**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 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.

### 3.3.4 Thermal Expansion

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*

### 3.3.5 Modulus of Elasticity

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.

### 3.3.6 Creep

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 **[**

**]**

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 ($\Delta$cal/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 $\Delta$cal/g. Test rod OS-1 failed at 38 $\Delta$cal/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$^{TM}$–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.*

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/kg$UO_2$ 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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D3 Bump Ramp Test

- 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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SCIP-II Ramp Test

- 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 **[**

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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.

<u>3.6.2.7 Fuel Pellet Overheating (Power-to-Melt)</u>

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.

<u>3.6.2.8 Pellet-Clad Interaction</u>

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. **[**

**]**

**[**

**]**

*Decay Heat*

**[**

**]**

*Clad Deformation-Burst Strain, Relocation*

**[**

**]**

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 **[**

**]**

**[**

**]**

*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). **[**

**]**

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. **[** **]**

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 **[ ]**

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 **[**

**]**

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


1. Letter from Z. Harper, Westinghouse, to NRC Document Control Desk (DCD), “Submittal of Topical Report ‘WCAP-18482-P, Revision 0, ‘Westinghouse Advanced Doped Pellet Technology (ADOPT$^{TM}$ Fuel)’,” LTR-NRC-20-33, May 8, 2020 (ADAMS Accession No. ML20132A014).

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. **[**

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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:<br><br>[<br>] | 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:<br><br>[ ] | 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:<br><br>[ ] | 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:<br><br>[ ] | 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:<br><br>**[**<br>**]** | 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:<br><br>**[**<br>**]** | 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:<br><br>"…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:<br><br>"…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/kg$UO_2$" | Comment not accepted.<br><br>Text, including BU definition, MWd/kgU, consistent with TR (See Table 4-6). |
| 19 | 18 | Proprietary Markings | Please mark proprietary as shown:<br><br>"…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. |

**<u>Table: Resolution of comments (Continued)</u>**

| Draft SE Page No. | Line No. | Comment Type | Westinghouse Suggested Revision | NRC Resolution |
|---|---|---|---|---|
| 20 | 4 | Editorial | [ ]<br>[ ] | 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:<br>[ ] | 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:<br><br>[ ] | 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:<br><br>[ ] | 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:<br>[ ] | 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:<br>[ ] | 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:<br>[<br>] | 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:<br><br>[ ] | 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.<br><br>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:<br><br>[<br>]| 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:<br><br>"...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:<br><br>"…found the **[**<br><br>**]** 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<br>33 | 46-47<br>1-2 | Proprietary Markings | Please mark proprietary as shown:<br><br>**[**<br>**]** | 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:<br><br>**[**<br>**]** | 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.<br><br>Suggested wording: "With or without annular pellets and application of ZrB2 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<br>40 | 43<br>3, 15 | Editorial | Suggest adding “(Nonpublicly available, Proprietary).” to end of Reference 35, 36, and 39. | Comment acceptable. Editorial change made. |