reorientation occurs when hydride precipitates formed during reactor operation reorient from the circumferential to the radial direction. The radial hydrides can reduce the cladding ductility and increase the potential for brittle failure during fuel rod handling. The RIP analysis performed for no-liftoff criterion I confirms that the threshold pressures for hydride reorientation are not exceeded. This analysis is performed using the approved PAD5 fuel performance methodology. No clad liftoff is confirmed on a cycle specific basis. Analyses have shown that ADOPT fuel has no impact on the cladding's hydride reorientation.

The NRC staff reviewed the methodology for evaluating RIP, DNB propagation, and hydride reorientation for ADOPT fuel and determined that these evaluations are performed correctly, and the results are acceptable.

3.6.2.4 Clad Fatigue

The design basis for clad fatigue is that the fuel system will not be damaged due to fatigue. The acceptance limit for cladding fatigue is that the fatigue life usage factor is limited to less than 1.0 to prevent reaching the material fatigue limit, considering a safety factor of 2 on the stress amplitude or a safety factor of 20 on the number of cycles, whichever is more limiting (Reference 1). Fatigue is the accumulated effects of cycle strains associated with daily load follow and normal shutdown and return to full power. Due to the reduced densification of the ADOPT fuel, its cladding gap is closed earlier, which results in additional cyclic loading. The amplitude of such cyclic stresses is not expected to be significantly different from standard UO_2 fuel.

The NRC staff reviewed the fatigue analysis for ADOPT fuel and determined that the increased cladding fatigue for ADOPT fuel is acceptable since the addition is within the UO_2 limits for clad fatigue.

3.6.2.5 Fuel Rod Axial Growth

The acceptance limit for fuel rod axial growth is that the 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 (Section 6.1.2.5 of the TR (Ref. 1)). The PAD5 UB fuel rod axial growth models are used in the calculation of the fuel rod shoulder gap as a function of fast neutron fluence.

Section 3.5.2 of this SE describes Westinghouse's fuel rod growth measurements on both BWR and PWR ADOPT fuel rods. An important impact of ADOPT fuel's reduced in-reactor densification is an earlier closure of the fuel pellet-to-cladding gap. After gap closure, irradiation-induced fuel swelling will influence fuel rod growth. Section 5.2.2 of Ref. 1 demonstrates that the earlier pellet-cladding contact for rods of ADOPT fuel results in increased axial growth. Fuel axial growth occurs during early life due to the reduced in-pile densification. 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.

The NRC staff determined that the licensing criteria for ADOPT fuel axial growth is acceptable as per Section 3.5.2 of this SE.

3.6.2.6 Cladding Free Standing, Flattening and Densification

The acceptance limit for cladding free standing is that the cladding is short-term free standing at beginning of life, at power, and during hot hydrostatic testing. However, clad free standing does not depend on pellet properties or models, and is therefore not analyzed by the NRC staff in this SE. The acceptance limit for cladding flattening is that the fuel rod design shall preclude clad flattening during projected exposure. The fuel fabricated by Westinghouse is sufficiently stable with respect to the fuel densification and as such axial shrinkage is too small to allow clad flattening to occur. Westinghouse's 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. In Section 5.2.1 of the TR, Westinghouse states that during the manufacturing process, the pellets are checked to ensure they are compliant with the material specification. A re-sintering test was performed for 24 hours at 1700°C to check the thermal stability, a measurement of the pellet's expected densification behavior during irradiation.

Westinghouse performed a manufacturing analysis on all ADOPT pellets manufactured at the Westinghouse fuel facility over a two-year period, totaling **[ ]** ADOPT pellets tested. The normally and non-normally distributed data obtained from the manufacturing analysis was analyzed using methods specified in NRC RG 1.126 (Ref. 46). The results from this analysis show **[**
]

The NRC staff reviewed the controlling processes of fuel manufacturing and found that there are no axial gaps large enough to allow clad flattening and that ADOPT fuel is acceptable with respect to the clad fattening. The NRC staff also found that the advantage of ADOPT fuel in densification has an impact on the allowable plastic strain criteria of cladding. The NRC staff determined that the reduction in densification due to the ADOPT fuel material achieving higher density during sintering processes gives ADOPT fuel a clear advantage over undoped UO_2 fuel.

3.6.2.7 Fuel Pellet Overheating (Power-to-Melt)

The acceptance limit for fuel pellet overheating is that the fuel rod centerline temperature will not exceed the fuel melt temperature during Condition I and II operation, accounting for degradation of the melt temperature due to burnup and the addition of integral burnable absorbers. Section 3.2.3 of the TR and Section 3.3.3 of this SE concluded that there is no difference in the melting point of standard UO_2 fuel and ADOPT fuel. Figure 6-4 of Reference 1 shows that the fuel centerline temperature for ADOPT fuel is slightly lower relative to standard UO_2 fuel.

The NRC staff concluded that, since the design limit for the centerline melt for ADOPT fuel is the same and calculated centerline temperatures are lower, the power-to-melt limit for ADOPT fuel is as conservative as standard UO_2 fuel.

3.6.2.8 Pellet-Clad Interaction

The NRC SRP does not recommend a specific design criterion for PCI or PCMI. Rather, two existing design limits, one percent transient clad strain and no fuel centerline melt, should be satisfied to provide protection against PCI or PCMI fuel failure. PCI addresses stress-corrosion cracking mechanisms due to fission product embrittlement of the cladding, while PCMI is a stress driven failure mechanism. The one percent uniform clad strain criterion limits the clad strain during a transient to a range where the cladding has sufficient ductility to preclude strain related fuel failures. The fuel pellet overheating criterion precludes fuel melting and the associated large volume increase in the fuel due to the phase change that results in excessive cladding stresses and strain.

The NRC staff concluded that since the ADOPT fuel design meets the design limits, no additional PCI calculations are required.

The NRC staff reviewed all the fuel design criteria as identified in WCAP-18482-P/WCAP-18482-NP, Revision 0, and determined that the FRD design limits and the upper and lower bounds calculated with all relevant uncertainties are applied to the ADOPT fuel design. The NRC staff found FRD design limits, and the upper and lower bounds based on all relevant uncertainties acceptable.

3.6.2.9 Pellet-to-Cladding Interaction Stress-Corrosion Cracking Plant Maneuvering Guidelines

While no analytical acceptance criterion exists for PCI stress-corrosion cracking (PCI/SCC) cladding failure, licensees develop barriers to prevent cladding failure under normal operations. One such barrier to PCI/SCC cladding failure during normal operations is plant maneuvering and fuel pre-conditioning guidelines.

Section 3.3.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes fuel creep and testing done on ADOPT fuel to characterize performance. It concludes:

> 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 response to the RAI 12 regarding PCI/SCC plant maneuvering guidance (Ref. 2), Westinghouse stated that based on the data presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, it is **[**

] Westinghouse goes on to state that **[**

]

Based on the information presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, and in response to the RAI 12, the NRC staff finds the proposed strategy being employed to avoid PCI/SCC cladding failure acceptable.

3.6.3 Safety Analyses

3.6.3.1 FULL SPECTRUM™ Loss-of-Coolant Accident Methodology for ADOPT Fuel

Design-basis LOCA analyses is performed to demonstrate that the emergency core cooling system (ECCS) meets requirement of 10 CFR 50.46. ADOPT fuel design does not affect the overall goal of the LOCA analysis. However, it introduces potentially different physical effects which can change the results. This section describes how Westinghouse FULL SPECTRUM LOCA (FSLOCA™) methodology as described in WCAP-16996-P-A (Ref. 32) and the NOTRUMP evaluation model described in WCAP-10054-P-A (Ref. 33) and WCAP-10079-P-A (Ref. 34) are applied to a core with ADOPT fuel. The FSLOCA best-estimate EM is applicable to all PWR fuel designs with Zirconium alloy cladding for a full spectrum of pipe breaks for LOCA analysis. A modified version of the LOCTA-IV code was approved for use in the NOTRUMP EM to calculate the peak cladding temperature in the core during a small break LOCA transient.

In FSLOCA methodology (Ref. 32), a Phenomena Identification and Ranking Table assesses relative importance of various phenomena for small-break LOCA (SBLOCA) and large-break LOCA (LBLOCA). WCAP-16996-P-A discusses fuel-related phenomena that could be affected by ADOPT fuel and are described below:

Stored Energy

For small breaks, the core remains covered during the early periods of the transient, and reactor trip occurs early and the temperature difference between the fuel centerline temperature and the coolant is small. This removes much of the stored energy of the fuel. **[**

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Decay Heat

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Clad Deformation-Burst Strain, Relocation

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The NRC staff reviewed the physical effects of FSLOCA methodology and NOTRUMP EM on ADOPT fuel and determined that the introduction of ADOPT fuel will not affect the overall goal of the LOCA analyses.

The NRC staff determined that application of FSLOCA methodology and NOTRUMP EM on ADOPT fuel is acceptable.

<u>3.6.3.2 FULL SPECTRUM Evaluation Model Thermal Properties</u>

This section describes the aspects of the FSLOCA evaluation model (EM) (Ref. 32) that could be affected by the ADOPT fuel pellet.

Density

The chromia and alumina additives adjust the TD of ADOPT fuel downward from approximately 10.96 g/cm^3 to **[** **]** of the TD of UO_2. The TD of 684.86 lbm/ft3 as

assumed in the FSLOCA EM therefore remains applicable for ADOPT fuel pellets. This value of increased fraction of TD is modeled through user input to FSLOCA analysis.

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 Reference 32, the modified NFI model represents the thermal conductivity for as-fabricated density of 95 percent of TD, and an adjustment is made to account for as-fabricated fractions other than 95 percent. As mentioned in Section 2.0 of the SE, ADOPT fuel pellets have a **[**

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Specific Heat

No change is necessary to the models used for standard UO_2 fuel specific heat when modeling ADOPT fuel pellets.

Thermal Expansion

There is negligible difference in thermal expansion between standard UO_2 fuel and ADOPT fuel pellets. As such, the model described in Section 8.4.1 of Reference 32 remains applicable for ADOPT fuel pellets.

Thermal Conductivity of Relocated Fuel

Section 8.6.2 of WCAP-16996-P-A (Ref. 32) describes the model used to represent relocated fuel (fuel fragments axially relocated within the location of a rupture). **[**

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In section 3.3.2 of this SE, the NRC staff concluded that the models and uncertainty ranges for the thermal conductivity of relocated fuel remain applicable for ADOPT pellets.

3.6.3.3 NOTRUMP EM

This section addresses the impact of the ADOPT fuel pellets on the NOTRUMP EM as described in References 33 and 34, including the impact on the NOTRUMP EM version of the LOCTA-IV code used to calculate the peak cladding temperature in the core during a SBLOCA transient. The models and correlations used in the NOTRUMP EM **[**
] as discussed in the following subsections.

Density

The room temperature TD of standard UO_2 fuel is assumed to be 10.96 g/cm^3 and is adjusted to account for the user input percent of TD. Section 3.5.2 of this SE states that the chromia and alumina additives adjust the TD of ADOPT fuel downward from approximately 10.96 g/cm^3 to **[** **]** of the TD of UO_2. Therefore, the TD of 684 pounds per cubic feet (lbm/ft3) assumed in the NOTRUMP EM remains applicable for ADOPT fuel pellets. The increased percent of TD is modeled through adjustment to the user input for NOTRUMP LOCA analysis.

Thermal Conductivity

Section 3.3.2 of this SE indicates that the standard Westinghouse methodology for standard UO_2 fuel can be used to calculate the thermal conductivity for ADOPT fuel. The modified 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 PAD5 fuel performance code.

Specific Heat

No change is necessary to the models used for standard UO_2 fuel specific heat when modeling ADOPT fuel pellets.

Thermal Expansion

Appendix T of References 34 describes the thermal expansion of the fuel pellet in the NOTRUMP EM. There is a slight difference in thermal expansion between standard UO_2 fuel and ADOPT fuel. In view of this, the model described in Appendix T of Reference 33 is sufficient for thermal expansion of ADOPT fuel.

3.6.3.4 Radiological Consequence Analyses

Section 6.2.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes potential impacts of ADOPT fuel pellets on radiological consequence analyses and the applicability of established guidance in RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” (Ref. 47) and RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors” (Ref. 48). In addition, Westinghouse provided supplemental information in response to an RAI (RAI 13, Ref. 2). With respect to fuel rod performance and predicting the number of failed fuel rods under accident conditions, changes in material properties and performance characteristics between standard UO_2 fuel and ADOPT fuel have been identified and are accounted for, as necessary, in safety analysis models and methods.

With respect to the Maximum Hypothetical Accident (MHA) (a.k.a. LOCA) radionuclide release fractions, timing of releases, and elemental composition of releases, Westinghouse states that the release fractions are based on accident scenarios involving significant core melt, not impacted by ADOPT fuel pellets, and independent of ECCS performance demonstrations (i.e., 10 CFR 50.46 LOCA evaluations). **[**

] which could impact the timing of releases, are not significantly impacted. In response to the RAI 13, Westinghouse provided information as to why the addition of dopants does not significantly change the chemistry of the fission products released during a core melt accident. An independent NRC assessment of the impacts of large grain doped UO_2 fuel pellets on the MHA-LOCA source term also concluded that any potential impacts were insignificant (Ref. 28).

With respect to steady-state radionuclide release fractions (i.e., Table 3 gap fractions in RG 1.183 (Ref. 47)) used in the non-LOCA dose assessments, Westinghouse stated that given there is **[** **]**

Based on the information presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, and in response to the RAI 13 (Ref. 2), the NRC staff found the **[**

] for radiological consequence analyses acceptable.

<u>3.6.4 Non-LOCA Transient Analyses</u>

This section documents the non-LOCA transient analyses evaluation and the ADOPT fuel non-LOCA input to transient analysis.

Acceptance Criteria

Non-LOCA analyses are performed to satisfy the acceptance criteria for fuel rod failure and coolability. Westinghouse defines two categories of non-LOCA events that need to be considered due to the change in fuel pellet makeup: (1) events that are dependent on core-average effects, and (2) events analyzed to address local effects in the fuel rods.

For category 1, 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, overpressurization of the secondary system, or overfilling of the pressurizer. **[**

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For category 2 analysis to address local effects, the analyses are performed in two steps: 1) predictions of average core response to an initiating event, and 2) such local effects as fuel enthalpy, minimum DNB Ratio (DNBR), fuel melting, and peak cladding temperature (PCT). **[**

] Section 6.4 of the TR indicates that ADOPT fuel pellets do not affect the fuel cladding DNB performance as determined from DNB experiments or its method of determining the DNBR. Section 3.2.3 of the TR concludes that there is no appreciable difference in the melting point of standard UO_2 fuel and ADOPT fuel pellets, and the fuel centerline temperature for ADOPT fuel is slightly lower relative to standard UO_2 fuel. Since the ADOPT fuel pellets do not impact the properties of the fuel rod cladding, there is no impact on the PCT limits.

The NRC staff, upon review of the acceptance criteria, has determined that the existing non-LOCA acceptance criteria remain applicable to ADOPT fuel design.

In summary, 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.

3.6.5 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 changes in the mass and energy (M&E) released to containment due to a pipe rupture accident because the containment integrity analyses themselves do not model the fuel. Containment integrity analysis considers M&E released to containment from LOCA or a steam line break (SLB) event.

LOCA M&E release can be short term or long term. The short-term LOCA 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. Since the parameters that influence short-term M&E releases are break location, temperature of the fluid in the broken pipe, size of the break, and initial reactor coolant pressure, the fuel product and its performance do not influence the short-term M&E. The NRC staff concluded that any change in the fuel materials would not impact the short-term LOCA M&E releases.

For long-term LOCA M&E release calculations, Westinghouse has three licensed methodologies used for containment integrity, maximum sump temperature, and equipment qualification for Westinghouse and CE designs. The licensed/approved methodologies are:

- WCAP-10325-P-A (Ref. 35)
- WCAP-17721-P-A (Ref. 36)
- CENPD-132D (Ref. 37, 38)

The NRC staff reviewed the methodologies used for short-term and long-term LOCA M&E releases and determined that no methodological changes will be required for a full core ADOPT fuel design.

3.6.6 Short-Term and Long-Term Steam Line 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 sub-compartments inside or outside the containment building resulting from postulated secondary-side pipe ruptures. The transients are performed (typically 1 to 10 seconds 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. Since these transients are of short duration, they are influenced only 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. This means that any change in fuel pellet materials have no impact on the short-term SLB M&E releases.

Long-term SLB M&E release analyses use methods and models similar to those for non-LOCA analyses as described in Section 3.6.4 of the SE and remain valid for ADOPT fuel pellet design which is characterized by increased density and enlarged grain size. For the long-term SLB M&E analyses, there are three NRC-approved methodologies:

- LOFTRAN (Refs. 39, 40)
- RETRAN (Ref. 41)
- SGNIII (Ref. 42)

In summary, the NRC staff recognizes the computer codes, methods, and methodologies used for LOCA and SLB M&E releases for containment integrity analysis have been identified. These methodologies and codes were previously approved by the NRC staff. Therefore, the NRC staff determined that the above-mentioned containment integrity analyses methodologies are valid for ADOPT fuel design and are acceptable.

3.6.7 Nuclear Design Requirements

The ADOPT fuel characteristics of density, doping materials, and fuel temperature are inputs into the nuclear design methodology based on previously NRC-approved TRs assessing neutronics and nuclear design. The concentration of doping material is sufficiently low so that the doping materials have minimal impact on core reactivity due to its relatively low absorption cross sections. Since fuel temperature of ADOPT fuel is comparable to standard UO_2 fuel, fuel temperature changes will have minimal impact on neutronic behavior of ADOPT fuel. The pertinent ADOPT fuel characteristic which benefits nuclear design is the higher nominal density of **[   ]** in comparison to the current nominal density of **[   ]** The higher density of ADOPT fuel can potentially reduce up to four assemblies per cycle due to increased fissile material.

The NRC staff concluded that ADOPT fuel will be explicitly modeled using currently approved Westinghouse nuclear design methods and finds this approach acceptable.

3.6.8 Thermal-Hydraulic Design Methods

Westinghouse states that implementation for ADOPT fuel does not require modification or update to any previously NRC-approved methods and TRs for DNB and thermal-hydraulic analyses. The thermal-hydraulic methods applied to PWR DNB consists of a DNB correlation such as WRB-1 (Ref. 43), WRB-2 M (Ref. 16) and WSSV (Ref. 17), and WNG-1 (Ref. 18), Thermal-hydraulic subchannel code, VIPRE-W (Ref. 19), and a statistical method for determination of a 95/95 DNBR limit, such as the Revised Thermal Design Procedure (Ref. 44) and the Westinghouse Thermal Design Procedure (Ref. 45).

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.

Based on a review the methodologies, the NRC staff determined that the existing Westinghouse thermal-hydraulic design methods and codes and approved CHF correlations such as the ones listed above remain applicable to ADOPT fuel design thermal-hydraulic analyses and are acceptable.

3.6.9 Licensing Criteria Conclusions

The NRC staff concludes that due to the close similarities in performance between ADOPT fuel and standard UO_2 fuel, the existing Westinghouse's NRC-approved analytical methods and models for thermal-hydraulics, nuclear design, LOCAs, and non-LOCA transient analyses are appropriate with either minimal or no modifications for ADOPT fuel designs. The NRC staff determined that the acceptance criteria for safety analysis for standard UO_2 fuel are found appropriate for ADOPT fuel safety analyses and are acceptable

4.0 LIMITATIONS AND CONDITIONS

The NRC staff limits the applicability of the TR and associated methodology for fuel types, cladding, and reactors to the ranges listed below:

1. *Methodology*

- Licensees must demonstrate that the CRE analytical models, methods, and acceptance criteria are applicable to fuel designs containing ADOPT fuel pellets and capture all relevant fuel burnup and cladding corrosion related phenomena (Section 3.5.5 of this SE).

2. *Reactor and Cladding Types*

- ADOPT fuel must be used with the NRC-approved Westinghouse and CE PWR designs.
- ADOPT fuel must be used with the NRC-approved Westinghouse and CE fuel designs with corresponding pellet and assembly dimensions.
- ADOPT fuel shall be used with the NRC-approved zirconium based cladding materials, such as ZIRLO® and Optimized ZIRLO™.

3. *Fuel Limitations*

- ADOPT fuel may be used with or without annular pellets and application of ZrB2 integral fuel burnable absorber (IFBA) coating but must be used consistent with the defined IFBA parameters in applicable NRC-approved fuel performance or product TRs.
- Fuel burnup shall be limited to 62 GWd/MTU peak rod average for all cladding types.
- Nominal pellet density range will be **[ ]**
- Fuel grain size range will be **[ ]** as measured according to ASTM E112 as linear intercept without correction factor, which corresponds to **[ ]** with correction.
- Cr range from **[ ]** which corresponds to inclusion of Cr_2O_3 ranging from **[ ]**
- Al ranging from **[ ]** which corresponds to inclusion of Al_2O_3 ranging from **[ ]**

5.0 CONCLUSIONS

The NRC staff has reviewed the Westinghouse's ADOPT fuel TR, WCAP-18482-P/WCAP-18482-NP, Revision 0, for direct replacement for standard UO_2 fuel. ADOPT fuel is a modified UO_2 pellet doped with small amounts of chromia and alumina that results in higher density and enlarged grain size compared to undoped UO_2 fuel. The NRC staff's review of the TR has identified and confirmed the ADOPT fuel design has **[**

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The NRC staff's extensive review of the TR consisted of a prolonged virtual audit of supporting documents, requests for additional information, and review of the responses to RAIs. The review consisted of **[**

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The NRC staff completed its review of Westinghouse TR titled WCAP-18482-P/ WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel," and found that WCAP-18482-P/WCAP-18482-NP, Revision 0, is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the TR and Section 4.0 of the NRC staff's SE.

6.0 REFERENCES

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2. Letter from Z. Harper, Westinghouse, to NRC DCD, "Submittal of 'Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 for WCAP-18482-P/WCAP-18482-NP, Revision 0 (Proprietary/Non-Proprietary)'," LTR-NRC-21-9, March 19, 2021 (ADAMS Accession No. ML21078A486).

3. Letter from A. Schoedel, Westinghouse, to NRC DCD, "Submittal of Westinghouse Response to RAIs 1, 9, 19, 11, and Supplemental Information for RAI 6 for WCAP-18482-P/WCAP-18482-NP, Revision 0 (Proprietary/Non-Proprietary)," LTR-NRC-21-20, June 20, 2021 (ADAMS Accession No. ML21172A295).

4. Letter from A. Schoedel, Westinghouse, to NRC DCD, "Submittal of Westinghouse Revised Responses to RAIs 7a, 11, and Supplemental Response to RAI 9 for WCAP-18482-P/WCAP-18482-NP," LTR-NRC-21-34, November 12, 2021 (ADAMS Accession No. ML21316A138).

5. Letter from K. Hosack, Westinghouse, to NRC, DCD, "Submittal of WCAP-17642-P-A, Revision 1, 'Westinghouse Performance Analysis and Design Model (PAD5)'," (ADAMS Accession No. ML17335A826).

6. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987.

7. NRC, NUREG-0800, Section 4.2, "Fuel System Design," March 2007 (ADAMS Accession No. ML070740002).

8. NRC, NUREG-0800, Section 4.3, "Nuclear Design," March 2007 (ADAMS Accession No. ML070740003).

9. NRC, NUREG-0800, Section 4.4, "Thermal and Hydraulic Design," March 2007 (ADAMS Accession No. ML070550060).

10. NRC, NUREG-0800, SRP Section 6.2.1, "Containment Functional Design," March 2007 (ADAMS Accession No. ML070220505).

11. NRC, NUREG-0800, SRP Chapter 15.0, "Transient and Accident Analyses."

12. Transmittal of Approved Versions (WCAP-9272-P-A/WCAP-9273-NP-A) of Topical, "Westinghouse Reload Safety Evaluation Methodology," July 31, 1985 (ADAMS Package Accession ML19269B536).

13. WCAP-12488-P-A, "Westinghouse Fuel Criteria Evaluation Process," October 31, 1994 (ADAMS Accession No. ML020430107 (Non-publicly available, Proprietary)).

14. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A (Proprietary) and WCAP-14342-A & CENPD-404-NP-A, Addendum 1-A (Non-Proprietary), “Optimized ZIRLO™,” July 2006, (ADAMS Accession No. (Non-publicly available, Proprietary)).

15. WCAP-10444-P-A, “Westinghouse Reference Core Report Vantage 5 Fuel Assembly”, September 1985 (ADAMS Accession No. ML080650257 (Non-publicly available, Proprietary)).

16. WCAP-15025-P-A, “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” April 1999 (ADAMS Accession No. ML081610106 (Non-publicly available, Proprietary)).

17. WCAP-16523-P-A, “Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes,” August 2007 (ADAMS Accession No. ML072570633 (Non-publicly available, Proprietary)).

18. Letter from J. A. Gresham, Westinghouse, to NRC DCD, “Submittal of Approved Version of WCAP-16766-P-A/WCAP-16766-NP-A, ‘Westinghouse Next Generation Correlation (WNG-1) for Predicting Critical Heat Flux in Rod Bundles with Split Vane Mixing Grids’ (TAC No. 7230) (Proprietary/Non-Proprietary),” LTR-NRC-2-8, March 2, 2010 (ADAMS Accession No. ML100850530).

19. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 30, 1999 (ADAMS Accession No. ML993160153 (Non-publicly available, Proprietary)).

20. WCAP-16500-P-A, Revision 0, “CE 16x16 Next Generation Fuel Core Reference Report,” August 31, 2007 (ADAMS Accession No. ML072500331 (Non-publicly available, Proprietary)).

21. J. Arborelius, K. Blackman, U. Hallstadius, et.al.,” Advanced Doped UO_2 Pellets in LWR Applications,” Journal of Nuclear Science and Technology, Vol. 43, No. 9, p. 967–976, 2006.

22. ASTM E831-86, “Standard Test Method for Linear Thermal Expansion of Solid Materials by Thermomechanical Analysis,” ASTM International.

23. J. Noirot, L. Desgranges, and J. Lamontagne, “Detailed characterizations of high burn-up structures in oxide fuels,” Journal of Nuclear Materials, Volume 372.

24. HWR-872, OECD Halden Reactor Report, “The high Initial Rating Test IFA-677-1: Final Report In-Pile Results,” April 2008.

25. Letter from D. Morey, NRC, to C. Zozula, Westinghouse, “Audit Report for the November 9, 10, 12, and 13, 2020, Regulatory Audit for Westinghouse Electric Company TR WCAP-18482-P/WCAP-18482-NP, Revision 0, ‘Westinghouse Advanced Doped Pellet Technology (ADOPT™)’,” February 9, 2021 (ADAMS Accession No. ML21047A466).

26. “Framatome, Inc. - Publication of ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods,” June 2018 (ADAMS Package Accession No. ML18171A107).

27. [ ]

28. NRC Memorandum, “Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment,” Letter from M. Case (NRC) to J. Donoghue (NRC), May 13, 2020 (ADAMS Accession No. ML20126G376).

29. Nuclear National Laboratory, “The characteristics of LWR fuel at high burnup and their relevance to AGR spent fuel,” NNL (10) 10930, Issue 2, June 2011.

30. Regulatory Guide 1.236, “Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents,” June 2020 (ADAMS Accession No. ML20055F490).

31. WCAP-8964-A, Addendum 1-A, Revision 1-A “Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)’ (Proprietary),” June 20, 2006 (ADAMS Accession No. ML061810388 (Non-publicly available, Proprietary)).

32. Letter from J. A. Gresham, Westinghouse, to NRC DCD, “Westinghouse - Submittal of WCAP-16996-P-A/WCAP-16996-NP-A, Volumes I, II, III and Appendices, Revision 1, ‘Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)’,” LTR-NRC-16-75, November 30, 2016 (ADAMS Accession No. ML16343A232).

33. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” August 19, 1985 (ADAMS Accession No. ML100050586 (Non-publicly available, Proprietary)).

34. WCAP-10079-P-A, “NOTRUMP A Nodal Transient Small Break and General Network Code,” August 19, 1985 (ADAMS Accession No. ML100060364 (Non-publicly available, Proprietary)).

35. WCAP-10325-P-A, “Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version,” May 1983 (Non-publicly available, Proprietary).

36. WCAP-17721-P-A, “Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology,” September 2015 (ADAMS Package Accession No. ML15272A050 (Non-publicly available, Proprietary)).

37. CENPD-132D, “Calculational Methods for the C-E Large Break LOCA Evaluation Model,” August 8, 1974 (ADAMS Accession No. ML20275A177 (Non-publicly available, Proprietary)).

38. CENPD-132D, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1974 (ADAMS Accession No. ML20275A187 (Non-publicly available, Proprietary)).

39. "Acceptance for Referencing of Licensing TR WCAP-8821-P/8859-NP, 'TRANFLO Steam Generator Code Description', and WCAP-8822-P/8860-NP, 'Mass and Energy Release Following a Steam Line Rupture'," August 22, 1983 (ADAMS Accession No. ML012340064 (Non-publicly available, Proprietary)).

40. WCAP-8822-S1-P-A, "Mass and Energy Releases Following a Steam Line Rupture, Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986 (ADAMS Accession No. 8610170195 ((Non-publicly available, Proprietary)).

41. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 30, 1999 (ADAMS Accession No. ML093421329 ((Non-publicly available, Proprietary)).

42. CE Standard Safety Analysis Report (CESSAR), Appendix 6B, "Description of the SGNIII Digital Computer Code Used in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis," January 1974 (Non-publicly available, Proprietary).

43. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 31, 1984 (ADAMS Accession No. ML080630433 (Non-publicly available, Proprietary)).

44. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (ADAMS Accession No. ML080630437 (Non-publicly available, Proprietary).

45. Letter from K. Hosack, Westinghouse, to NRC, "Submittal of WCAP-18240-P-A/ WCAP-18240-NP-A, 'Westinghouse Thermal Design Procedure (WTDP)'," LTR-NRC-20-29, April 2020 (ADAMS Accession No. ML20104C042).

46. Regulatory Guide 1.126, Revision 2, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," NRC, March 2010 (ADAMS Accession No. ML093360318).

47. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

48. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003 (ADAMS Accession No. ML031490640).

49. LA-UR-18-25549, M. Cooper, R. Stanek, D. R. Anderson, "Milestone Report: Calculate parameters controlling grain growth in doped UO_2 (M3MS-18LA0201035)," Los Alamos National Laboratory, June 25, 2020.

50. L. S. Tong and Joel Weisman, “Thermal Analysis of Pressurized Water Reactors,” Third Edition, American Nuclear Society, La Grange Park, Illinois USA, ISBN-13: 978-0894480386, 1996.

51. SCIPII, N13/168 Studsvik-SCIP II 163, Non-Destructive Examination of ramp rodlets (after Ramp), Studsvik Nuclear AB, April 2013.

52. N-11-008, Revision 1, “Destructive examinations on two ADOPT rods irradiated in Oskarshamn 3,” Studsvik Nuclear AB report N-11/008.

53. “Fission Gas Release in a Multiscale context: 10 Years of Research at SCK*CKN,” K. Govers, S. E. LLemehov, and M. Verwerft, 2011 Water Reactor Fuel Performance Meeting, Chengdu, China, 2011.

Attachment: Comment Resolution Table

Principal Contributors: Mathew Panicker, NRR/DSS/SFNB
Paul Clifford, NRR/DSS

Date: June 13, 2022

NUCLEAR REGULATORY COMMISSION RESOLUTION OF COMMENTS TABLE
Comments on the NRC Draft Safety Evaluation for Westinghouse
Topical Report WCAP-18482-P/WCAP-18482-NP, Revision 0,
"Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel"

The table is a record of Westinghouse comments received on the draft SE (ADAMS Package Accession No. ML22098A058) and the NRC staff's response to them. Comment page and line number refer only to the draft SE and will not correspond to the final SE as pages and line numbers have shifted.

Table: Resolution of comments

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
1	27-28	Editorial	Please use 'µg/gU' units in lieu of 'ppm' when referring to the alumina and chromia content for consistency with Limitations and Conditions	Comment not accepted. No changes made.
1	27-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
4	13-14	Editorial	Suggest removing the following wording due to redundancy "…, because approval for Westinghouse FCEP is limited to nuclear fuel in Westinghouse plants"	Comment not accepted. No changes made.
5	17-20	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
5	24 & 29	Proprietary Markings	Suggest removing proprietary marking, since it is not needed.	Comment acceptable. Proprietary markings removed.
5	26-32	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
5	35-36	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
5	38-43	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
7	44	Clarification	Please change "ASTM E831-19" to "ASTM E831-86," which was the standard used at the time of testing.	Comment acceptable. Editorial change made.
8	12	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
10	26	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
10	39-44	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
11	11-14	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
11	41-45	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
12	7-8	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
14	32-34	Proprietary Markings	For consistency, please mark proprietary as shown: "…staff reviewed **[** **]**"	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
15	1-3	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
16	1-5	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
18	12	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
18	18	Proprietary Markings	Please mark proprietary as shown: "…the **[ ]** peak rod…"	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE
18	45	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
19	2	Technical clarification	Please change units from "MWd/kgU" to "Wd/kgUO_2"	Comment not accepted. Text, including BU definition, MWd/kgU, consistent with TR (See Table 4-6).
19	18	Proprietary Markings	Please mark proprietary as shown: "…62 MWd/kgU **[ ]** as Westinghouse..."	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
20	4	Editorial	[ ] [ ]	Comment acceptable. Editorial change made.
20	4-18	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
20	42-44	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
21	10	Editorial	Please change 'including' to 'excluding'	Comment acceptable. Editorial change made.
21	40-42	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
22	9-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
22	19-25	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
22	28-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
22	30-31	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
22	35-40	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
23	9-14	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
23	27-29	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
24	1-4	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
24	7-15	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
24	19-22	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
26	7	Reference	Please use Reference 1 or 5.	Comment acceptable. Change made.
26	21	Editorial	Please replace “Section 5.2.2” with “Section 6.1.2.5”	Comment acceptable. Editorial change made.
27	6	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
27	8-10	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
28	25-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
28	37, 42	Editorial	Please change “Full Spectrum” to "FULL SPECTRUM"	Comment acceptable. Editorial change made.
28	47-48	Technical clarification	Suggested wording: “A modified version of the LOCTA-IV code was approved for use in the NOTRUMP EM to calculate the peak cladding temperature in the core during a small break LOCA transient.”	Comment acceptable. Change made.
29	10-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
29	14-20	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
29	15	Proprietary Markings	Suggest removing proprietary marking, since it is not needed.	Comment acceptable. Proprietary markings removed.
29	24-28	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
29	27	Proprietary Markings	Suggest removing proprietary marking, since it is not needed.	Comment acceptable. Proprietary markings removed.
29	32-39	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable. Proprietary markings removed.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
29	41-44	Technical Clarification	Suggested change: "The NRC staff reviewed the physical effects of FSLOCA methodology and NOTRUMP EM on ADOPT fuel and determined that the introduction of ADOPT fuel will not affect the overall goal of the LOCA analyses. The NRC staff determined that application of FSLOCA methodology and NOTRUMP EM on ADOPT fuel is acceptable."	Comment acceptable. Last paragraph of Section 3.6.3.1 of the SE is replaced with "The NRC staff reviewed the physical effects of FSLOCA methodology and NOTRUMP EM on ADOPT fuel and determined that the introduction of ADOPT fuel will not affect the overall goal of the LOCA analyses. The NRC staff determined that application of FSLOCA methodology and NOTRUMP EM on ADOPT fuel is acceptable."

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
30	7	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
30	18-30	Proprietary Markings	Please mark proprietary as shown: [ ]	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
30	19	Editorial/ Proprietary Markings	Please change 'WCOBRA/TRACPF2' to 'WCOBRA/TRAC-TF2' and delete proprietary bracket	Comment acceptable. Editorial change made. Proprietary marking removed.
30	46-48	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
31	1-6	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
31	3	Editorial	[ ]	Comment acceptable. Editorial change made.
31	13-15	Technical Clarification	Suggested wording: “This section addresses the impact of the ADOPT fuel pellets on the NOTRUMP EM as described in References 33 and 34, including the impact on the NOTRUMP EM version of the LOCTA-IV code used to calculate the peak cladding temperature in the core during a SBLOCA transient.”	Comment acceptable. First sentence in Section 3.6.3.3 of the SE is replaced with “This section addresses the impact of the ADOPT fuel pellets on the NOTRUMP EM as described in References 33 and 34, including the impact on the NOTRUMP EM version of the LOCTA-IV code used to calculate the peak cladding temperature in the core during a SBLOCA transient.”
31	16	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
31	21, 23	Proprietary Markings	Please remove proprietary brackets from the room temperature theoretical density of standard UO_2 fuel	Comment acceptable. Proprietary markings removed.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
31	24	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
31	31	Editorial	Please change section “3.3.4” to Section “3.3.2”	Comment acceptable. Editorial change made.
31	43	Editorial	Please change “Reference 33” to “Reference 34”	Comment acceptable. Editorial change made.
32	16-17	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
32	26	Proprietary Markings	Please mark proprietary as shown: “...there is **[** **]**”	Comment acceptable – marked as proprietary information in the proprietary version and redacted proprietary information in the non-proprietary version of the final SE.
32	26	Editorial	Suggest removing proprietary marking, since it is not needed.	Comment acceptable. Proprietary markings removed.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
32	29-30	Proprietary Markings	Please mark proprietary as shown: "…found the **[** **]** for…"	Comment acceptable – marked as proprietary information in the proprietary version and redact proprietary information in the non-proprietary version of the final SE.
32 33	46-47 1-2	Proprietary Markings	Please mark proprietary as shown: **[** **]**	Comment acceptable - marked as proprietary information in the proprietary version and redact proprietary information in the non-proprietary version of the final SE.
33	7-8	Proprietary Markings	Please mark proprietary as shown: **[** **]**	Comment acceptable – marked as proprietary information in the proprietary version and redact proprietary information in the non-proprietary version of the final SE.
34	44	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
35	2	Editorial	Suggest deleting reference 43 and leaving this statement generic to be consistent with Sections 2.2.4 and 6.3 of the TR.	Comment acceptable. Editorial change made.
35	9	Editorial	Please delete 'WNG' from 'WNGWSSV'	Comment acceptable. Editorial change made.
35	20	Editorial	Suggested wording: "…approved CHF correlations such as the ones listed above…"	Comment acceptable. Editorial change made.
36	6	Technical Clarification	Please add IFBA and annular blanket usage language for consistency with the TR. Suggested wording: "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."	Comment acceptable. Change made – added to Sub-section 3, "Fuel Limitations," in Section 4.0, "Limitations and Conditions."
36	7	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.

Table: Resolution of comments (Continued)

Draft SE Page No.	Line No.	Comment Type	Westinghouse Suggested Revision	NRC Resolution
36	8	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
36	9	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
36	9	Technical clarification	Please add "with correction" to the end of the sentence, for consistency with the TR.	Comment acceptable. Editorial change made.
36	10-11	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
36	11, 13	Technical clarification	Please use "μg/gU" units in lieu of 'ppm' when referring to the alumina and chromia content	Comment not acceptable. No change made.
36	12-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
36	21-22	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
36	26-31	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	Comment acceptable.
39 40	43 3, 15	Editorial	Suggest adding "(Nonpublicly available, Proprietary)." to end of Reference 35, 36, and 39.	Comment acceptable. Editorial change made.

Section B

List of Changes

The following editorial changes have been made to the initially submitted topical report.

List of Changes

Change Number	Affected Report Section	Comments
1	Section 4.5 Pages 4-13 - 4-15	The burnup units in the Halden IFA-677 test discussion are MWd/kgUO_2. The original issuance of WCAP-18482-P/NP listed them as MWd/kgU.

Section C

Submittal of Topical Report

Westinghouse

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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)
Jason Drake (NRC)

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

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 <u>A</u>dvanced <u>Do</u>ped <u>P</u>ellet <u>T</u>echnology (**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."
2. U.S. NRC. NUREG-0800. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (Formerly Issued as NUREG-75/087).
3. WCAP-9272-P-A. "Westinghouse Reload Safety Evaluation Methodology." July 1985.
4. WCAP-12488-A. "Westinghouse Fuel Criteria Evaluation Process." October 1994.
5. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A (Proprietary) and WCAP-14342-A & CENPD-404-NP-A, Addendum 1-A, (Non-Proprietary), "Optimized ZIRLO™," July 2006.
6. WCAP-17642-P-A. Rev. 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.
7. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.
8. WCAP-16500-P-A, "CE 16x16 Next Generation Fuel Core Reference Report," August 2007.

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).

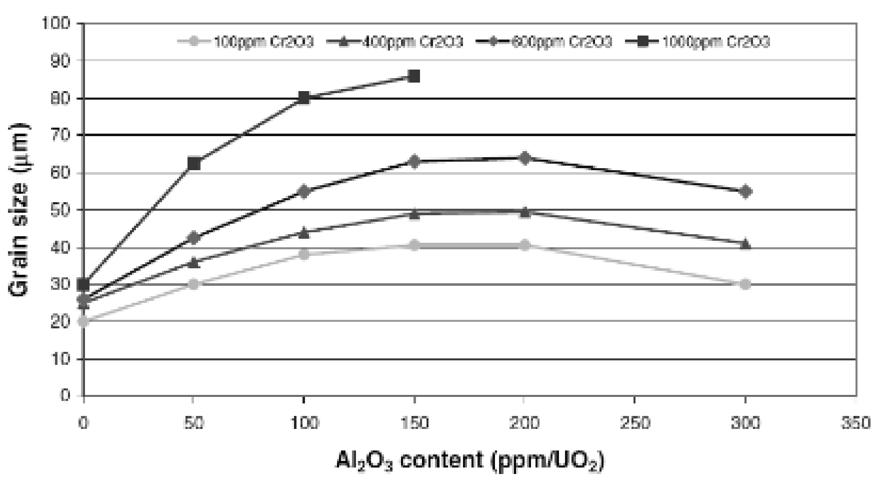


Figure 3-1 The variation of grain size with alumina and chromia content (Ref. 1)

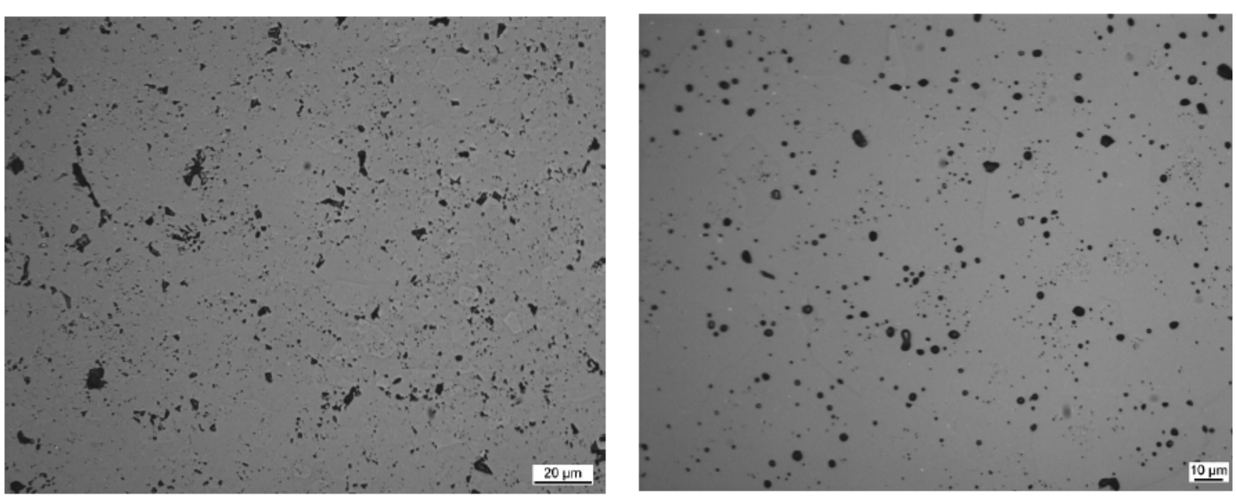
3.1.2 Microstructure

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).

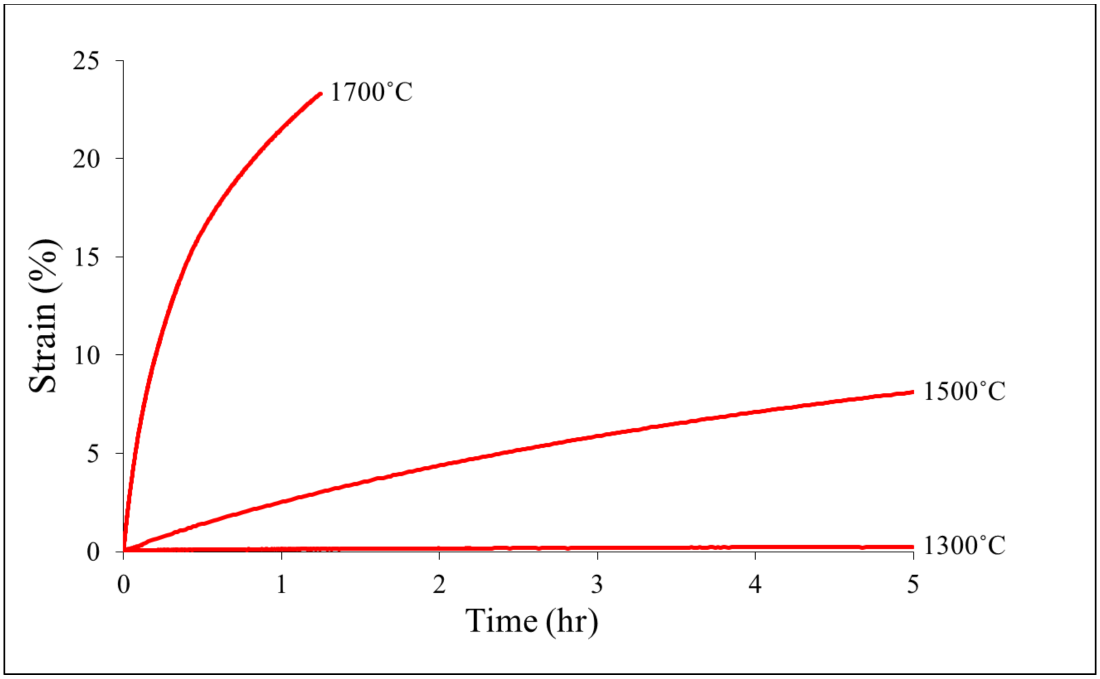


*Figure 3-11 Measured strain for **ADOPT** pellets under constant 60 MPa stress at 1300°C, 1500°C, and 1700°C*

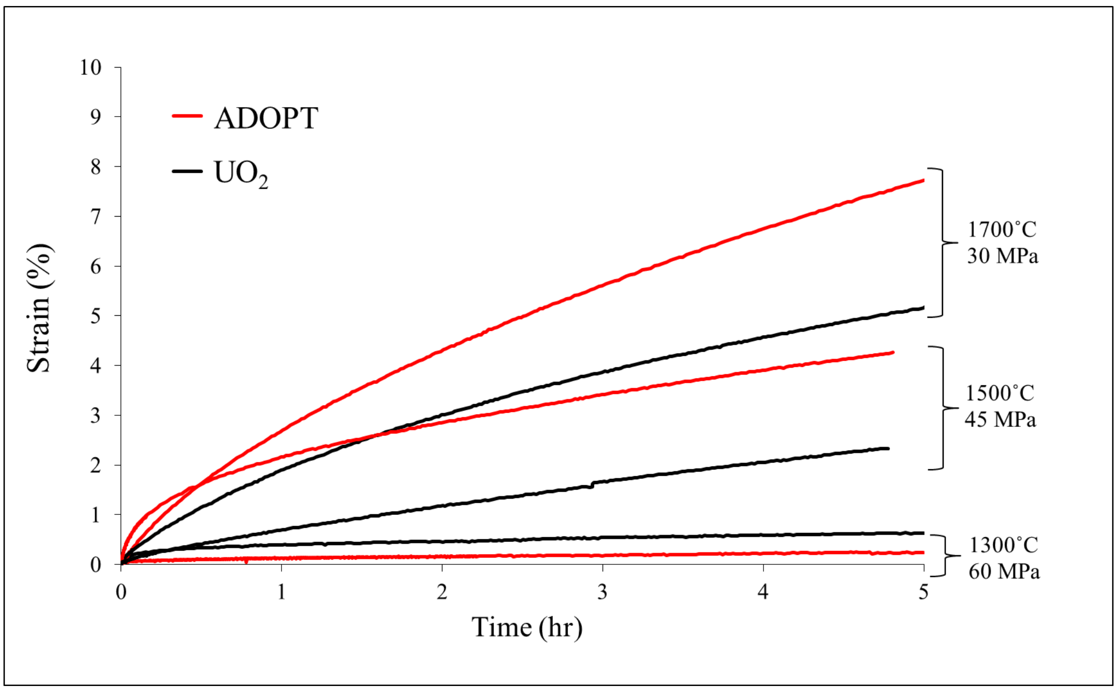


*Figure 3-12 Creep testing of **ADOPT** and reference UO_2 fuel under identical test conditions*

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).

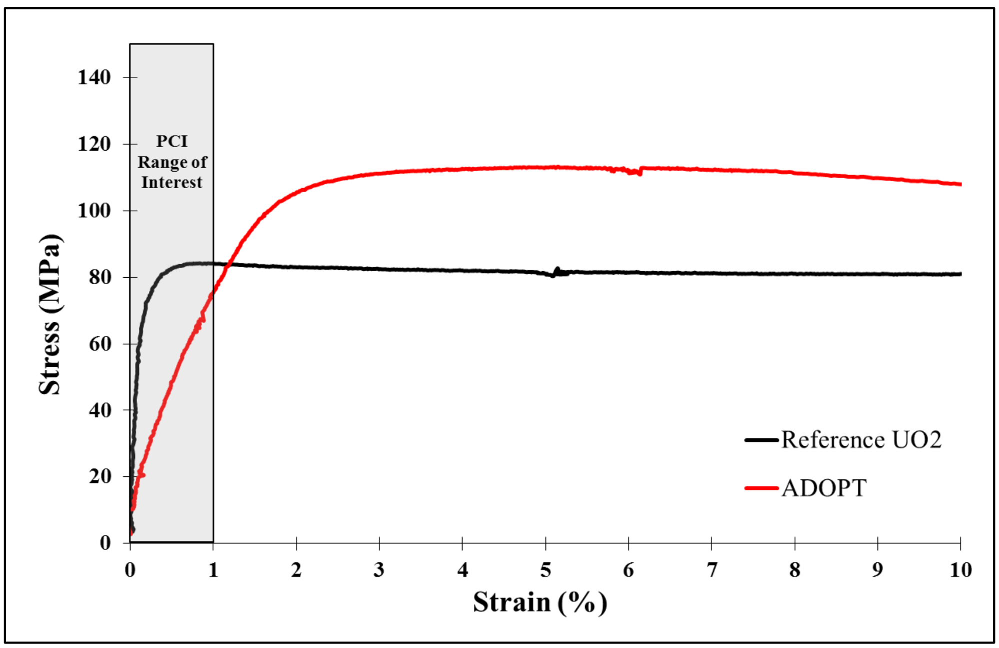


*Figure 3-13 Hardening test of **ADOPT** and reference UO_2 fuel at 1500°C (50%/h)*

3.4 CHAPTER 3 REFERENCES

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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	Nominal Density		Nominal 3D grain size	Volume change after irradiation at Barsebäck*
			[ppm]	[%]	[g/cm³]	[μm]	
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

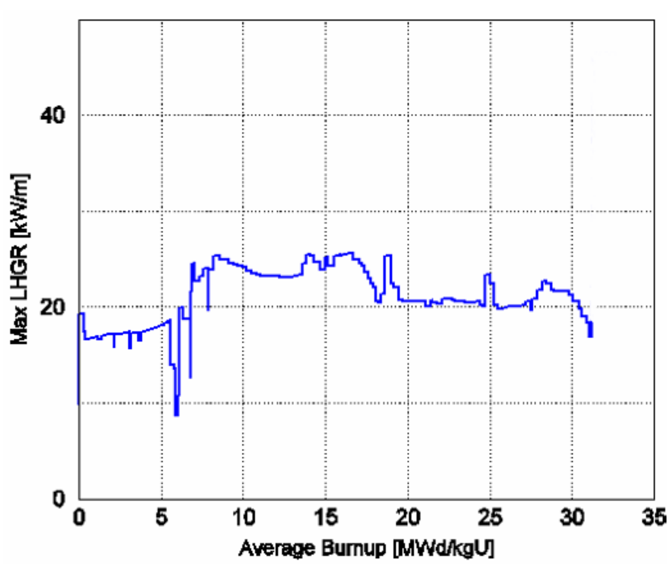


Figure 4-1 Power history of base irradiated segmented rods at Barsebäck 2 Reactor

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).

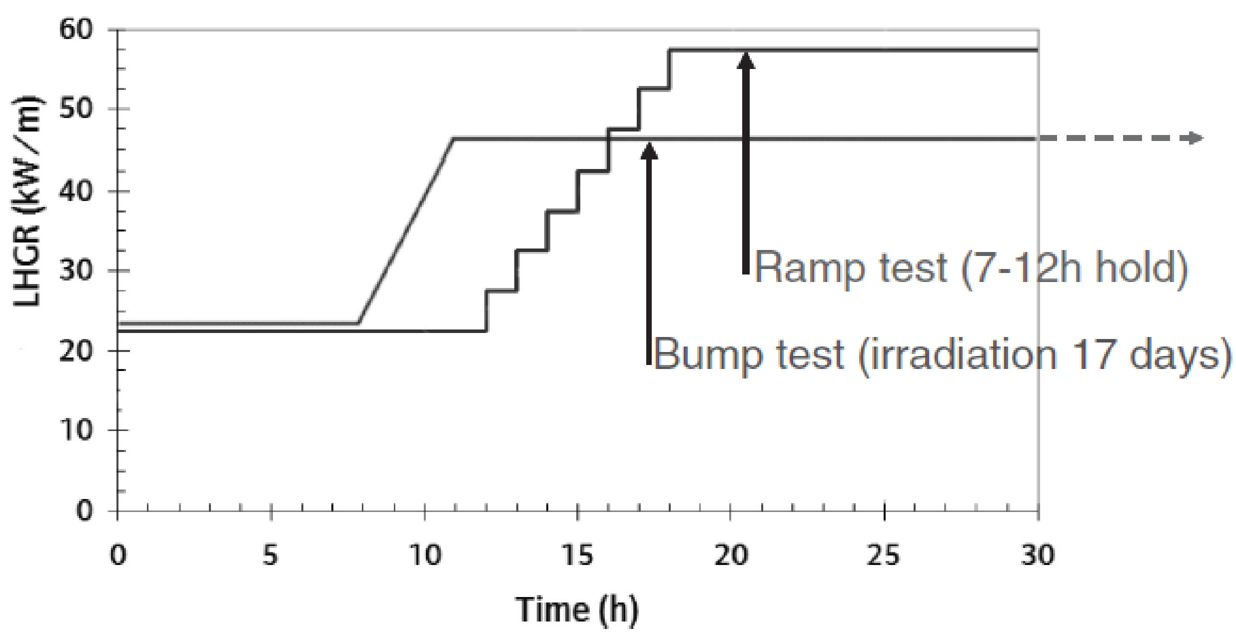
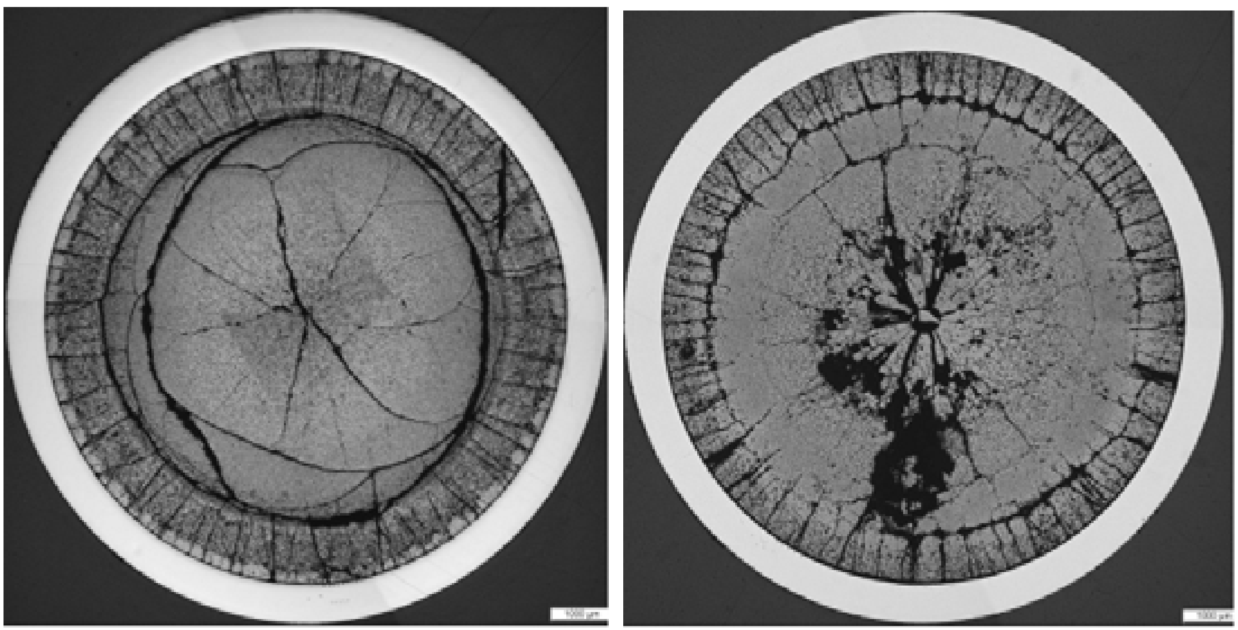


Figure 4-2 Power history of ramp and bump test segments at R2 reactor

4.1.3 Ramp and Bump Test PIE

Ceramography

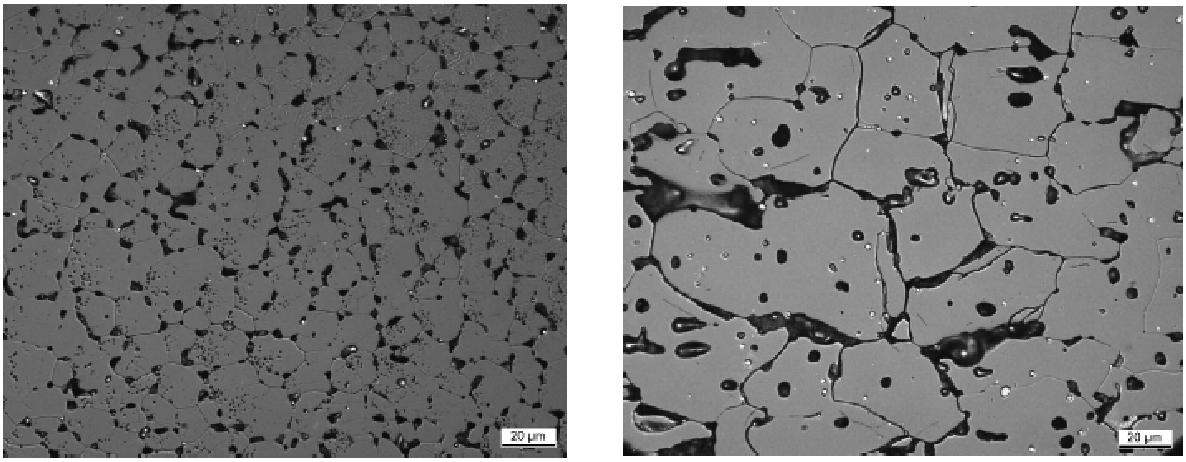
Ceramography of the ramp tested segment (D1) with 1000 ppm Cr_2O_3 additive revealed a central void, unlike the standard pellet, see Figure 4-3. The central void appeared only in the region of the segment which had been subjected to the highest powers. As demonstrated in Section 3.3.2, the additives 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

<u>Fission Gas Measurements</u>

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

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

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/kgUO_2. 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 $MWd/kgUO_2$ 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).

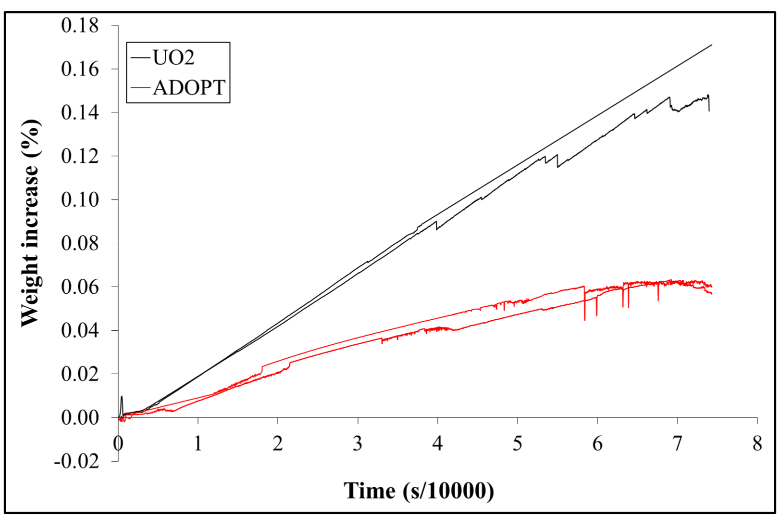

*Figure 5-1 Weight gain in 400°C steam of standard UO_2 and **ADOPT** fuel.*

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.

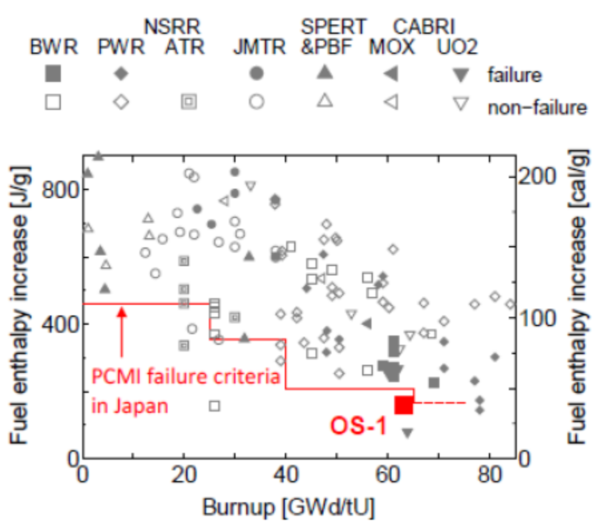

Figure 5-7 Existing RIA-simulated-experiment data and the current Japanese PCMI-failure criteria shown as a function of fuel burnup. (presented as Fig. 7 in Ref. 6)

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.

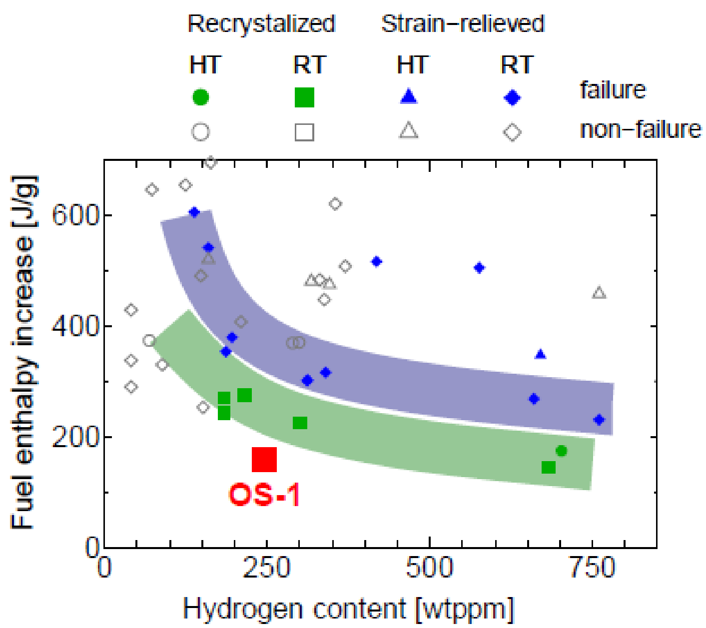

Figure 5-8 Existing RIA-simulated-experiment data plotted as a function of cladding hydrogen content. (presented as Fig. 8 in Ref. 6)

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.

<u>Densification Model</u>:

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.

<u>Thermal Conductivity Model</u>:

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.

Rod Growth Model

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.
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]^{a,c}

Decay Heat

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

]^{a,c}

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]a,c

Clad Deformation (Burst Strain, Relocation)

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]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). **[**

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]^{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.

Specific Heat

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

Thermal Expansion

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. **[**

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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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4. Letter from Westinghouse (J. Gresham) to NRC, LTR-NRC-16-5, "Response to the 120 day RAIs from RAI Set 2 for WCAP-17642, 'Westinghouse Performance Analysis and Design Model (PAD5)' (Proprietary/Non-Proprietary)," February 2016.
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19. CESSAR Appendix 6B, "Description of the SGNIII Digital Computer Code Used in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis," January 1974.
20. Regulatory Guide 1.195, Revision 0, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.
21. Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
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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:

[
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-
-
]^{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.

Section D

Submittal of Responses to Request for Additional Information #1

Westinghouse

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LTR-NRC-21-9

March 19, 2021

Subject: Submittal of "Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 for WCAP-18482-P/WCAP-18482-NP, Revision 0" (Proprietary/Non-Proprietary)

Enclosed is the first set of responses to the Nuclear Regulatory Commission's ("Commission's") request for additional information regarding Westinghouse Electric Company LLC ("Westinghouse") Topical Report WCAP-18482-P/NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel." Enclosure 4 to this letter provides a list of documents submitted to the docket as requested by the NRC to support its review of the subject document. All documents submitted as indicated in Enclosure 4 are considered proprietary in their entirety and non-proprietary versions have not been prepared as they would be of no value to the public.

This submittal contains proprietary information of Westinghouse. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the 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-21-5158 and should be addressed to Zachary S. Harper, Manager, Licensing Engineering, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Cranberry Township, PA 16066.

Zachary S. Harper, Manager
Licensing Engineering

cc: Ekaterina Lenning
Dennis Morey

Enclosures:
(1) Affidavit, AW-21-5158
(2) Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 on WCAP-18482-P/ WCAP-18482-NP (Proprietary)
(3) Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 on WCAP-18482-P/WCAP-18482-NP (Non-Proprietary)
(4) List of Documents Submitted to the Docket in Support of RAI Responses (Proprietary)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

COUNTY OF BUTLER:

(1) I, Zachary S. Harper, 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 the proprietary portions of LTR-NRC-21-9 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

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

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

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

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

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

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

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

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

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

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

(6) The attached documents in Enclosures 2 and 3 are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing

each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

The attached documents in Enclosure 4 and the documents listed within Enclosure 4 contain proprietary information throughout, for the reasons set forth in Sections (5)(a) through (f) of this Affidavit. Accordingly, a redacted version would be of no value to the public.

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: 3/19/2021

Zachary S. Harper, Manager
Licensing Engineering

Enclosure 3

Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 on WCAP-18482-P/WCAP-18482-NP

(Non-Proprietary)

(25 Pages)

March 2021

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

SFNB RAI 2

[

]^{b,c} This report shows that there is *dark zone* in the **ADOPT** fuel pellet and it is not centered in the pellet. Further, the TR states that the dark zone appearance is an effect of the large number of small pores. Grain boundaries cannot be detected inside the dark zone and grain size determination is not possible at various radii of the pellet. Also, the grain size was not possible to measure inside the dark zone, despite attempts with different etching times. Outside the dark zone the grain size is unchanged from the grain size as manufactured.

(a) Please provide details of the dark zone in **ADOPT** fuel pellet.

(b) What is the impact on the overall grain size measurement for **ADOPT** fuel due to the presence of the dark zone?

Response to SFNB RAI 2(a)

[

]^{b,c}

The 'dark zone,' which refers to the shaded concentric region of the pellet, originates after chemical etching in preparation for ceramography. Dark zones are the result of the etchant interacting with the intragranular pores and bubbles in the central region of the fuel, where higher temperatures are experienced. It is a common feature of UO_2 fuel irradiated to higher burnups and is not a property exclusive to the **ADOPT** pellet.

The etchant used interacts with grains differently depending on the crystal orientation. The H_2SO_4 acts as an acid and the H_2O_2 primarily as an oxidant, so some oxidation of the surface may also occur. Any inhomogeneities in the fuel will be attacked slightly preferentially. So, grain boundaries (GB) are preferentially attacked because of the edges and will appear dark because of the two crystal orientations (from each side of the GB) that are being etched creating a small ridge between the grains. Additionally, the freshness of the fuel sample will result in differences in the etching procedure, and also cause a slightly different result. Therefore, comparing the etching of two different samples is difficult.

The off-center dark zone, as depicted in Figure 1, indicates a power gradient across the fuel pin that is shifted, and it is an indicator that the moderation is stronger on that side of the fuel pin. Due to the etchant interacting with the intragranular pores and bubbles in the central region, grain boundaries cannot be distinguished in the dark zone, Figure 2.

[ ][b,c] is presented in Figure 3 and Figure 4. Similar to the **ADOPT** rod, a dark zone is present, and the grain boundaries are difficult to distinguish due to the formation of the dark zone.

Response to SFNB RAI 2(b)

As discussed above, the grain size measurement was indeterminate due to the high amount of intragranular bubbles in this region. For comparison, ceramography of an alternative **ADOPT** rod, [ ][b,c] is presented in Figure 5 and Figure 6. After etching, a dark zone is visible and individual grains are difficult to distinguish. The grain size measurements are presented in the Table 1.

[ ][b,c]

[ ][b,c]

b,c

b,c

b,c

b,c

b,c

SFNB RAI 3

Section 6.2.1.1.1 states: **[**

]a,c Describe the impact on loss-of-coolant accident (LOCA) analysis due to the use of modified NFI model **[**
]a,c Further justification is needed to provide clarification to the NRC staff.

Response to SFNB RAI 3

As described in Section 6.2.1.1.1 of WCAP-18482-P/WCAP-18482-NP, the pellet conductivity model used in the FULL SPECTRUM™ LOCA (FSLOCA™) evaluation model (EM) is used to determine the pellet conductivity at 95% theoretical density, and then the value is adjusted to account for densities other than 95%. Eq. 11-150 in WCAP-16996-P-A, Revision 1 shows the means by which the conductivity is adjusted for different theoretical density. It is effectively a linear scale factor which will result in a proportional change in adjusting from 95% to 97% or from 95% to **[** **]a,c**.

As further discussed in Section 6.2.1.1.1 of WCAP-18482-P/WCAP-18482-NP, the primary importance of pellet conductivity for the LOCA response is related to the pellet temperatures during normal operation. For that purpose, the initial, steady-state pellet temperatures used in the FSLOCA EM simulations **[**

]a,c

SFNB RAI 4

During the NRC regulatory audit for the Westinghouse TR WCAP-18482-P/WCAP-18482-NP, Revision 0, two (2) questions (topics) were asked regarding the distribution and re-distribution of the dopants in fresh fuel, as well as throughout irradiation. The intent of the following table is to resolve those questions, specifically those listed in the table below. Please provide the impact of distribution and re-distribution of dopants during irradiation on the behavior of **ADOPT** fuel during operation.

Topic Number	Subject	Response	Acceptable / Action Item	RAI Needed
4	Distribution of dopants in **ADOPT** fuel during irradiation	[ ]b,c		Combined with #6
6	Re-distribution of dopants during irradiation	[ ]b,c	Needs more	Yes

Response to SFNB RAI 4

Fresh fuel examination: The radial concentration of aluminum and chromium, in an unirradiated pellet, has been measured using wavelength dispersive spectrometry (WDS). The WDS examinations show a large statistical variation due to the low concentration of elements. [

]b,c More details are discussed in Section 3.1.3.1 of WCAP-18482-P/WCAP-18482-NP.

Steady-state, irradiated fuel examination: Electron probe microanalysis (EPMA) has been performed on two irradiated **ADOPT** rods to investigate the dopant migration behavior under steady state conditions.

- [

 ]b,c
- [

 ]b,c

[

]b,c

In addition to EPMA, laser ablation mass spectrometry was performed on the C1 **ADOPT** segment. This method offers an extremely high sensitivity to a wide range of elements. [

] b,c

Post-ramp, irradiated fuel examination: A section of the C1 **ADOPT** rod was ramp tested to a terminal power of 30 kW/m. Following ramp, laser ablation mass spectrometry, was performed on the segment. **[**

][b,c]

In summary, **[**

] [b,c]

b,c

Figure 7: *SEM/EPMA of rod B5. Radial distribution of Al and Cr.*

b,c

Figure 8: *SEM/EPMA of rod C1. Radial distribution of Al and Cr.*

b,c

Figure 9: *Radial profile using laser ablation for volatiles and additives in non-ramped* ***ADOPT*** *rod*

b,c

Figure 10: *Radial profile using laser ablation for volatiles and additives in ramped* ***ADOPT*** *rod*

SFNB RAI 5

Section 7.2.1.4.5 of PAD5 TR states: "Explicit calculation of the pellet radial power distribution (PD) is therefore only required for fuel designs not addressed within the range of the built-in pellet radial power distribution tables." What adjustment to PD is done to accommodate **ADOPT** fuel PD?"

Response to SFNB RAI 5

The PAD fuel pellet radial power distribution (RPD) model has been qualified for fuel enrichments between **[ ]**^{a,c} The low concentration of chromium oxide (Cr_2O_3) and aluminum oxide (Al_2O_3) doping materials have minimal impact on the core reactivity due to the small absorption cross section. Additionally, the increase in theoretical density (TD) of **ADOPT** fuel **[ ]**^{a,c} relative to standard uranium dioxide **[ ]**^{a,c} will not have an appreciable impact on the RPD. The RPD models for UO_2 fuel therefore remain applicable to **ADOPT** fuel. If the **ADOPT** fuel stays within the predefined RPD limits for PAD **[ ]**^{a,c} then no adjustments are required.

SFNB RAI 6

Based on item number 6 in the audit plan, the NRC staff requests Westinghouse to provide documentation with details of data analysis for irradiation programs described in Section 4; specifically ramp and bump testing, fission gas measurements, fuel pellet cracking, and cladding metallography. Chapter 5 of WCAP-18482-P/WCAP-18482-NP, Revision 0, characterizes the material properties and performance of ADOPT fuel pellets. Please confirm that the empirical data presented in this chapter was collected on fuel with the same specifications as the final production **ADOPT** fuel. Identify and disposition any differences, if they exist.

The following table shows the NRC staff concerns regarding Chapter 5 material properties and performance of **ADOPT** fuel during the audit. Please provide clarifications to the NRC staff concerns.

[ ]^{b,c}

Response to SFNB RAI 6

Irradiation Programs Analysis Documentation:

Westinghouse has docketed two documents detailing the fission gas release analysis of **ADOPT** fuel operated **[**

]^{b,c}

Characterization and details of data analysis of **ADOPT** ramp testing will be provided as part of RAI 11 in a subsequent submittal.

<u>**ADOPT** Fuel Specification Data:</u>

In the following two figures, the manufacturing inspection data for production runs of **ADOPT** fuel since 2009 is presented. The dashed lines indicate the manufacturing specification requirements, which are also reflected in the fuel design constraints of the topical report:

[ ]b,c

For comparison, all manufacturing information of the data contained within the topical report is presented.

In Figure 11, a comparison of the density and grain size is presented; all experimental data presented in the topical report falls within the fuel design constraints requested for density and grain size.

In Figure 12, a comparison of the Al and Cr content is presented; all experimental data, with exception of the B2/R2 33004 segment, falls within the fuel design constraints requested for Al and Cr content. The B2/R2 data is presented in the topical because it is similar in grain size, density, and total additive content to **ADOPT**; similar behavior is expected.

Note that the KTH and IFA-677 Cr and Al content are presented as-manufactured (prior to sintering). Aluminum and chromium content were not measured after sintering for these samples. **[**

][b,c]

b,c

Figure 11: ADOPT *Manufacturing Data, Density vs grain size*

b,c

***Figure 12: ADOPT** Manufacturing Data, Al v. Cr content*

SFNB RAI 7

Chapter 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, demonstrates that **ADOPT** fuel pellets promote larger fuel rod growth. In Figure 6-3, PWR fuel rod growth measurements are plotted against the PAD5 upper bound fuel rod growth model. As shown in Figure 6-3, the level of confidence in the upper bound growth model has clearly diminished based on the increased **ADOPT** fuel rods growth.

a. Please explain why the existing upper bound model remains valid.

b. If applicable, please describe the process for gathering future rod length measurements to confirm or revise the empirically based, high confidence growth model(s) used in fuel design analyses.

Response to SFNB RAI 7a.

Available measured data for **ADOPT** fuel demonstrates increased fuel rod growth relative to uranium dioxide (UO_2) fuel. This is evidenced by Figures 5-3, 5-4 and 5-5 of the **ADOPT** fuel topical report (WCAP-18482-P/WCAP-18482-NP). The increased growth is caused by the earlier pellet-cladding contact due to the increased density and reduced densification of **ADOPT** fuel. Earlier contact results in generally higher axial length growth. The effects of the **ADOPT** rod growth deviations at low and high fluence values are discussed below.

<u>Low Fluence Axial Rod Growth</u>

The axial growth for Pressurized Water Reactor (PWR) **ADOPT** fuel is clearly underpredicted at low fluence values. Lower growth at this time in life would cause an increase in rod internal pressure (RIP) due to the reduction in void volume (pressure and volume being inversely related). Higher RIP delays pellet-cladding gap closure, which is non-conservative for transient cladding strain and cladding fatigue. For transient cladding strain, which is limiting near the point of gap closure, the delayed contact means that the fuel has less potential energy during a limiting Condition II overpower transient. Lower Condition II transient powers cause less thermal expansion of the fuel during the accident. This is non-conservative for transient cladding strain. However, scoping studies have shown that rod growth uncertainties have a negligible impact on the strain criterion (**[ ]**^{a,c} as defined in Section 7.3.2 of WCAP-17642-P-A, Rev. 1).

Earlier pellet-cladding contact also affects cladding fatigue because of the increased cyclic loading seen by the fuel cladding. Axial growth uncertainty has a greater impact on cladding fatigue than it does for transient cladding stress **[ ]**^{a,c}**.** However, cladding fatigue is a non-limiting design criterion, and the increase in upper bound (UB) cumulative fatigue because of slightly higher RIP early in life is expected to be minor, especially as the growth uncertainty is statistically convoluted with other fatigue uncertainty cases. In other words, any slight increase in cumulative fatigue resulting from of an underprediction in axial rod growth is inconsequential in comparison to the available margin to the design limit.

<u>High Fluence Axial Rod Growth</u>

The measured growth data for **ADOPT** falls within the PAD5 UB growth model at high fluences, as shown in Figure 6-3 of the **ADOPT** fuel topical report (WCAP-18482-P/WCAP-18482-NP). As discussed in Section 5.9 of the PAD5 topical report (WCAP-17642-P-A, Rev. 1), measurements from two plants, designated as Plant L and Plant T, contain significantly higher rod growth data compared with all other samples. A conservative approach was taken for PAD5 by **[**

]^{a,c}. The result is a conservatively high UB growth model in PAD5 which remains bounding with respect to many of the **ADOPT** fuel measured data, including all data points at high fluence. However, the confidence level for axial growth at high fluence has diminished.

The primary fuel performance criteria affected by axial rod growth at high fluence (and burnup): RIP and fuel rod growth. Decreased axial growth is conservative for RIP because of the associated reduction in available void volume. RIP calculations are therefore performed with LB growth models, so the application of the UO_2 growth model for **ADOPT** fuel is conservative and acceptable.

The acceptance limit for fuel rod axial growth is that the 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. A loss of confidence for the UB axial growth of **ADOPT** fuel would be non-conservative because it would lead to earlier contact between the fuel rods and the top and bottom nozzles. However, as with cladding fatigue, there is margin to the fuel rod growth design limit. The UB axial growth rate can be increased by **[** **]**^{a,c}, and all current Westinghouse fuel lattices with **ZIRLO** High Performance Fuel Cladding Material will continue to meet the design limit up to **[** **]**^{a,c} burnup. For **Optimized ZIRLO** High Performance Fuel Cladding Material, which demonstrates reduced axial growth relative to **ZIRLO** cladding, the UB axial growth rate can be increased by **[** **]**^{a,c} and still meet the design limit up to **[** **]**^{a,c} burnup.

As additional measured data for **ADOPT** fuel is gathered (see response 'b' below), it may become preferential to re-evaluate the growth model for PAD5. In the interim, however, the current PAD5 growth model remains applicable for **ADOPT** fuel. The loss of confidence in the UB growth is either conservative (e.g., for RIP), negligible (e.g., for strain), or can be offset with available design margin (e.g., fatigue and fuel rod growth).

Response to SFNB RAI 7b.

Rod growth measurements will continue to be taken to confirm or revise the current empirically based growth model. Post-irradiation exams (PIEs) are planned for current and future Lead Test Rod (LTR) and Lead Test Assembly (LTA) programs containing **ADOPT** fuel. This includes both the ongoing Byron Unit 2 Accident Tolerant Fuel (ATF) LTR program and future LTR and LTA programs. Growth measurements are obtained using the Fuel Rod Handling Tool (FRHT) motor resolver in conjunction with the High Definition Shielded Video (HDSV) camera and High Definition Fuel Inspection

Recording Station (HD FIRSt) video recording system to measure and record the individual length of each inspected rod.

All additional measured data for **ADOPT** fuel shall eventually be incorporated into the PAD5 growth model. As stated in the executive summary for WCAP-18482-P/WCAP-18482-NP, Westinghouse is not currently seeking to take full advantage of all the benefits of **ADOPT** fuel. Subsequent licensing submittals are planned to further expand the approval of **ADOPT** fuel and more fully exercise the benefits of the fuel material. The appropriateness for the use of "simplified" models with the current submittal is based on extensive operating experience with **ADOPT** fuel in Europe, as well as demonstration that the inclusion of fuel dopants to UO_2 fuel does not introduce any new failure modes or require new or revised specified acceptable fuel design limits (SAFDLs). Discussion on the continued applicability of the existing PAD5 growth model is included in response 'a' above.

Future submittals for **ADOPT** fuel shall explicitly incorporate the planned additional test data, including axial fuel rod growth measurements. It may be preferential to modify the existing PAD5 growth model based on this new data. The current growth model is a function of only fluence and cladding material. An improved model can be developed to explicitly account for the differences between UO_2 and **ADOPT** fuel and the role of pellet-cladding contact on axial growth. As shown in Figures 5-3, 5-4 and 5-5 of the **ADOPT** fuel topical report (WCAP-18482-P/WCAP-18482-NP), Boiling Water Reactor (BWR) rods close the gap later than PWR rods and show the growth difference at higher fluences. For both BWR and PWR rods, the differences between **ADOPT** and UO_2 fuel rods are about **[ ]^{a,c}** at high fluences. While the final form of the model depends on the new measured axial growth data garnered from current and future ATF LTR and LTA programs, it is conceivable that a delta growth can be added to the UO_2 rod growth to account for the difference in gap closure, a saturated growth difference **[ ]^{a,c}**, and the transition from gap closure to the saturated delta growth. The uncertainty of **ADOPT** fuel rod growth is expected to be smaller due to high density and inherently smaller uncertainty in densification. Regardless of the form these model changes may or may not take, it is incumbent on Westinghouse to ensure that it continues to utilize an empirically based, high confidence growth model to perform fuel design analyses both now and in the future.

SFNB RAI 8

Section 5.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the empirical database used to confirm the adequacy of the fission gas release model for **ADOPT** fuel pellets. Most 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. Please provide details of the qualification of the poolside gamma scanning technique and comparison to hot-cell data.

Response to SFNB RAI 8

Poolside measurements of fission gas release (FGR) are relative comparisons of Kr-85 in the plenum and Cs-137 in the pellet stack. Kr-85 is the only radioactive fission gas with a half-life long enough to be measured nondestructively. The amount of Kr-85 released to the free rod volume can be determined by gamma spectrometry measurements in the plenum of a fuel rod. By comparing the amount in the free rod volume with the total amount formed by fission, the released portion can be calculated. By assuming that release rates for all fission gases are similar, the total amount of released fission gases can be assessed.

The qualification and verification of the poolside gamma scanning technique has been performed by comparison with destructive testing performed in a hotcell of the same or symmetrical fuel rods. Table 2 presents a comparison of poolside data and hotcell destructive testing on **[**
][b,c] and the results agree well, with only a very minor deviation between poolside and hotcell measurements, see Figure 13.

Poolside FGR measurements taken from large campaigns are typically confirmed by destructive hotcell examination of a smaller number of rods. If necessary, the examination data is used to develop an adjustment factor. In this case, the data agrees well, and no adjustments of the poolside data is necessary.

Table 2: *Comparison of poolside FGR measurements and hotcell FGR measurements*

[ ]b,c

b,c

[ ]

Figure 13: *Comparison of poolside FGR measurements and hotcell FGR measurements*

SFNB RAI 12.

With the expansion of load following in the U.S. nuclear fleet, PWRs may become increasingly susceptible to pellet-clad interaction (PCI)/SCC cladding damage during power maneuvering. While explicit NRC-approved fuel maneuvering limits and guidance are not mandated, licensees must have well defined power maneuvering guidance (e.g., ramp rate restrictions, pre-conditioning), based on irradiated fuel rod ramp testing, to ensure no fuel damage during normal operation. How has PCI/SCC been addressed for **ADOPT** fuel designs? Has power ramp testing been conducted to confirm existing power maneuvering guidance or establish new guidance?

Response to SFNB RAI 12

[

]^{b,c} Post ramp destructive PIE indicates that **ADOPT** fuel should provide more PCI/SCC margin than UO_2 fuel. The Westinghouse PWR maneuvering guidance is based on an extensive Zircaloy-4, **ZIRLO** and **Optimized ZIRLO** cladding ramp test database. The purpose of the limits is to allow for stress relaxation to limit the maximum cladding tensile stress from PCMI to levels below the threshold for PCI/SCC. The limits for the rate of power increase are designed to allow for stress relaxation due to cladding creep during continuous power increase.

With respect to **ADOPT**'s PCI resistance, it is expected that **ADOPT** fuel will exhibit improved PCI behavior. As discussed in Section 3.3.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, the high temperature creep rate is higher in **ADOPT** fuel than that of conventional UO_2; this enhanced viscoplastic behavior should provide a PCI benefit, as the pellet deforms under its own internal stresses and fills in as-manufactured dishes. Ceramography of ramped doped fuel similar in grain size and density, confirmed this behavior, as presented in Section 4.1.3 of the topical report. Additionally, incipient cracking of the liner was determined to be slightly improved.

[

]^{b,c,e}

SFNB RAI 13

Section 6.2.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, dispositions **ADOPT** fuel with respect to the applicability of existing guidance related to radionuclide release fractions, timing of releases, and elemental composition of releases. The potential impacts of the dopants and the larger grain size with respect to intergranular hold-up, rate of release of short-lived radionuclides, chemical interactions with dopants, fuel fragmentation, fuel melting temperature, etc. need to be addressed.

Response to SFNB RAI 13

A larger grain size results in lower fission product release rates from the fuel pellets according to Equation 5.7 of NUREG/CR-6261 ("A Summary of ORNL Fission Product Release Tests With Recommended Release Rates and Diffusion Coefficients," September 1995). Similarly, **[**

]^{a,c}. WCAP-18482-P/WCAP-18482-NP Section 3.2.3 concludes that there is no appreciable difference in the melting point of UO_2 and **ADOPT** fuel, so that would not result in earlier or higher release. **[**

]^{a,c}

The chemistry of fission products released from the fuel in a severe accident is not affected by the addition of the dopants. Obviously, noble gas fission products are nonreactive and are unaffected by anything. The chemistry of the iodine released from the fuel is dominated by the formation of CsI salt. Cesium is a fission product that is in abundance with respect to iodine and cesium is released from the fuel at a rate similar to iodine (both are volatile), and therefore iodine and cesium releases are not affected by the addition of Al_2O_3 and Cr_2O_3. The formation of other fission product compounds is affected primarily by the distribution of oxygen between the elements in the fuel pellet. There is already plenty of oxygen in the system in the UO_2 and PuO_2, as well as in the water, so the addition of oxygen from the dopants is negligible. As shown on the Ellingham diagram, at temperatures near the melting temperature, the oxygen potential is dominated by U and Pu with respect to the dopants Cr and Al. Therefore, the addition of the dopants does not significantly change the chemistry of the fission products released from the pellets during a core melt accident.

SFNB RAI 14

Please submit on the docket to the NRC Document Control Desk the following documents for further information to enable accurate safety evaluation:

1. **[**

   ]^{b,c}

2. **[**

   ]^{b,c}

3. **[**

   ]^{b,c}

4. JNST- 2006, "Advanced Doped UO2 Pellets in LWR Applications," Jakob ARBORELIUS, Karin BACKMAN, et.al Journal of Nuclear Science and Technology

5. [

]b,c

Response to SFNB RAI 14

With this transmittal, requested documents 1,2,3, and 5 are being submitted to the NRC Document Control Desk. Document 4 is publicly available.

Section E

Submittal of Responses to Request for Additional Information #2

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-6118
e-mail: schoedaj@westinghouse.com

LTR-NRC-21-20

June 20, 2021

Subject: Submittal of "Westinghouse Responses to RAIs 1, 9, 10, 11, and Supplemental Information for RAI 6 for WCAP-18482-P/WCAP-18482-NP, Revision 0" (Proprietary/Non-Proprietary)

Enclosed is the second set of responses to the Nuclear Regulatory Commission's ("Commission's") request for additional information regarding Westinghouse Electric Company LLC ("Westinghouse") Topical Report WCAP-18482-P/NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel." The RAI response labelled "Supplemental Information for SFNB RAI 6," gives information to supplement the response for RAI 6 previously transmitted by LTR-NRC-21-9 (ML21078A486).

This submittal contains proprietary information of Westinghouse. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the 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-21-5190 and should be addressed to Anthony J. Schoedel, Manager, eVinci™ Licensing & Configuration Management, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Cranberry Township, PA 16066.

Anthony J. Schoedel, Manager
eVinci™ Licensing & Configuration Management

cc: Ekaterina Lenning
Dennis Morey

Enclosures:
(1) Affidavit, AW-21-5190
(2) Westinghouse Responses to RAIs 1, 9, 10, 11 and Supplemental Information for RAI 6 on WCAP-18482-P/WCAP-18482-NP (Proprietary)
(3) Westinghouse Responses to RAIs 1, 9, 10, 11 and Supplemental Information for RAI 6 on WCAP-18482-P/WCAP-18482-NP (Non-Proprietary)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

COUNTY OF BUTLER:

(1) I, Anthony J. Schoedel, 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 the proprietary portions of LTR-NRC-21-20 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

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

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

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

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

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

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

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

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

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

(6) The attached documents in Enclosures 2 and 3 are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

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

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

Executed on: 6/20/2021

Anthony J. Schoedel, Manager
eVinci™ Licensing & Configuration
Management

Enclosure 3

Westinghouse Responses to RAIs 1, 9, 10, 11 and Supplemental Information for RAI 6 on WCAP-18482-P/WCAP-18482-NP

(Non-Proprietary)

(41 Pages Including Coversheet)

June 2021

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

SFNB RAI 1

Westinghouse has provided several technical reports and technical papers as part of its list of documents for staff review during the regulatory audit. Many of these documents are from International Atomic Energy Agency (IAEA), Journal of Nuclear Science and Technology (JNST), Organization of Economic Cooperation and Development Halden, Studsvik Cladding Integrity Program (SCIP), Barsebäck, and Studsvik. These documents are not supported by quality control (QC) requirements for appropriate quantitative or qualitative acceptance criteria as per Appendix B to Part 50, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*, of Title 10 of the *Code of Federal Regulations* (10 CFR).

What measures were taken to validate the above-mentioned 3rd party documents to adopt under Westinghouse QC procedures in accordance with 10 CFR Part 50 Appendix B.

An interim response from Westinghouse during the audit is reproduced in the table below. Westinghouse stated that more work is necessary, and a RAI question is needed.

a,c

Response to SFNB RAI 1

Westinghouse understands the question as to what steps were taken to confirm that the 3rd party documents satisfy established design requirements and comply with Westinghouse applicable procedures.

Consistent with Westinghouse procedures, the **ADOPT** fuel verification was conducted by qualified design engineers who are independent of and not directly involved in the original design. Records documenting evidence of the verification were collected, stored and maintained as lifetime records. Verification of **ADOPT** fuel data was based on UO_2 established properties and performance in Westinghouse BWR and PWR design. Testing data were generated by laboratories with proven capabilities in testing unirradiated and irradiated material and in accordance with written procedures that were reviewed and accepted by Westinghouse at the time the tests were performed. Test reports were reviewed by the Westinghouse responsible design engineer and documented in accordance with Westinghouse applicable procedures. Literatures and data generated by nationally and internationally recognized organizations in the nuclear science and development were reviewed by Westinghouse responsible design engineer in accordance with Westinghouse applicable procedures and were used for validation of Westinghouse Sweden (WSE) testing program and data. Some of the data generated by national and international recognized organizations were developed as a part of multi nations research programs in which the US government was a contributing member. Verification was performed in accordance with the Westinghouse Quality Management System which complies with the requirements of 10CFR50, Appendix B by implementing the guidance in NRC Regulatory Guide 1.28.

In addition, data generated within these material characterization and irradiation programs are evaluated for consistency. Within each campaign, reactor grade UO_2 is also tested and used as an accepted benchmark to confirm the validity of the experiment. For materials characterization, a comparison to widely accepted scientific literature has been performed, and it has been concluded that the test results are reasonable and valid. For data collected within an irradiation campaign, the cases are benchmarked using Westinghouse fuel rod design methods. This work is performed in accordance with the Westinghouse Quality Management System.

For thermo-physical and mechanical property characterization work, independent verification of the test reports has been performed by the responsible design engineer in accordance with the Westinghouse Quality Management System. Supplier generated spreadsheets and reports have been checked for accuracy and archived as lifetime records for configuration control. When possible, an independent verification of the data is performed using alternative methods, e.g. a comparison to widely accepted scientific literature. If concluded that the test results are reasonable and valid, they are accepted. Below is a brief description of the verification method for each material property:

Thermal diffusivity – The thermal diffusivity of **ADOPT** fuel was measured at the KTH Royal Institute of Technology in 1999. Measurements were made on two separate unirradiated **ADOPT** samples and compared with an unirradiated standard UO_2 sample. The UO_2 sample

was manufactured in accordance with the manufacturing specification and was inspected prior to testing. The **ADOPT** specimen satisfied all design constraints requested within the topical report with respect to density, grain size, and additive content. The thermal diffusivity analysis revealed an insignificant difference between **ADOPT** and standard UO_2 fuel.

Specific heat – The specific heat of **ADOPT** and standard UO_2 was measured using the differential scanning calorimeter (DSC) technique. Prior to measuring the **ADOPT** and HD UO_2 samples, the DSC instrumentation was calibrated with a sapphire sample. The average of the individual sapphire runs was shown to be within +/− 2% the known value. Following calibration, both UO_2 and **ADOPT** specimens were tested and found to be within the measurement capability of the method and concluded to be similar. Furthermore, the data was compared to published literature, and concluded to be valid.

Thermal expansion – The thermal expansion of **ADOPT** and standard UO_2 was measured using the differential dilatometer technique. Prior to measurement of the samples, the dilatometer instrumentation was calibrated using a known reference material, Al_2O_3. The testing procedures adhered to ASTM Standard Method E 831-86. Following calibration, a commercial grade UO_2 was tested. The results show that for the commercial UO_2 pellet the measured thermal expansion data coincides with the recommended average values for pure UO_2. This corroborates the validity of the experiments. Following confirmation of the validity of the experiments, both Westinghouse specimens (UO_2 and **ADOPT)** were tested and found to be within the measurement capability of the method and concluded to be similar. Furthermore, the data was compared to published literature, and concluded to be valid.

Melting temperature – For melting point determination, a procedure based on the laser heating to above the melting temperature followed by controlled cooling, was used. Prior to measurements of the sample, the equipment was calibrated using a standard tungsten strip lamp certified to the highest possible grade (temperature scale reproducibility at 2500 K is ±7 K). Following calibration, both UO_2 and **ADOPT** specimens were tested and concluded to be similar. Furthermore, the measured melting temperature of **ADOPT** (3122±7 K) was compared to published literature (3120±30 K for stoichiometric UO_2) and concluded to be valid.

Creep testing – Creep and hardening testing on unirradiated fuel pellets were performed at CEA's facilities. Westinghouse manufactured and inspected the **ADOPT** and standard UO_2 pellets prior to testing. The main objective of the experiment was to characterize the viscoplastic behavior of the fuel materials at elevated temperatures from 1300 to 1700°C. A review of the data was performed and accepted. It worth noting that this data is not directly being employed in safety related methods development; in other words, credit is not being taken for **ADOPT**'s enhanced viscoplasticity.

For irradiation programs, Westinghouse provides **ADOPT** and reference UO_2 fuel that is fabricated to Westinghouse manufacturing specification. Throughout the program, Westinghouse worked closely with the supplier to examine preliminary reports and provide input. Data were then reviewed and

archived. Results were evaluated for consistency with previous operating experience, as well as consistency with one another. Below is a brief description of **ADOPT** irradiation testing programs.

Studsvik Program – The Studsvik programs include: **[**

]^{a,c}

No anomalies were observed in each of the above described PIE campaigns, and the performance of UO_2 reference samples agreed with known UO_2 irradiation behavior. During the data postprocessing period, Westinghouse reviewed preliminary reports and provided approval prior to issuance of the final report. Qualitative and quantitative assessments of the PIE reports were performed and were consistent with previous experience and with each other. Additionally, Westinghouse FRD methods have been utilized with nominal input fuel parameters to simulate the behavior of rods throughout irradiation. The Studsvik generated reports and data have been accepted and archived as lifetime records.

OECD Halden Program – Westinghouse reviewed the preliminary reports and provided input prior to issuance of the final report. A qualitative analysis of the ceramography and metallography was performed; no anomalies were observed, and the performance of UO_2 reference sample agreed with the known UO_2 irradiation behavior. Furthermore, FRD methods have been utilized with nominal input fuel parameters to simulate the behavior of rods throughout irradiation. The experimental results were found to agree with Westinghouse methods. For these reasons, the data have been accepted and archived as lifetime records.

In conclusion, data generated within these material characterization and irradiation programs have been evaluated for consistency and are accepted per Westinghouse procedures.

SFNB RAI 6

Based on item number 6 in the audit plan, the NRC staff requests Westinghouse to provide documentation with details of data analysis for irradiation programs described in Section 4; specifically ramp and bump testing, fission gas measurements, fuel pellet cracking, and cladding metallography. Chapter 5 of WCAP-18482-P/WCAP-18482-NP, Revision 0, characterizes the material properties and performance of ADOPT fuel pellets. Please confirm that the empirical data presented in this chapter was collected on fuel with the same specifications as the final production ADOPT fuel. Identify and disposition any differences, if they exist.

Supplemental Information for SFNB RAI 6

Previously, Westinghouse docketed two documents detailing the steady state fission gas release analysis of **ADOPT** fuel operated **[**

]^{b,c}

Characterization and details of data analysis of **ADOPT** ramp and bump testing are provided as part of RAI 11. <u>This response details fuel pellet cracking and cladding metallography.</u>

EXAMINATION OF [ ]^{b,c} RODS

Two **ADOPT** variants and one UO_2 Westinghouse rods were irradiated in the **[** **]**^{b,c} experiment from 2004 to 2007, to a rod-average burnup 29 MWd/kgU. The fuel rods were operated at rod average LHRs of approximately 35-45 kW/m, which corresponds to around 1300-1500°C in the pellet center, see Table 6-1.

[ ]^{b,c}

CERAMOGRAPHY (FUEL PELLET CRACKING)

Macrographs of the fuel demonstrate **[** **]**^{b,c} This is because **[** **]**^{b,c} operated at high LHGRs, see ***Figure 6-1***, ***Figure 6-2***, and ***Figure 6-3***.

EXAMINATION OF OSKARSHAMN 3 RODS

Four fuel assemblies of the Westinghouse SVEA-96 Optima design, each containing a mixture of **ADOPT** rods and standard UO_2 rods for reference, were loaded in Oskarshamn 3 (O3) NPP in 2000. Two of these assemblies were chosen for higher burnup irradiation up to approximately 60 MWd/KgU, see Table 6-2. Following irradiation, one standard rod and one **ADOPT** rod from each assembly were transported to hot cell for post irradiation examinations. Power histories, and further discussion on base irradiation, for these rods is presented within Section 4.2 of WCAP-18482-P Rev. 0.

[ ]^{b,c}

CERAMOGRAPHY (FUEL PELLET CRACKING)

Macrographs of **ADOPT** and standard UO_2 fuel operated under steady-state conditions **[**

]^{b,c} The high-temperature conditions of the Halden reactor is the most likely differentiator. This crack pattern is much more prototypic of steady-state operations. See ***Figure 6-4***, ***Figure 6-5***, and ***Figure 6-6*** for steady-state irradiation ceramography.

Post-ramp macrographs show that **ADOPT** fuel had **[**

]^{b,c} This behavior is expected. Under high temperatures during a power ramp, **ADOPT** fuel will have a bigger central region with plastic creep deformation than conventional UO_2 due to its enhanced creep. A thinner brittle rim region has a finer crack pattern, which decreases the concentration of stresses onto the cladding, thereby reducing the mechanical interactions related to PCI. See ***Figure 6-7***, ***Figure 6-8***, ***Figure 6-9***, and ***Figure 6-10*** for post-ramp ceramography.

METALLOGRAPHY

Prior to ramp testing metallography was performed on the rods. Images of cladding hydriding is documented in the WCAP-18482-P Rev. 0. **[**

]^{b,c}

After power ramping, **[**

]^{b,c} see ***Figure 6-11***, ***Figure 6-12***, ***Figure 6-13***, and ***Figure 6-14*** for post-ramp metallography.

EXAMINATION OF THE RAMPED BARSEBÄCK 2 SEGMENTS

Two segmented fuel rods were irradiated in Barsebäck 2 nuclear power plant up to segment burnup average of 33.5 MWd/kgU. Following base irradiation, several segments were ramp tested in the Studsvik R2 reactor. Additional details of the segment's irradiation history are located in Section 4.1 of WCAP-18482-P.

[ ]^{b,c}

CERAMOGRAPHY (FUEL PELLET CRACKING)

Ceramography of the ramp segment 33004 with 1000 ppm Cr_2O_3 revealed a central void, unlike the standard pellet from segment 33002. The void appeared in the region of the segment which had been subjected to the highest powers. The additives enhance the pellet's viscoplasticity at these high temperatures, filling the pellet's dished area during power ramp test. Small radial cracks are observed at the periphery in both pellet types; however, the rim structure is smaller in the doped pellet, see ***Figure 6-15*** and ***Figure 6-16***.

Ceramography of segments 33013 and 33003 post ramp revealed many cracks at the periphery of the pellets. As previously demonstrated, there is more radial cracking in the **ADOPT** pellets, see ***Figure 6-17*** and ***Figure 6-18***.

METALLOGRAPHY

Metallography of segments 33002 and 33004 revealed a large number of incipient cracks on the inner cladding. The length of the incipient cracks was measured **[** **]**^{b,c} No incipient cracks had propagated further than to the boundary of this layer, see ***Figure 6-19***.

In an attempt to estimate a qualitative difference in PCI performance between the two pellet types, the number of non-oxidized cracks, formed during the ramp test, was counted. The number of incipient cracks can be assumed to be a measure of the pellet's PCI performance. During ramp testing, **[** **]**^{b,c}

[

]^{b,c} This is expected as the fuel was ramped to high powers for a long hold time, see ***Figure 6-20***.

Summary and Observations

The performance of the **ADOPT** pellet has been examined up to a rod average burnup of about **[** **]**^{b,c}

Ceramography indicates **[** **]**^{b,c} Power ramped **ADOPT** fuel exhibits a larger central plastic region, and a thinner rim structure with more prominent radial cracking, which is advantageous for pellet clad mechanical interaction. Metallographic examination of incipient cracking in ramped fuel confirmed this beneficial PCI behavior.

b,c

b,c

b,c

b,c

b,c

b,c

b,c

b,c

b,c

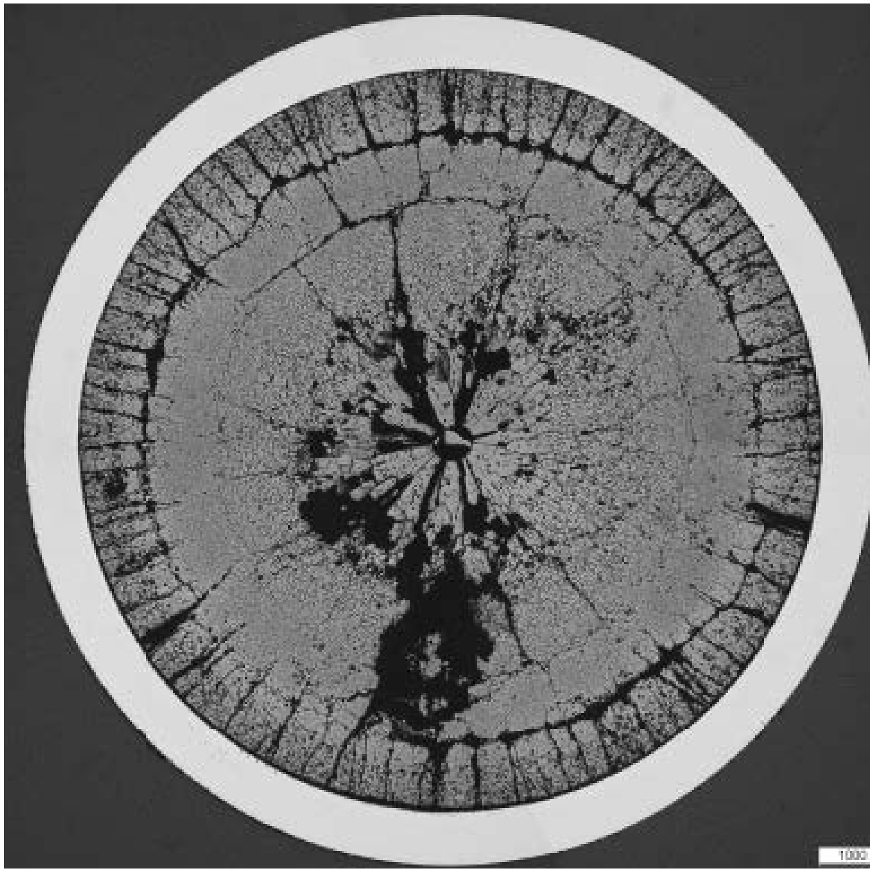


Figure 6-15: *Macrograph of 1000ppm Cr_2O_3 pellet post ramp. Etched condition.*

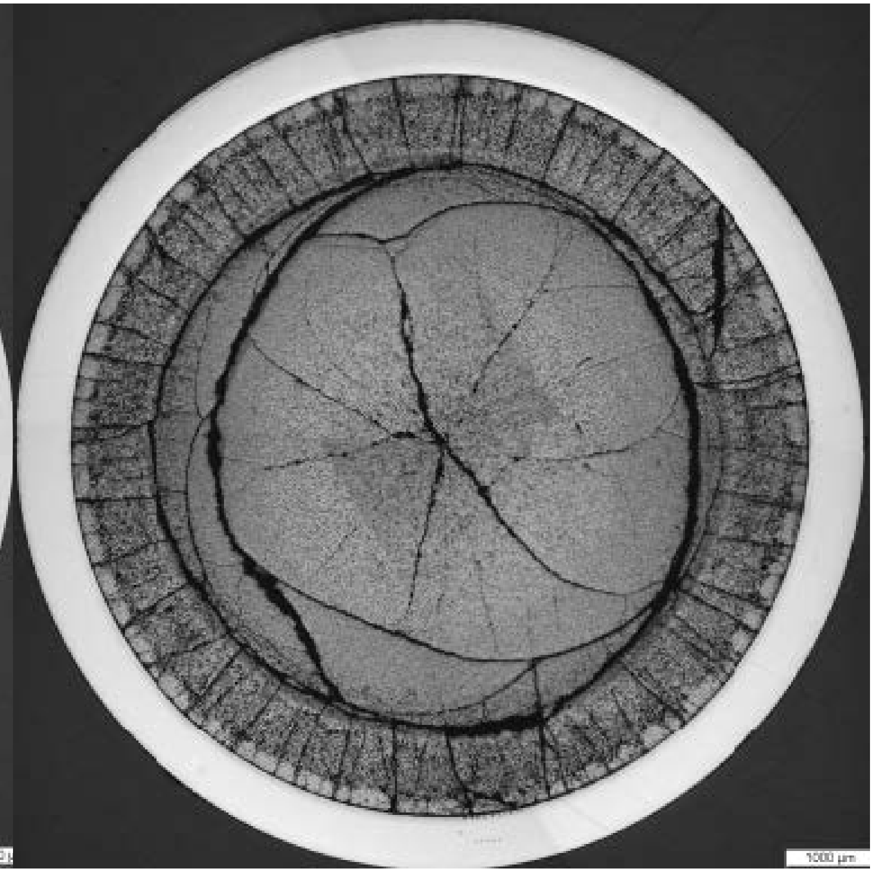


Figure 6-16: *Macrograph of UO_2 pellet post ramp. Etched condition.*

b,c

b,c

SFNB RAI 9

Section 5.5 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes potential impacts of ADOPT fuel pellets on fuel rod performance under reactivity-initiated accident (RIA) conditions. Japan Atomic Energy Agency (JAEA) Top Fuel technical paper (Takeshi Mihara, et. al.) documents an investigation of recent prompt pulse tests conducted on doped UO2 fuel pellets. Test rod OS-1, which contained ADOPT fuel pellets, failed at 38 Δcal/g, which is below all previous test failures for rods close in BU and hydrogen content. The JAEA technical paper concluded the following with respect to the results of the OS-1 test:

> *The pre- and post-test examinations suggested that one of the reasons of the lower failure limit may be the effect of the hydrides radially oriented and precipitated more densely in the specific angle range in the cladding tube. However, since the possible contribution of ADOPT™–pellets specific effects cannot be ruled out at the present, further investigation is needed on fuel pellet behavior under both normal-operation and pulse-irradiation conditions.*

a. Please provide evidence (e.g., irradiated ADOPT fuel rod segments used in expansion due to compression testing) that differences between UO2 and ADOPT fuel performance (i.e., steady-state swelling, pellet-clad mechanical interaction) will not introduce differences in cladding properties and microstructure (e.g., hydride morphology) which may influence fuel rod performance under RIA conditions.

b. Hundreds of prompt pulse tests have been conducted in research reactors such as the Nuclear Safety Research Reactor (NSRR) to evaluate the performance of irradiated fuel rods under RIA conditions. In these test programs, comparisons have been made between large grain and standard grain fuel rod performance (e.g., NSRR test OI-10 vs. OI-11). Tests have also been performed on a variety of different fuel compositions and doping agents. Please review the extensive RIA empirical database and compile data to characterize the fuel thermal expansion of ADOPT fuel under RIA conditions and support your finding that ADOPT fuel will behave similarly to standard UO_2 fuel.

Response to SFNB RAI 9a

WCAP-18482-P/WCAP-18482-NP, Rev. 0 provides the details of the expansion due to compression (EDC) test, a high strain test intended to simulate the PCMI behavior during RIAs, of an **ADOPT** rod and a UO_2 rod. The test results, which are detailed in Section 5.5.1, suggest that the test temperature is the controlling factor for a cladding failure. **[**

]^{b,c}

Metallography of irradiated samples from the same fuel rod tested in the EDC tests are presented in Figure 4-11 and Figure 4-12 of the topical report. These images were taken at Studsvik as part of PIE on the parent rods before segments were transported to Japan for RIA testing. They are reproduced here as Figures 9-1 9-2 for convenience. A visual comparison of the rods reveals that both rods are very similar. 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. Both rods appear to have a consistent ratio of radial to tangential hydrides around the rod. **ADOPT** fuel's increased density, and earlier in life pellet clad contact, does not appear to contribute any excessive radial hydride reorientation.

b,c

Response to SFNB RAI 9b

A review of the available literature has been performed to characterize the strain associated with fission gas induced swelling of large grained fuels. The Nuclear Safety Research Reactor (NSRR) has tested a wide variety of fuels, however, there is limited database for large grained fuels, similar to that of **ADOPT** fuel. Relevant tests are summarized below and in Table 9-1:

1) NSRR Test OI-10 [Ref. 9-1]

The OI-10 test rodlet contained a large grained UO_2 (28 μm) fuel with 60 MWd/kgU burnup and Mitsubishi Developed Alloy (MDA) cladding, a zirconium-based alloy for high burnup PWR applications. In terms of performance, MDA cladding should behave closer to that of **ZIRLO** or **Optimized ZIRLO** cladding, with its reduced propensity for hydrogen pickup and radial hydriding, as it is designed for a better performance in terms of corrosion resistance. The rod was subjected to pulse-irradiation conditions of 435 J/g (104 cal/g) peak fuel enthalpy and 5.6 ms pulse-width. At the conclusion of the test, the fuel remained intact, and there was minimal rod deformation.

The residual strain in NSRR OI-10 was measured to be below 0.7%, significantly less than those observed in previous tests. For comparison, the residual hoop strain in high burnup PWR fuel rods after pulses at a similar peak fuel enthalpy level was measured to be approximately 5%. This suggests the pellet expansion was largely generated by solid thermal expansion, and that the fission gas swelling associated with the enhanced fission gas retention of large grained fuels (like the **ADOPT** pellet) is minimal.

2) NSRR Test MR-1 [Ref. 9-2]

The MR-1 test fuel rod contained a large grained (40 μm) UO_2 pellet and an (New Developed corrosion resistant Alloy) NDA cladding, which is a zirconium-based cladding alloy for PWR applications. The fuel had been base irradiated in McGuire nuclear power plant (NPP) and the R2 research reactor up to a rod burnup of 71 MWD/kgU. The rod was subjected to a $4.6 reactivity insertion pulse and a peak fuel enthalpy of 371 J/g (89 cal/g) in room temperature water and atmospheric pressure.

Rod puncturing showed a fission gas release of ~8.7% during the pulse-irradiation. The cladding residual hoop strain was estimated as ~0.4% with rod diametral measurements before and after the pulse. Similar to the previous large grained UO_2 fuel test [OI-10], the residual strain was quite low, suggesting the pellet expansion was largely generated by solid thermal expansion. The fuel remained intact, i.e. no failure occurred.

3) NSRR Test LS-4 [Refs 9-3, 9-4]

The LS-4 test rod contained a large grained chromia doped UO_2 with 48 MWd/kgU burnup and Zircaloy 2 cladding for BWR applications. Per Reference 4, this fuel design nominally contains 0.16 wt.% chromium oxide and a grain size up to 50 to 60 μm. The LS-4 test fuel rod was pulse irradiated to a maximum enthalpy increase of 559 J/g and did not fail. The residual hoop strain of

the cladding was calculated to be about 2.4% at maximum. This strain exceeds the level achieved by pellet thermal expansion, which was evaluated to be less than 2%.

Table 9-1, Summary of Large Grain RIA Tests

Test	Alloy	Reactor Type	Burnup, MWd/kgU	Pellet Additive	Grain Size, µm	Max Enthalpy Increase, J/gm	Residual Strain, %
OI-10	MDA	PWR	60	None	28	435	< 0.7
MR-1	NDA	PWR	71	None	40	371	~0.4
LS-4	Zr-2	BWR	48	~ 0.16 wt.% Cr_2O_3	~ 50 to 60	559	2.4

In both the NSRR Test OI-10 and MR-1, the residual hoop strain was quite low and most likely was generated by solid thermal expansion. Both tests contained pellets similar in grain size to that of **ADOPT**, and it can be reasoned that they would both exhibit similar fission gas swelling. In neither case did fission gas swelling appear to contribute to the hoop strain. Furthermore, both test rods remained intact. The LS-4 test is the only other large grained, additive (in similar wt% quantities) fuel which has been RIA tested. The LS-4 fuel pellet nominally contains 0.16 wt.% chromia and has a grain size of 50 to 60 µm [Ref. 9-4]. **[**

]^{b,c} It is reasonable to assume that the LS-4 test fuel should behave similarly. Furthermore, the test rod did not fail and remained intact.

The above-mentioned tests are the only large-grained test data available, and it is expected that **ADOPT** fuel will behave similarly. Although it is a limited dataset, there is no information to suggest that additives or large grains have a negative impact on the PCMI failure enthalpy.

References:

9-1 T. Fuketa, T. Sugiyama, & F. Nagase (2006). “Behavior of 60 to 78MWd/kgU PWR Fuels under Reactivity-Initiated Accident Conditions.” Journal of Nuclear Science and Technology, 43:9, 1080-1088

9-2 T. Fuketa, T. Sugiyama, M. Umeda, K. Tomiyasu, H. Sasajima. “Behaviour of High Burnup PWR Fuels During Simulated Reactivity-Initiated Accident Conditions.”

9-3 T. Mihara, Y. Udagawa, M. Amaya, Y Taniguchi, K. Kakiuchi. “Behavior of LWR Fuels with Additives Under Reactivity-Initiated Accident Conditions.” Top Fuel. September 2019.

9-4 C. Delafoy, P. Dewes, T. Miles. “AREVA NP Cr_2O_3-Doped Fuel Development for BWRs.” Proceedings of the 2007 International LWR Fuel Performance Meeting. September 2007.

SFNB RAI 10

WCAP-18482-P/WCAP-18482-NP, Revision 0, Section 6.1.1 describes potential impacts of ADOPT fuel pellets on individual PAD5 models. **[**

]a,c Please provide further details of the PAD5 assessment.

Response to SFNB RAI 10

The PAD5 calibration and validation presented in PAD5 topical report (Reference 10-1) are based on an extensive database for fuel rods with UO_2 pellets, including many data for fuel rods with burnup higher than **[** **]a,c** In addition to the fuel centerline temperature results presented in Section 6.1.1 of the **ADOPT**™ topical report (Reference 10-2) , PAD5 results for **ADOPT** integral fuel performance data in thermal FGR, athermal FGR, transient FGR, and steady-state diameter change are shown below to confirm the applicability of PAD5 for **ADOPT** fuel. The cladding diameter change results for the ramp tests are provided in the response to RAI-11.

<u>Thermal FGR</u>

Thermal FGR results for IFA-677.1 were also provided in Section 6.1.1 of the **ADOPT** topical report (Reference 10-2). 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. 10-3 were provided in Table 8-1 and Figure 8-1 in the response to RAI-8a of PAD5 Topical Report (Reference 10-1). 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 thermal fission gas release model is confirmed to be applicable to **ADOPT** fuel. **[**

]a,c

<u>Athermal FGR</u>

Figure 5-6 of the **ADOPT** topical report (Reference 10-2) showed the steady state FGR from O3 (Oskarshamn 3 Reactor) and **[**

]a,c

Transient FGR

Sections 4.1.3 and 4.3 of the **ADOPT** topical report has shown that **ADOPT** fuel has less transient FGR for the same test conditions than UO_2 fuel. To conservatively model transient fission gas release for **ADOPT** fuel, the same transient FGR model for UO_2 fuel is also used for **ADOPT** fuel. PAD5 results for these ramp test confirmed the conservatism for **ADOPT** fuel. Table 10-3 provides comparisons of transient fission gas release results for the Barseback2-Studsvik ramp tests. **[**

][a,c] Table 10-4 provides comparison of transient fission gas release results for the O3-SCIP II ramp tests. **[** **]**[a,c]

Steady-State Cladding Diameter Change

The cladding diameter change after fuel-cladding gap closure is impacted by the pellet deformation, i.e., thermal expansion, pellet relocation, fuel solid swelling, and fission gas swelling. There are limited profilometry data. The pre-ramp diameters were measured for the Barseback2-Studvik and O3-SCIP II ramp tests. The measured and predicted profilometries for the pre-ramp are shown in Figures 10-1 and 10-2 for Barseback2-Studsvik ramp tests and Figures 10-3 and 10-4 for O3-SCIP II tests. There is good agreement between the predicted and measured cladding diameters.

Cladding Diameter Change from Ramp Tests

Diameter change from ramp tests are modeled with PAD5 in the response to RAI-11. The results show similar results for UO_2 and **ADOPT** fuel.

In summary, PAD5 models are well calibrated for UO_2 fuel, with an extensive database, including high burnup data above **[** **]**[a,c] The PAD5 results for **ADOPT** presented in this response demonstrate that PAD5 models are adequate for **ADOPT** fuel.

References:

10-1 WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.

10-2 WCAP-18482-P, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel," May 2020.

10-3 Halden Report HWR-968, "PIE Report on Six UO_2 Fuel Rods Irradiated in IFA-677 High Initial Rating Test," March 2010.

[ ]a,c

a,c

[ ]a,c

[ ]a,c

a,c

a,c

a,c

a,c

SFNB RAI 11

Section 6.1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes validation of the PAD5 gaseous swelling model against a Studsvik ramp test with ADOPT rods. In this discussion Westinghouse states that the volume change from the Studsvik test show that **[**

][a,c]

Based on past fuel design reviews, the NRC staff is aware of multiple ramp tests on irradiated fuel rod segments containing large grain fuel pellets which exhibited larger incremental cladding strain relative to tests conducted on irradiated fuel rod segments with standard UO_2 pellets. **[**

][a,c]

a. Please provide incremental cladding strain measurements from ramp testing on irradiated fuel rod segments containing ADOPT fuel pellets.

b. Please provide details of the PAD5 fuel thermal expansion and gaseous swelling model validations, including PAD5 predictions of incremental cladding strain versus measurements.

Response to SFNB RAI 11a

There are two test series: Barseback-Studsvik and O3-SCIP II (Sections 4.1 and 4.3 of Reference 11-2). The measurements for the incremental cladding strain are presented in part b of this response along with the predicted value. Consistent with Section A.2.4 of the PAD5 topical Report (Reference 11-1), the incremental cladding strain are presented as cladding diameter change from ramp tests.

Response to SFNB RAI 11b

The volume change of the UO_2 and **ADOPT** fuel from the Barseback-Studsvik test at power level of 57 kW/m are shown in Figure 4-6 of the **ADOPT** topical report. The cladding diameter change for the same two rods are shown in Figure 11-1 for **ADOPT** fuel and Figure 11-2 for UO_2 fuel. **[**

][a,c] The result for other two rods from the same test at a lower power level (45 kW/m) are shown in Figure 11-3 for **ADOPT** fuel and Figure 11-4 for UO_2 fuel, where **[**

][a,c]

The diameter changes for the O3-SCIP II ramp tests are shown in Figures 11-5 to 11-8. Since the ADOPT rods are ramped at much higher power level than UO_2 rods, the ADOPT fuel rodlets show incremental diameter change while the UO_2 rods shown very little diameter change in measurement.

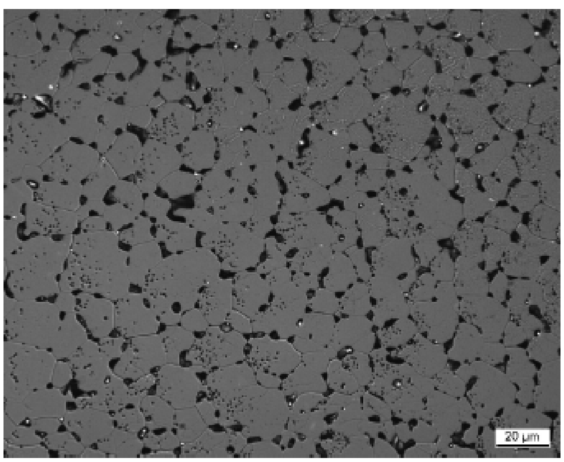
[

][a,c]

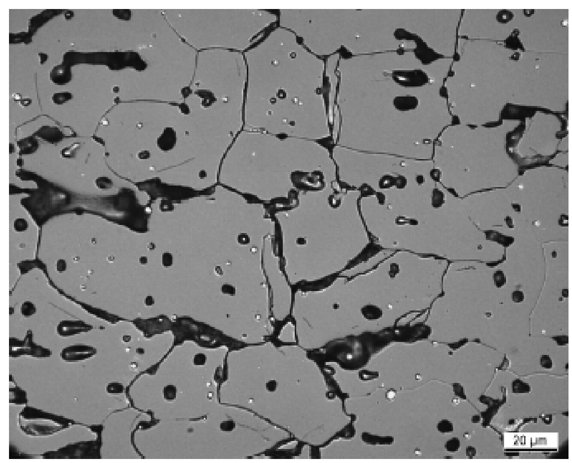
For irradiated UO_2 fuel, fission gas atoms will diffuse through the fuel grains and accumulate at the grain boundary. When gas concentration reaches a certain amount, release tunnels will form, and fission gas will be released to the free volume. During this process, fission gas bubbles form inside the grains as very fine bubbles and on the grain boundary as larger bubbles (diameters in the microns). It is the grain boundary bubbles that contribute to the majority of the fuel gas swelling. Fission gas release will stabilize the gas swelling of the fuel.

Much of the mechanism is the same for **ADOPT** fuel, except that larger intragranular bubbles can form as shown in Figure 4-4 of the **ADOPT** topical (also shown below for convenience). It should be noted that **[**

][a,c]



UO_2



ADOPT

[

][a,c]

[

]^{a,c}

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]^{a,c}

References:

11-1 WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.
11-2 WCAP-18482-P, "Westinghouse Advanced Doped Pellet Technology (**ADOPT**™) Fuel," May 2020.

a,c

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Section F

Submittal of Responses to Request for Additional Information #3

Westinghouse

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LTR-NRC-21-34
November 12, 2021

Subject: Submittal of "Westinghouse Revised Responses to RAIs 7a, 11, and Supplemental Response to RAI 9 for WCAP-18482-P/WCAP-18482-NP"

Enclosed are supplemental and revised responses to the Nuclear Regulatory Commission's ("Commission's") request for additional information regarding Westinghouse Electric Company LLC ("Westinghouse") Topical Report WCAP-18482-P/NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel." The enclosed responses give information to supersede or supplement responses for RAIs previously transmitted by LTR-NRC-21-9 (ML21078A485) and LTR-NRC-21-20 (ML21172A294).

This submittal contains proprietary information of Westinghouse. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the ("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-21-5237 and should be addressed to Anthony J. Schoedel, Manager, eVinci™ Licensing & Configuration Management, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Cranberry Township, PA 16066.

Anthony J. Schoedel, Manager
eVinci™ Licensing & Configuration Management

cc: Ekaterina Lenning
Dennis Morey

Enclosures:

(1) Affidavit, AW-21-5237

(2) Westinghouse Revised Responses to RAIs 7a, 11, and Supplemental Response to RAI 9 for WCAP-18482-P/WCAP-18482-NP (Proprietary)

(3) Westinghouse Revised Responses to RAIs 7a, 11, and Supplemental Response to RAI 9 for WCAP-18482-P/WCAP-18482-NP (Non-Proprietary)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

COUNTY OF BUTLER:

(1) I, Anthony J. Schoedel, 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 the proprietary portions of LTR-NRC-21-34, Enclosure 2 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

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

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

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

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

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

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

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

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

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

(6) The attached documents in Enclosures 2 and 3 are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

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

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

Executed on: 11/11/2021

Anthony J. Schoedel, Manager
eVinci™ Licensing & Configuration Management

Enclosure 3

Westinghouse Revised Responses to RAIs 7a, 11, and Supplemental Response to RAI 9 for WCAP-18482-P/WCAP-18482-NP

(Non-Proprietary)

(17 pages Including Coversheet)

November 2021

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

SFNB RAI 7a

Chapter 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, demonstrates that **ADOPT™** fuel pellets promote larger fuel rod growth. In Figure 6-3, PWR fuel rod growth measurements are plotted against the PAD5 upper bound fuel rod growth model. As shown in Figure 6-3, the level of confidence in the upper bound growth model has clearly diminished based on the increased **ADOPT** fuel rods growth.

a. Please explain why the existing upper bound model remains valid.

NRC Clarification Request on RAI 7

Section 5.2.2 of WCAP-18482-P describes Westinghouse's fuel rod growth measurements on both BWR and PWR ADOPT™ fuel rods. An important impact of ADOPT fuel reduced in-reactor densification is an earlier closure of the fuel pellet-to-cladding gap. After gap closure, irradiation-induced fuel swelling will strongly influence fuel rod growth. The empirical database clearly shows an increase in fuel rod growth compared to standard UO2 fuel rods. In response to an RAI regarding the continued applicability of the PAD5 upper bound growth model (RAI #7a Reference b), Westinghouse states that the current PAD5 growth model remains applicable for ADOPT fuel and any loss of confidence in the UB growth is either conservative (e.g., for RIP), negligible (e.g., for strain), or can be offset with available design margin (e.g., fatigue and fuel rod growth). Please clarify how credit for available margins relative to design criteria (to offset known non-conservative aspects of an analytical method) is captured and tracked. The staff has concerns with this response in that credit for available margins relative to design criteria, to offset known non-conservative aspects of an analytical method, need to be captured and tracked.

Response to SFNB RAI 7a.

Following clarification discussions with the NRC, the response to RAI #7a has been revised as provided below. The following response supersedes the response provided previously in Reference 7-1.

Figure RAI-7-1 updates Figure 6-3 of Reference 7-2 with **[** **]**^{a,c}. The figure displays rod axial growth data for **ADOPT [** **]**^{a,c} and standard UO_2 pellets **[** **]**^{a,c}. As stated in Reference 2, these data were collected as part of **[** **]**^{a,c}. The data presented in Figure RAI-7-1 suggest that the use of **ADOPT** fuel pellets results in an increase in rod axial growth. Westinghouse has developed an adjustment to the current PAD5 model for rod axial growth that accounts for this increase in axial growth. Westinghouse performed a quantitative evaluation of the impact of this adjustment on the fuel rod design criteria that are sensitive to rod growth.

Figure RAI-7-2 shows the data in Figure RAI-7-1 overlaid on the PAD5 database for the rod growth model for UO_2 fuel. For clarity, **[**]^{a,c}

[ ][a,c]. Figure RAI-7-2 also shows plots of the PAD5 rod growth model for UO_2 fuel.

In Figure RAI-7-2 the PAD5 database is **[** **]**[a,c]. As mentioned in Reference 7-1, there were **[** **]**[a,c]. This provides an extra level of conservatism when considering the entire dataset.

In developing an adjustment for **ADOPT** fuel, Westinghouse **[** **]**[a,c].

[ ] a,c

*Figure RAI-7-1 Rod axial growth data for **ADOPT** and standard UO_2 pellets. The PAD5 upper-bound model is also shown*

a,c

Figure RAI-7-2 Comparison of the rod growth data from Figure RAI-7-1 to the PAD5 database and model

[

][a,c]. The resulting adjusted model is plotted in Figure RAI-7-4 along with the data for **ADOPT** fuel. **[**

][a,c].

[ ]^{a,c}

Figure RAI-7-3 A model for the effect of ***ADOPT*** *fuel on rod growth*

Westinghouse used this adjusted model to quantitatively evaluate the impact on the fuel rod design criteria more sensitive to rod growth: rod internal pressure, fuel rod growth, cladding fatigue, and cladding transient strain. As expected, the impact on the rod internal pressure criterion was found to be in the direction of increasing the margin. The impact on the fuel rod growth criterion is to reduce margin, as expected. **[** **]**^{a,c}. The impact on cladding fatigue **[** **]**^{a,c}.

Westinghouse **[** **]**^{a,c}. As stated in the response to RAI #7b (Reference 7-1) Westinghouse plans to collect more performance data for **ADOPT** fuel. Future rod growth data for **ADOPT** fuel will be incorporated into Westinghouse fuel performance models and methods using Westinghouse processes for model and methods development, as were used to develop the approved PAD5 model and methods. This process will continue to ensure that the models used to evaluate the fuel rod design criteria include adequate level of confidence and conservatism.

a,c

*Figure RAI-7-4 Rod axial growth data for **ADOPT** and adjusted PAD5 model*

References

7-1 LTR-NRC-21-9, "Submittal of "Westinghouse Responses to RAIs 2, 3, 4, 5, 6, 7, 8, 12, 13, and 14 for WCAP-18482-P/WCAP-18482-NP, Revision 0 (Proprietary/Non-Proprietary)"", March 2021.

7-2 WCAP-18482-P, "Westinghouse Advanced Doped Pellet Technology (ADOPT^{TM}) Fuel," May 2020.

7-3 WCAP-17642-P-A, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.

SFNB RAI 9

Section 5.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes potential impacts of ADOPT fuel pellets on fuel rod performance under reactivity-initiated accident (RIA) conditions. Japan Atomic Energy Agency (JAEA) Top Fuel technical paper (Takeshi Mihara, et. al.) documents an investigation of recent prompt pulse tests conducted on doped UO2 fuel pellets. Test rod OS-1, which contained ADOPT fuel pellets, failed at 38 Δcal/g, which is below all previous test failures for rods close in BU and hydrogen content. The JAEA technical paper concluded the following with respect to the results of the OS-1 test:

The pre- and post-test examinations suggested that one of the reasons of the lower failure limit may be the effect of the hydrides radially oriented and precipitated more densely in the specific angle range in the cladding tube. However, since the possible contribution of ADOPT^{TM}–pellets specific effects cannot be ruled out at the present, further investigation

is needed on fuel pellet behavior under both normal-operation and pulse-irradiation conditions.

a. Please provide evidence (e.g., irradiated ADOPT fuel rod segments used in expansion due to compression testing) that differences between UO2 and ADOPT fuel performance (i.e., steady-state swelling, pellet-clad mechanical interaction) will not introduce differences in cladding properties and microstructure (e.g., hydride morphology) which may influence fuel rod performance under RIA conditions.

b. Hundreds of prompt pulse tests have been conducted in research reactors such as the Nuclear Safety Research Reactor (NSRR) to evaluate the performance of irradiated fuel rods under RIA conditions. In these test programs, comparisons have been made between large grain and standard grain fuel rod performance (e.g., NSRR test OI-10 vs. OI-11). Tests have also been performed on a variety of different fuel compositions and doping agents. Please review the extensive RIA empirical database and compile data to characterize the fuel thermal expansion of ADOPT fuel under RIA conditions and support your finding that ADOPT fuel will behave similarly to standard UO_2 fuel.

Supplemental Response to SFNB RAI 9

The submittal of this supplement follows a clarification discussion with the NRC on August 8th, 2021.

This is a supplement to the previous response to a request for additional information (RAI) 9 [9-1] on ADOPT fuel pellets. This supplement adds the basis for applicability of the current UO_2 pellet cladding mechanical interaction (PCMI) cladding failure criterion under a rod ejection accident (REA) to the **ADOPT** fuel pellets. The PCMI cladding failure enthalpy for **ZIRLO** or **Optimized ZIRLO** cladding with **ADOPT** fuel pellets in an REA is expected to be represented by the failure curves for stress relieved annealed (SRA) cladding at temperatures > 500 °F, as presented in RG 1.236 [9-2]. The main factors impacting the PCMI failure criteria are the expansion of the pellet, the overall cladding hydrogen level in excess of the solubility levels, H_{EX}, and the orientation of hydrides in the cladding, in particular, the fraction of hydrides that are radially oriented.

In the response to RAI 9, **ADOPT** fuel pellets were shown to be equivalent to UO_2 pellets in terms of the resulting pellet expansion from the energy pulse associated with a REA. In addition, the pellet to clad gap at low power levels (< 5% Full Power) at the end of cycle where the REA is more limiting would be similar. Therefore, the cladding strain during an REA would be similar.

Previously, **Optimized ZIRLO** cladding had been justified to have the same PCMI enthalpy rise limits as **ZIRLO** cladding. This position was documented internally at Westinghouse and was demonstrated based on the following evidence:

a. Comparison of ductility as a function of hydrogen for irradiated cladding for **ZIRLO** and **Optimized ZIRLO** cladding.

b. Comparison of hydride orientation and reorientation under tensile stress for un-irradiated **ZIRLO** and **Optimized ZIRLO** cladding.
c. Comparison of hydride orientation of irradiated **ZIRLO** and **Optimized ZIRLO** cladding.

The evaluation of this evidence demonstrated that the trend in cladding ductility as a function of hydrogen, hydride distribution after irradiation, and hydride reorientation behavior under stress are similar for **ZIRLO** and **Optimized ZIRLO** cladding. This position was reviewed and approved following an audit by the NRC [9-3]:

> "The NRC staff agreed with this comment. On February 13, 2020, the staff conducted an audit of the Westinghouse documentation supporting the application of the SRA PCMI cladding failure threshold to Optimized ZIRLO. Based on the results of the audit (ADAMS Accession No. ML20049F944), the staff accepts the use of the SRA PCMI cladding failure thresholds for Optimized ZIRLO cladding."

Based on this, the cladding failure strain should be the same for **ADOPT** and UO_2 fuel under a REA. There is no change in any UO_2 fuel failure threshold and criteria for the fuel rods containing the ADOPT pellets. Therefore, the PCMI limits of RG 1.236 are applicable to **ADOPT** fuel pellets with either **ZIRLO** or **Optimized ZIRLO** cladding.

References:

9-1 LTR-NRC-21-20, Submittal of "Westinghouse Responses to RAIs 1, 9, 10, 11, and Supplemental Information for RAI 6 for WCAP-l 8482-P/WCAP-18482-NP, Revision 0" (Proprietary/Non-Proprietary), June 20, 2021.
9-2 ADAMS Accession No. ML20055F490, REGULATORY GUIDE RG 1.236, Revision 0, PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION AND BOILING-WATER REACTOR CONTROL ROD DROP ACCIDENTS, June 2020.
9-3 ADAMS Accession No. ML20055F489, Response to Second Round of Public Comments, Draft Regulatory Guide (DG)-1327, (NRC Docket-2016-0233; ADAMS Accession No. ML20055F489), June 2020.

SFNB RAI 11

Section 6.1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes validation of the PAD5 gaseous swelling model against a Studsvik ramp test with ADOPT rods. In this discussion Westinghouse states that the volume change from the Studsvik test show that **[**

][a,c]

Based on past fuel design reviews, the NRC staff is aware of multiple ramp tests on irradiated fuel rod segments containing large grain fuel pellets which exhibited larger incremental

cladding strain relative to tests conducted on irradiated fuel rod segments with standard UO_2 pellets. **[**

][a,c]

a. Please provide incremental cladding strain measurements from ramp testing on irradiated fuel rod segments containing ADOPT fuel pellets.

b. Please provide details of the PAD5 fuel thermal expansion and gaseous swelling model validations, including PAD5 predictions of incremental cladding strain versus measurements.

NRC Clarification Request on RAI 11

Section 4.1 of WCAP-18482-P describes a base irradiation and subsequent ramp testing program conducted on two BWR, barrier lined, segmented rods containing standard UO_2 (D0) and two different variants of doped UO_2 fuel (D1 and D3). Ceramography (D0, D1), fission gas release measurements (D0, D1, D3), and fuel volume change (D0, D1), calculated from profilometry and gap measurements, are presented. Based on the predicted volume change (D0 versus D1), Section 6.1.1 of WCAP-18482-P concludes that 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.

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] [a,c]

Revised Response to SFNB RAI 11

Following clarification discussions with the NRC, the response to RAI 11 has been revised as provided below. The following response supersedes the response provided previously.

a. There are two test series: Barseback-Studsvik and O3-SCIP II (Sections 4.1 and 4.3 of Reference 11-2). The test conditions (burnup, Ramp Terminal Level (RTL), and hold time) for doped fuel and reference UO_2 fuel are listed in Table RAI-11-1 and Table RAI-11-2. The measurements for the incremental cladding strain are presented in part b of this response along with the predicted value. Consistent with Section A.2.4 of the PAD5 topical Report (Reference 11-1), the incremental cladding strain are presented as cladding diameter change from ramp tests.

[ ]a,c

[ ]a,c

b. [

]a,c

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] a,c

References:

11-1 WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.
11-2 WCAP-18482-P, "Westinghouse Advanced Doped Pellet Technology (ADOPT^{TM}) Fuel," May 2020.

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