Enclosure 3

Comments and Proprietary Markings on the Draft Safety Evaluation for Westinghouse Topical Report WCAP-18482-P/NP, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel"

(Non-Proprietary)

(49 pages including this cover page)

April 2022

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

Westinghouse has reviewed the NRCs Draft Safety Evaluation (SE) for WCAP-18482-P/NP and offers the following editorial comments, and proprietary markings. The proprietary markings are highlighted in the attached mark-up of the Draft SE.

Please consider the following general comment:
- Ensure the chemical formulas are consistent with subscripts (e.g., UO_2 vs. UO2)

Comment No.	Page No.	Line No.	Comment Type	Suggested Rev.	NRC Disposition
1	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	
2	1	27-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
3	1	37	Editorial	Suggested wording: "The ADOPT fuel is intended to be used…"	
4	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"	
5	5	6	Editorial	Section title should read CHARACTERIZATION OF ADOPT FUEL BEHAVIOR	
6	5	17-20	Proprietary Markings	Please mark proprietary as shown: [$]^{b,c}$	
7	5	24 & 29	Proprietary Markings	Suggest removing proprietary marking, since it is not needed.	
8	5	26-32	Proprietary Markings	Please mark proprietary as shown: [$]^{b,c}$	
9	5	35-36	Proprietary Markings	Please mark proprietary as shown: [$]^{b,c}$	
10	5	38-43	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	

11	7	44	Clarification	Please change "ASTM E831-19" to "ASTM E831-86," which was the standard used at the time of testing.	
12	8	12	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
13	9	45	Editorial	Suggest changing "FRG" to "FGR"	
14	10	26	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
15	10	39-44	Proprietary Markings	Please mark proprietary as shown: [ ]a,c	
16	11	11-14	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
17	11	41-45	Proprietary Markings	Please mark proprietary as shown: [ ]b,c	
18	12	7-8	Proprietary Markings	Please mark proprietary as shown: [ ]b,c	
19	14	32-34	Proprietary Markings	For consistency, please mark proprietary as shown: "…staff reviewed [ ]a,c "	
20	15	1-3	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
21	16	1-5	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
22	18	12	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
23	18	18	Proprietary Markings	Please mark proprietary as shown: "…the [ ]a,c peak rod…"	
24	18	45	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
25	19	2	Technical clarification	Please change units from 'MWd/kgU' to 'MWd/kgUO$_2$'	
26	19	18	Proprietary Markings	Please mark proprietary as shown: "…62 MWd/kgU [ ]a,c as Westinghouse..."	

27	20	4	Editorial	Section 3.3.2 of the SE shows there is negligible difference in the thermal diffusivity of ADOPT and UO_2. It is suggested that the SER references the TR. "As discussed in Section 6.6.1 of WCAP-18482-P/WCAP-18482-NP"	
28	20	4-18	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
29	20	42-44	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
30	21	10	Editorial	Please change 'including' to 'excluding'	
31	21	40-42	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
32	22	9-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
33	22	19-25	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
34	22	28-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
35	22	30-31	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
36	22	35-40	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
37	23	9-14	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
38	23	27-29	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
39	24	1-4	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
40	24	7-15	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	

41	24	19-22	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
42	25	40	Editorial	Criterion 'I' appears to be typo. Suggest removing 'I'	
43	26	7	Reference	Please use Reference 1 or 5.	
44	26	21	Editorial	Please replace "Section 5.2.2" with " Section 6.1.2.5"	
45	27	6	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
46	27	8-10	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
47	28	25-29	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
48	28	37	Editorial	Please change section 3.6.3.1 title as suggested	
49	28	37, 42	Editorial	Please change "Full Spectrum" to "FULL SPECTRUM"	
50	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."	
51	29	10-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
52	29	14-20	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
53	29	17	Editorial	Suggest removing proprietary marking, since it is not needed.	
54	29	24-28	Proprietary Markings	Please mark proprietary as shown: [ ] a,c	
55	29	27	Proprietary Markings	Suggest removing proprietary marking, since it is not needed.	
56	29	32-39	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	

57	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 anlyases. The NRC staff determined that application of FSLOCA methodology and NOTRUMP EM on ADOPT fuel is acceptable."	
58	29	46	Editorial	Please change "Full Spectrum" to "FULL SPECTRUM"	
59	30	7	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
60	30	18-30	Proprietary Markings	Please mark proprietary as shown: [$]^{a,c}$	
61	30	19	Editorial	Please change 'WCOBRA/TRACPF2' to 'WCOBRA/TRAC-TF2' and delete proprietary bracket	
62	30	46-48	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
63	31	1-6	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
64	31	3	Editorial	Please change 'Section 3.2.1' to 'Section 4.2.2, 4.2.3, and 5.4.'	
65	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."	
66	31	16	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
67	31	21, 23	Proprietary Markings	Please remove proprietary brackets from the room temperature theoretical density of standard UO_2 fuel	
68	31	24	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
69	31	27	Editorial	Please add period at end of sentence	
70	31	31	Editorial	Please change section '3.3.4' to Section '3.3.2'	

71	31	43	Editorial	Please change 'Reference 33' to 'Reference 34'	
72	32	16-17	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
73	32	26	Proprietary Markings	Please mark proprietary as shown: "...there is [ ]^{a,c} "	
74	32	26	Editorial	Suggest removing proprietary marking, since it is not needed.	
75	32	29-30	Proprietary Markings	Please mark proprietary as shown: "…found the [ ]^{a,c} for…"	
76	32 33	46-47 1-2	Proprietary Markings	Please mark proprietary as shown: [ ]^{a,c}	
77	33	7-8	Proprietary Markings	Please mark proprietary as shown: [ ]^{a,c}	
78	34	44	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
79	35	2	Editorial	Suggest deleting reference 43 and leaving this statement generic to be consistent with Section 2.24 and 6.3 of the TR.	
80	35	9	Editorial	Please delete 'WNG' from 'WNGWSSV'	
81	35	20	Editorial	Suggested wording: "…approved CHF correlations such as the ones listed above…"	
82	36	6	Technical clarification	Please add IFBA and annular blanket usage language for consistency with the TR. Suggested wording: "With or without annular pellets and application of ZrB_2 integral fuel burnable absorber (IFBA) coating consistent with the defined IFBA parameters in applicable NRC-approved fuel performance or product topical reports."	
83	36	7	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	

84	36	8	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
85	36	9	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
86	36	9	Technical clarification	Please add 'with correction' to the end of the sentence, for consistency with the TR.	
87	36	10-11	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
88	36	11, 13	Technical clarification	Please use 'µg/gU' units in lieu of 'ppm' when referring to the alumina and chromia content	
89	36	12-13	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
90	36	18	Editorial	Please delete 'P' from 'WCAP-18482-P-NP'	
91	36	21-22	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
92	36	26-31	Proprietary Markings	Westinghouse agrees with the staff's proprietary markings.	
93	38	37	Technical clarification	Suggest updating "ASTM E831-19" to "ASTM E831-86" for consistency with Comment 11	
94	39	8-10	Editorial	Please confirm accuracy of Reference 27. [ ]a,c	
95	39	30,31	Editorial	Please change "Full Spectrum Local" to "FULL SPECTRUM LOCA"	
96	39 40	43 3, 15	Editorial	Suggest adding '(Non-publicly available, Proprietary).' to end of Reference 35, 36, and 39.	
97	40	30-32	Editorial	Suggest deleting Reference 43 to be consistent with comment no. 79	

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

1   **U. S. NUCLEAR REGULATORY COMMISSION**
2
3   **DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION**
4
5   **FOR THE WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT**
6
7   **WCAP-18482-P/WCAP-18482-NP, REVISION 0, "WESTINGHOUSE ADVANCED DOPED**
8
9   **PELLET TECHNOLOGY (ADOPT™) FUEL"**
10
11   **DOCKET NO. 99902038 EPID L-2020-TOP-0025**
12
13
14   1.0   **INTRODUCTION AND BACKGROUND**
15
16   By letter dated May 8, 2020 (Agencywide Documents Access and Management System
17   (ADAMS) Accession No. ML20132A014), Westinghouse Electric Company (Westinghouse)
18   submitted Topical Report (TR) WCAP-18482-P/WCAP-18482-NP, Revision 0, "Westinghouse
19   Advanced Doped Pellet Technology (ADOPT™) Fuel," (Ref. 1) to the U.S. Nuclear Regulatory
20   Commission (NRC) for review and approval. By letters dated March 19, 2021 (Ref. 2), June 20,
21   2021 (Ref. 3), and November 12, 2021 (Ref. 4), Westinghouse supplemented the TR with
22   responses to the NRC staff's requests for additional information (RAI). Westinghouse proposes
23   ADOPT fuel as a direct replacement for standard uranium dioxide (UO_2) fuel. Westinghouse
24   asserts that ADOPT fuel provides enhanced fuel pellet properties to enable higher burnup and
25   improved accident tolerance. ADOPT fuel is a standard UO_2 pellet doped with small amounts of
26   chromium oxide Cr_2O_3 (chromia) and aluminum oxide Al_2O_3 (alumina). Westinghouse
27   requested the chromia content in the range of [   ]^{a,c} and alumina in the range of [
28   ]^{a,c} However, Westinghouse has indicated that the ADOPT fuel has a nominal value
29   of [   ]^{a,c} chromia and of [   ]^{a,c} alumina as additive content.   The additives purportedly
30   facilitate greater densification and diffusion during sintering, resulting in a higher density and an
31   enlarged grain size as compared to undoped UO_2.
32
33   The purpose of the TR is to provide a detailed description of the ADOPT fuel pellets and to
34   describe and characterize the material properties through a review of past operating history and
35   qualification data. This safety evaluation (SE) reviews generic qualifications of the ADOPT fuel
36   material, its properties and performance, and the modeling approach in safety analysis
37   methods, as presented in the TR. The ADOPT fuel ~~in~~ is intended to be used with all current NRC-
38   licensed and approved Westinghouse and Combustion Engineering (CE) pressurized water
39   reactors (PWRs). The ADOPT fuel design is intended to be used with two NRC-approved
40   zirconium-based cladding materials - ZIRLO® and Optimized ZIRLO™ - and fuel enrichments
41   up to 5 percent.
42
43   This NRC staff review focused on the manner in which additives affected the following major
44   material properties (Section 3, Ref. 1): microstructure, melting temperature, theoretical density,
45   thermal expansion, thermal diffusivity and conductivity, specific heat, grain size and growth,
46   creep, yield stress, modulus of elasticity, strain hardening coefficient and tangent modulus,
47   plastic Poisson's ratio, and rim structure effects. The NRC staff also reviewed the following
48   in-reactor performance concerns (Section 5, Ref. 1) for the use of additive fuel: impact of

Enclosure

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

2

1   additives on fuel oxidation resulting in fuel washout when exposed to primary coolant water in
2   the event of fuel failure; impact of additives on fuel melting limits; impact of the additive fuel on
3   reactivity insertion accident (RIA) thresholds; impact of the additive fuel on in-reactor
4   densification; impact of the additive fuel on rod growth; impact of the additive fuel on fission gas
5   release (FGR); impact of the additive fuel on fuel fragmentation, relocation, and dispersal
6   (FFRD); and impact of additive fuel on accident source terms. The NRC staff further reviewed
7   the in-reactor (irradiation) data and used it to examine the performance of additive fuel
8   (Section 4.0 of the TR). The licensing criteria assessment (Section 6.0 of the TR) describes
9   performance of the additive fuel during steady state and anticipated operating occurrences
10   (AOOs) using fuel rod design criteria, safety analyses requirements, and applicable thermal-
11   hydraulic design requirements.
12
13   In the TR, Westinghouse uses the most recent NRC-approved fuel performance methodology,
14   as documented in WCAP-17642-P-A (PAD5), to model the mechanical performance of ADOPT
15   fuel (Ref. 5). Key differences in ADOPT fuel from standard UO_2 fuel are higher density and
16   lower fuel densification, both of which Westinghouse modeled via modification to the existing
17   PAD5 input variables. The acceptability of this modeling is discussed below.
18
19   Section 2.0 of the SE describes the regulatory basis for the SE. Section 3.0 and its sub-
20   sections contain technical evaluation of ADOPT fuel: Section 3.1 of the SE focuses on ADOPT
21   fuel definition; Section 3.2 describes the characterization of ADOPT fuel properties; Section 3.3
22   describes ADOPT fuel thermal and mechanical properties; Section 3.4 describes irradiation
23   programs and experience with ADOPT fuel; Section 3.5 discusses characterization of ADOPT
24   fuel behavior; Section 3.6 briefly describes ADOPT fuel licensing criteria assessment. Section
25   4.0 of this SE lists the limitations and conditions (L&Cs).
26
27   **2.0   REGULATORY EVALUATION**
28
29   Regulatory guidance for the review of fuel system designs and adherence to General Design
30   Criteria (GDC)-10, "Reactor design," GDC-27, "Combined reactivity control systems capability,"
31   GDC-28, "Reactivity limits," and GDC-35, "Emergency core cooling," is provided in NUREG-
32   0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power
33   Plants – LWR Edition," (SRP) (Ref. 6), Section 4.2, "Fuel System Design (Ref. 7)." SRP Section
34   4.3, "Nuclear Design" (Ref. 8) and Section 4.4, "Thermal and Hydraulic Design" (Ref. 9), are
35   also pertinent to the review of fuel systems.
36
37   SRP Section 4.2 acceptance criteria are based on meeting the requirements of GDC-10 in
38   Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.
39
40   GDC 10 states:
41
42   *The reactor core and associated coolant, control, and protection systems shall be*
43   *designed with the appropriate margin to assure that specified acceptable fuel*
44   *design limits are not exceeded during any condition of normal operation,*
45   *including the effects of anticipated operational occurrences.*
46
47   GDC 10 establishes specified acceptable fuel design limits to ensure that the fuel is "not
48   damaged." That means that fuel rods do not fail, fuel system dimensions remain within

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

3

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.

4

1   In Section 2.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse provides a
2   roadmap of the TR contents to applicable regulatory guidance, including SRP 4.2. This
3   information is provided to assist the reader and does not require NRC review.
4
5   Section 2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes how Westinghouse will
6   expand the limits of applicability for existing NRC-approved TRs to include ADOPT fuel
7   properties and performance, and then defines how licensees will apply these expanded TRs.
8   Justification for the expansion of the NRC-approved TRs is addressed in Section 3.0 of this SE.
9   Section 2.2.2 of the TR concludes that upon approval of this TR, WCAP-12488-A and WCAP-
10   12488-A, Addendum 1-A, which define the Westinghouse fuel criteria evaluation process
11   (FCEP), will be applicable to the fuel designs containing ADOPT fuel at Westinghouse plants.
12   Approval for Westinghouse's FCEP is limited to nuclear fuel in Westinghouse plants; however,
13   extending the approval of FCEP to CE plants is beyond the scope of this review, because
14   approval for Westinghouse FCEP is limited to nuclear fuel in Westinghouse plants. Thus, with
15   the exception of the FCEP, the NRC staff finds the process for implementing ADOPT fuel via
16   expanded approval of existing TRs acceptable. As noted above, the NRC staff will determine
17   whether this TR can expand the relevant TRs' applicability to include doped pellets below.
18   Approval for Westinghouse FCEP is limited to nuclear fuel in Westinghouse plants.
19
20   Section 2.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes anticipated licensee
21   actions to implement ADOPT fuel. Westinghouse states that a license amendment request
22   (LAR) would be required and lists appropriate content for such a licensing action. The NRC
23   staff agrees that a LAR will be required to implement ADOPT fuel. However, given many
24   variants in plants' licensing bases, it is difficult to accurately define the necessary content for
25   future LARs.
26
27   **3.0   TECHNICAL EVALUATION**
28
29   Westinghouse has developed ADOPT fuel technology to improve performance and enhance the
30   accident tolerance of UO_2 fuel pellets. ADOPT fuel is a modified UO_2 pellet doped with small
31   amounts of chromia and alumina. The additives are expected to facilitate greater densification
32   and diffusion during sintering, resulting in a higher density and an enlarged grain size as
33   compared to undoped UO_2. This review focused on the impact of the additive dopants on major
34   material properties and in-reactor performance.
35
36   **3.1   ADOPT FUEL DEFINITION**
37
38   Section 1.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, provides the ADOPT fuel
39   definition. This definition consists of nominal dopant concentrations, nominal pellet density, and
40   a range in grain size. The NRC staff's approval of ADOPT fuel is based upon this definition,
41   allowable ranges in composition, including dopant concentrations, and microstructure
42   (described later in this SE), and documented fuel properties and performance.
43
44   Section 1.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes ADOPT's extensive in-
45   reactor operating experience, including full batch implementation in European boiling water
46   reactors (BWRs).
47

> Suggest removing because it appears redundant

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

5

1 Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, defines the range of applicability
2 for this TR. Although Westinghouse requested these limits of applicability, the NRC staff may
3 not find all these limits acceptable. Based on the assessments described in this document, the
4 NRC staff will either restate the limits or establish new limits in Section 4.0 of this SE.
5
6 **3.2   CHARATERIZATION OF ADOPT FUEL PROPERTIES**

CHARACTERIZATION

7
8 ADOPT Fuel Additives and Microstructure
9
10 Section 3.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the impact of the
11 alumina and chromia dopants on densification and grain growth during fuel sintering and,
12 ultimately, on the pellet microstructure. It also describes dopant solubility and residence within
13 the pellet microstructure. In request for information (RAI) 4 (Ref. 2), the NRC staff requested a
14 clarification regarding the distribution and re-distribution of the dopants in fresh fuel, as well as
15 during irradiation. Westinghouse responded that for fresh ADOPT fuel, it measured the radial
16 concentration of Aluminum and Chromium in an unirradiated pellet, using wavelength dispersive
17 spectrometry (WDS). [
18
19
20   ]b,c The NRC
21 staff agrees that the higher vapor pressure of chromia is the reason for redistribution of chromia
22 in ADOPT fuel.

Prop marking not needed here

23
24 [In order to further investigate the distribution of chromia during power ramp, a section of the
25 ADOPT rod was ramp tested to a terminal power of 30 kilowatts per meter (kW/m). Following
26 the ramp, laser ablation spectrometry was performed on the segment. [
27
28
29
30
31
32   ]b,c
33
34 Westinghouse performed electron probe microanalysis (EPMA) on two irradiated ADOPT rods
35 to investigate the dopant migration under steady-state conditions and found that [
36   ]b,c
37
38 In summary, [
39
40
41
42
43   ]b,c The NRC staff reviewed the irradiation tests and the results of EPMA, laser ablation,
44 and WDS examinations and found that the results are consistent and acceptable.
45
46 Based upon the information provided in the TR and response to RAI 4, the NRC staff finds that
47 Westinghouse understands the impact of alumina and chromia dopants on the pellet's
48 microstructure. Understanding microstructure, and the evolution of microstructure under

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

6

1   irradiation, is the first step to characterizing the material properties and performance of ADOPT
2   fuel. These are the topics of the next sections.
3
4   **3.3.   THERMAL AND MECHANICAL PROPERTIES**
5
6   *Thermal Properties*
7
8   3.3.1   Specific Heat
9
10 Specific heat (C_p) capacity is important in determining the stored energy for use in transient
11 analysis and loss-of-coolant accident (LOCA) analysis, and in determining thermal conductivity.
12 Section 3.2.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of
13 specific heats of both ADOPT fuel and standard UO_2 fuel pellets.
14
15 Specific heat was determined using a differential scanning calorimeter at the Institute for
16 Transuranic Elements (ITU) in Karlsruhe, Germany. The tests were performed with flowing
17 argon gas at the rate of 0.1 liters per minute (l/min) and a temperature ramp rate of 25 Kelvin
18 per minute (K/min). The referral material used in this measurement was sapphire. The
19 temperature range was 400 – 1400 K, the range at which this measurement technique is most
20 accurate. Two unirradiated ADOPT fuel samples were analyzed and compared with two pure,
21 unirradiated standard UO_2 samples (Figure 3-7 of Ref. 1). The measurements revealed no
22 appreciable difference between the specific heats of ADOPT fuel and reference UO_2 pellets for
23 the temperature range up to 1200°C as per Figure 3-7 in the TR.
24
25 The NRC staff reviewed the specific heat measurement and concluded that there is no
26 appreciable difference between specific heats of standard UO_2 fuel and ADOPT fuel as
27 illustrated in Figure 3-7 of the TR.
28
29 3.3.2   Thermal Diffusivity and Thermal Conductivity
30
31 Thermal conductivity is an important material property that is used to determine the
32 temperature distribution in the fuel rod. Thermal conductivity is determined indirectly by
33 measuring the thermal diffusivity using the laser flash technique. Section 3.2.2 of WCAP-
34 18482-P/WCAP-18482-NP, Revision 0, describes measurements of thermal diffusivity on both
35 ADOPT fuel and standard UO_2 fuel pellets. Thermal diffusivity is measured using the laser flash
36 technique which consists of irradiating the sample material surface with a laser pulse and
37 monitoring the temperature rise of the material using a photovoltaic infra-red detector.
38
39 The thermal diffusivity, α, is calculated from $\alpha \; = \; k \times \dfrac{L^2}{t_{1/2}}$
40
41 where k is a constant, L is the thickness of the specimen, and $t_{1/2}$ is the time for the back face of
42 the sample to reach half of its maximum temperature rise in seconds.

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

7

1   Thermal conductivity, $\lambda = \alpha \times \rho \times C_p$,
2
3   where ρ is the density and C_p is its specific heat.
4
5   Thermal diffusivity of unirradiated ADOPT fuel was measured at the KTH Royal Institute of
6   Technology in 1999 and was compared with an unirradiated standard UO_2 sample.
7   Measurements were taken between 20°C and 1400°C in approximately 100°C increments
8   during heating while correcting for thermal expansion of the samples. Figure 3-8 of the TR
9   compares thermal diffusivity of ADOPT fuel and UO_2 fuel and it shows no appreciable difference
10   between the thermal diffusivities of ADOPT fuel and UO_2 fuel.
11
12   Based upon these measurements, Westinghouse concludes that there is no appreciable
13   difference in the thermal diffusivity of standard UO_2 fuel and ADOPT fuel. The NRC staff
14   reviewed the above information and based on this information; the NRC staff finds this
15   conclusion acceptable.
16
17   3.3.3   Melting Temperature
18
19   Fuel melting temperature is a safety limit defined in a plant's Technical Specifications, an
20   important property in safety analyses, and evaluated against accident analyses. Section 3.2.3
21   of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes measurements of fuel melting
22   temperature on both ADOPT fuel and standard UO2 fuel pellets simultaneously using a
23   pyrometer and a spectrometer, providing both the true temperature and the spectral emissivity
24   function of a specimen. The emissivity was used to convert the temperature brightness into the
25   true value. The reliability of the procedure was confirmed by the melting point measurement on
26   stoichiometric reactor-grade UO2, which was within 0.5 percent of the recommended value.
27   Figure 3-9 of the TR shows melting temperatures of ADOPT fuel and UO2 fuel during melting of
28   the two specimens using laser-pulse. This figure shows that there is no appreciable difference
29   between the measured melting temperature, 3122 ± 7 K, and the measured value of the
30   reference UO2.
31
32   Based upon review of these measurements, the NRC staff concludes that there is no
33   appreciable difference in the melting temperature of standard UO2 fuel and ADOPT fuel.
34
35   3.3.4   Thermal Expansion
36
37   Thermal expansion changes a fuel's volume, and, thus, density at a given temperature relative
38   to a standard temperature. Thermal expansion is an important consideration in pellet cladding
39   mechanical interaction (PCMI). Section 3.2.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0,
40   describes measurements of thermal expansion on both ADOPT fuel and standard UO_2 fuel
41   pellets.
42   ASTM-E831-86
43   Thermal expansion of unirradiated ADOPT fuel samples and unirradiated UO_2 were measured
44   at the ITU in Karlsruhe, Germany according to ASTM E831-19, the "Standard Test Method for
45   Linear Thermal Expansion of Solid Materials by Thermomechanical Analysis" (Ref. 22). Data
46   was collected over a temperature range from 20°C - 1490°C with a heating rate of 5°C/min.
47   Figure 3-10 of TR shows thermal expansion of ADOPT fuel and UO_2 fuel and shows no
48   appreciable difference between thermal expansions of the two specimens.

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

8

1 Based upon the review of these measurements and the results, the NRC staff concludes that
2 there is no appreciable difference in the thermal expansion of standard UO_2 fuel and ADOPT
3 fuel.
4
5 *Mechanical Properties*
6
7 3.3.5   Modulus of Elasticity
8
9 Westinghouse reports that the impact of the discussed dopants on the elastic moduli can be
10 determined using the same procedure employing the rule of mixtures as the calculation of
11 theoretical density (TD) in Section 3.1.4 of the TR. Based on that calculation, the addition of the
12 specified nominal amount chromia **[   ]a,c** and alumina **[   ]a,c** to UO_2 will have no
13 significant effect on the elastic properties of ADOPT fuel compared to standard UO_2 fuel. Fuel
14 temperature has a much more significant impact on the elastic moduli.
15
16 The NRC staff reviewed the impact of dopants on the elastic moduli of ADOPT fuel and based
17 on the above-described calculation, determined that the dopant has no significant effect on the
18 elastic moduli.
19
20 3.3.6   Creep
21
22 Section 3.3.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes creep and hardening
23 testing performed on both ADOPT fuel and standard UO_2 fuel pellets. In the creep tests,
24 ADOPT fuel and reference UO_2 were tested at three different temperatures (1300°C, 1500°C,
25 and 1700°C) and three compressive stresses (30 megapascals (MPa), 45 MPa, and 60 MPa).
26 The measurements revealed a classical creep curve with a strong temperature dependency,
27 such that measured strain increases dramatically with rising temperature, and sensitivity to
28 applied stress. Figure 3-11 of TR illustrates this fact. At temperatures greater than 1500°C, the
29 ADOPT fuel exhibited higher viscoplasticity as compared to the reference UO_2 pellets. This is
30 illustrated in figure 3-12 of TR. At temperatures lower than 1300°C, there appears to be no
31 creep benefit relative to the reference UO_2. Thus, in steady-state operation, the creep behavior
32 between ADOPT fuel and standard UO_2 fuel shows no appreciable difference because radial
33 average fuel temperature remains below 1300°C.
34
35 Hardening tests were performed for ADOPT fuel and UO_2 fuel at temperatures ranging from
36 1100°C to 1700°C. At each temperature, a constant strain rate of 10%/hr or 50%/hr was
37 applied to the specimen. Hardening tests showed a strong temperature dependency to the
38 applied strain rate. Figure 3-13 in the TR shows that ADOPT fuel is more ductile than the
39 standard UO_2 fuel, which means that ADOPT fuel requires less stress than standard UO_2 fuel to
40 achieve a given strain rate. This shows more viscoplasticity capability for ADOPT fuel in the
41 strain levels of interest for pellet clad interaction (PCI). Thus, the specimen will show rate-
42 dependent inelastic behavior, meaning material will undergo unrecoverable deformations when a
43 load level is reached.
44
45 The NRC staff reviewed the details of the test process and the results of both creep and
46 hardening tests performed on standard UO_2 fuel and ADOPT fuel pellets and the staff
47 determined that the creep behavior and hardening behavior of ADOPT fuel is acceptable
48 because: (1) with regard to creep behavior, ADOPT fuel and standard UO_2 fuel have the same

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

9

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 ~~FRG~~ 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

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

10
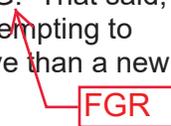
1   model attempting to credit FGR benefit, it bounds ADOPT fuel. Therefore, the NRC staff finds
2   that the use of the existing UO_2 transient FGR model for ADOPT is acceptable.
3
4   **3.5   CHARACTERIZATION OF ADOPT FUEL BEHAVIOR**
5
6   Section 5 of WCAP-18482-P/WCAP-18482-NP, Revision 0 (Ref. 1), describes the empirical
7   database used to characterize the performance of ADOPT fuel pellets during normal operations,
8   AOOs, and postulated design-basis accidents (DBAs).
9
10   3.5.1   Corrosion and Washout Characteristics
11
12   Section 5.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes: (1) high-temperature
13   furnace tests on unirradiated fuel specimens to characterize fuel oxidation performance, and (2)
14   in-pile, irradiated testing on damaged fuel rod segments to characterize erosion (washout)
15   performance. The data show enhanced fuel corrosion resistance and slightly better washout
16   performance relative to standard UO_2 fuel. Fuel corrosion and washout performance are not
17   modeled as part of any AOO or postulated DBA safety analysis. However, these performance
18   aspects are important when operating a reactor with damaged fuel (i.e., leakers). Based upon
19   these furnace tests and in-pile tests, the NRC staff finds ADOPT's oxidation and washout
20   performance acceptable and well characterized.
21
22   3.5.2   Swelling Behavior (Pellet Densification and Rod Growth)
23
24   Section 5.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes re-sintering tests
25   performed on unirradiated ADOPT fuel pellets to characterize fuel densification. Re-sintered
26   pellet densification measurements were collected and analyzed on **[   ]a,c** ADOPT fuel pellets.
27   There is a clear difference in the densification of standard UO_2 and ADOPT fuel pellets. ADOPT
28   fuel pellets exhibit less densification compared with standard UO_2. The impact of the change in
29   densification will be explicitly accounted for in fuel performance and safety analyses. Based
30   upon these re-sintering tests, the NRC staff finds ADOPT's densification behavior acceptable
31   and well characterized.
32
33   Section 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes Westinghouse's fuel
34   rod growth measurements on both BWR and PWR ADOPT fuel rods. An important impact of
35   ADOPT fuel's reduced in-reactor densification is an earlier closure of the fuel pellet-to-cladding
36   gap. After gap closure, irradiation-induced fuel swelling will influence fuel rod growth. The
37   empirical database clearly shows an increase in fuel rod growth compared to standard UO_2 fuel
38   rods. In response to an RAI regarding the continued applicability of the PAD5 upper bound
39   (UB) growth model (RAI 7a, Ref. 2), **[**
40
41
42
43
44   **]a,c**
45
46   In a revised response to RAI 7 (Ref. 4), Westinghouse proposed an additive term applied to the
47   standard UO_2 fuel rod growth model to account for the increased growth exhibited by the
48   ADOPT fuel rods. The revised model provides an improved upper and lower bound prediction

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

11

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.

3.5.3   Steady State FGR Database

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 **[**

]b,c

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. **[**

]b,c 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.

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

12

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. [

]^{a,c}

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

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

13

clarification call if Westinghouse planned on (1) incorporating RG 1.236 guidance as part of the ADOPT methodology, and (2) justifying the use of this guidance for ADOPT fuel. Westinghouse would not commit to incorporating this guidance as part of a common CRE methodology for ADOPT fuel.

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

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

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

- High-Temperature Cladding Failure Threshold (Section C.3.1 of RG 1.236)

    - Because ductile failure depends on cladding temperature and differential pressure (i.e., RIP minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (MPa).

- PCMI Cladding Failure Threshold (Section C.3.2 of RG 1.236)

    - Because fuel cladding ductility is sensitive to hydrogen content, zirconium hydride orientation, and initial temperature, separate PCMI failure curves are provided for fully recrystallized annealed (RXA) and stress relief annealed (SRA) cladding types at both low initial cladding temperature conditions (i.e., below 500°F down to BWR cold startup) and high initial cladding temperature conditions (i.e., at or above 500°F). The failure threshold curves are expressed as change in radial average fuel enthalpy (Δcal/g) versus fuel cladding excess hydrogen content.

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

    - Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions. Fuel melt calculations are

14

1 sensitive to burnup characteristics, prompt pulse characteristics, and local fuel
2 melting temperature.
3
4 With respect to high-temperature and molten fuel cladding failure, ADOPT fuel properties have
5 a negligible impact on the ability of analytical models to predict local burnup, initial RIP, local
6 transient power, and local transient fuel temperature. Sections 2 and 6 of WCAP-18482-P/
7 WCAP-18482-NP, Revision 0, describe the applicability of existing Westinghouse nuclear
8 design and fuel thermal-mechanical design models to ADOPT fuel. Based on the ability of
9 analytical models to predict these parameters, the NRC staff finds that these cladding failure
10 thresholds are applicable to ADOPT fuel designs.
11
12 With respect to PCMI cladding failure, the performance of ADOPT fuel has the potential to
13 impact important initial fuel rod conditions (e.g., fuel-to-cladding gap size) and local transient
14 conditions (e.g., fuel thermal expansion, cladding strain). With respect to ADOPT fuel's
15 enhanced creep performance, Westinghouse states that diffusion driven processes do not have
16 time to occur during the short RIA pulse, and so the greater high-temperature fuel creep does
17 not have time to reduce the clad stress. Based on a review of the information provided, the
18 NRC staff agrees that ADOPT's enhanced fuel creep will not impact fuel performance during the
19 postulated CRE accident.
20
21 Sections 3 and 4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describe the impact of the
22 dopant addition on fuel pellet densification and pellet-to-cladding gap size. A smaller initial gap
23 size would have a detrimental impact on RIA PCMI performance since it allows less fuel
24 volumetric expansion prior to cladding contact. But this effect disappears when the fuel pellet
25 contacts the cladding due to irradiation-induced fuel swelling (and cladding creep down) at
26 approximately middle-of-life.
27
28 While the implementation of the PCMI cladding failure curves requires analytical models
29 capable of predicting both local fuel enthalpy and cladding hydrogen content, the development
30 of the failure thresholds was based on in-pile prompt pulse testing of irradiated fuel rod
31 segments. To date, only a single in-pile prompt pulse test has been conducted on ADOPT fuel.
32 The NRC staff reviewed **[**
33
34   **]**[a,c] The paper details two Nuclear Safety
35 Research Reactor (NSRR) prompt power tests conducted on chromia doped fuel and ADOPT
36 fuel. Fuel specimen and testing specifications are shown below (extracted from the technical
37 paper).
38
39

15

1   [

2
3   ]
4
5   Test rod LS-4 did not fail with a maximum increase in fuel enthalpy of 133 ∆cal/g. Test rod OS-
6   1 failed at 38 ∆cal/g, which is below all previous test failures for rods close in burnup and
7   hydrogen content. The paper concluded with the following observations:
8
9
10   *The pre- and post-test examinations suggested that one of the reasons of the*
11   *lower failure limit may be the effect of the hydrides radially oriented and*
12   *precipitated more densely in the specific angle range in the cladding tube.*
13   *However, since the possible contribution of ADOPT^{TM}–pellets specific effects*
14   *cannot be ruled out at the present, further investigation is needed on fuel pellet*
15   *behavior under both normal-operation and pulse-irradiation conditions.*
16
17
18   Figure 5-1 shows the OS-1 test plotted against the RG 1.236 RXA PCMI cladding failure
19   threshold presented in Reference 1 and the supporting empirical database. Note that OS-1 was
20   BWR Zry-2 cladding with a liner. Examination of the figure reveals that the OS-1 failure
21   enthalpy is below the RG 1.236 failure threshold curve.
22
23
24

16

1   [

2
3
4
5   ]
6
7 Japan Atomic Energy Agency (JAEA) investigated the failure of OS-1 and made the following
8 observations (Ref. 27):

9
10   *The morphology of the hydrides precipitated in the fuel cladding of OS-1 was*
11   *investigated by metallography and compared with previous results obtained in JAEA in*
12   *connection focusing fuel failure limit. It was suggested that the observed lower limit of*
13   *fuel failure was related to the amount and length of the hydride precipitated along the*
14   *radial direction of cladding.*

15
16 Based on the performance of OS-1, the NRC staff had concerns that ADOPT fuel may
17 (1) introduce differences in cladding zirconium hydride morphology, and (2) behave differently
18 under prompt pulse conditions. In response to a request to provide evidence that differences
19 between standard UO_2 fuel and ADOPT fuel performance (i.e., steady-state swelling, PCMI) will
20 not introduce differences in cladding properties and microstructure (RAI 9a, Ref. 3),
21 Westinghouse provided micrographs showing the hydrogen morphology of irradiated fuel rod
22 cladding for both standard UO_2 fuel and ADOPT fuel. Westinghouse states that the images
23 appear to show a consistent ratio of radial to tangential hydrides around the rod. Therefore,
24 ADOPT fuel's increased density, and earlier-in-life pellet-clad contact, does not appear to
25 contribute any excessive radial hydride reorientation. Westinghouse also provided results from
26 expansion due to compression testing on irradiated fuel cladding segments which showed that

17

1   both the standard UO$_2$ fuel and ADOPT fuel cladding survived relatively large strains at PWR
2   operating temperatures.
3
4   In a response to an RAI to characterize the fuel thermal expansion of ADOPT fuel or similar
5   large grain fuel pellets (RAI 9b, Ref. 3), Westinghouse summarized the tests results from NSRR
6   Tests OI-10 (28-micron grain size), MR-1 (40-micron grain size), and LS-4 (50–60-micron grain
7   size). Based on the measured residual cladding strain in OI-10 and MR-1, Westinghouse
8   claimed that the data suggests that pellet expansion was largely generated by solid thermal
9   expansion, and that the fission gas swelling associated with the enhanced fission gas retention
10   of large-grained fuels (like the ADOPT fuel pellet) was minimal. And while the larger grained
11   LS-4 did experience additional cladding strain, it was not unexpected given the larger power
12   pulse. Based on this limited data set, Westinghouse concluded that there is no information to
13   suggest that additives or large grains have a negative impact on the fuel rod's susceptibility to
14   PCMI failure.
15
16   Based on the information described above, the NRC staff accepts the premise that the low
17   failure enthalpy of Test OS-1 was likely caused by the cladding hydride morphology and not an
18   intrinsic effect of the dopant (or large grain structure) on fuel thermal expansion characteristics.
19   Given that the empirical database supporting the PCMI failure curves consists of a wide range
20   of fuel designs and operating experience, including the three tests on large grain specimens
21   identified above, the NRC staff finds that the RG 1.236 PCMI cladding failure thresholds are
22   applicable to ADOPT fuel designs.
23
24   Sections 4 and 5 of RG 1.236 are independent of fuel design and hence would be applicable to
25   ADOPT fuel. Section 6 of RG 1.236 provides guidance associated with maintaining a known
26   geometry amenable to continued core cooling. The criteria consist of an empirical-based fuel
27   rod fracture limit on peak fuel enthalpy and an analytical limit on fuel melting. Westinghouse
28   states that the minor doping additions will not affect the fissile isotope consumption of U-235
29   and production of Pu-239. Thus, there will be no impact on the local, transient power, and fuel
30   temperatures experienced during the prompt pulse. In addition, the minor doping additions
31   have a negligible impact on fuel thermal conductivity and fuel melting temperature. Thus, the
32   margin to fuel melting is not impacted. As described earlier, the Westinghouse core physics,
33   and fuel rod models have been shown to be applicable to ADOPT fuel. Hence, predictions of
34   deposited energy and fuel temperatures will account for ADOPT fuel properties. Based on
35   these considerations, the NRC staff finds the coolable geometry criteria applicable to ADOPT
36   fuel designs.
37
38   Based on the discussion above, the NRC staff found that RG 1.236 is applicable to the ADOPT
39   fuel designs.
40
41   3.5.6   Comparison of Doped UO$_2$ Fuel Properties and Performance
42
43   In 2018, the NRC reviewed and approved a revised set of Framatome analytical models and
44   methods for their chromia-doped UO$_2$ fuel pellet design (Ref. 26). While the proprietary
45   information in the Framatome submittal may not be used as a basis for the staff's approval of
46   Westinghouse's ADOPT fuel pellet, it is a valuable source of independent data and performance
47   trends which helped focus the staff's review.
48

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

18

1   Key fuel properties and performance aspects of the two doped UO$_2$ fuel designs were compiled
2   and compared in the NRC memorandum, "Applicability of Source Term for Accident Tolerant
3   Fuel, High Burn Up and Extended Enrichment" (Ref. 28). This direct comparison provided
4   either independent confirmation or focused attention on fuel properties and performance where
5   deviations were identified. For the latter, RAIs were developed to better explain different trends.
6
7   3.5.7   Evaluation of Fuel Burnup Limit
8
9   In Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse defined the
10 following constraint for ADOPT fuel applications:
11
12   Fuel burnup up to **[   ]^{a,c}** under the following provisions:

13   • No rod burst is predicted to occur using an NRC-approved methodology.
14   • Additional information is submitted to the NRC and approved for performance of ADOPT
15   fuel at higher burnups prior to exceeding a peak rod average burnup of 62 MWd/kgU.
16
17 Westinghouse plans on addressing the above provisions in a separate topical report. As such,
18 the **[   ]^{a,c}** peak rod average burnup limit and the associated provisions are beyond the
19 scope of the staff's review. Nevertheless, a burnup limit on ADOPT fuel has not yet been
20 established and does not simply default to that currently imposed on standard UO$_2$ fuel. The
21 bases for establishing a limitation on fuel burnup lies with the extent of in-reactor operating
22 experience and with the empirical database supporting ADOPT fuel's properties and analytical
23 models. Section 1.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, details the extensive in-
24 reactor operating experience Westinghouse has with ADOPT fuel in European commercial
25 reactors. The NRC staff has reviewed this operating experience and determined that, with
26 significant reload quantities of ADOPT fuel, this operating experience supports a fuel burnup
27 limit of 62 MWd/kgU.
28
29 Section 3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the characterization of
30 ADOPT pellets' microstructure and fuel properties. Most of these observations and material
31 property measurements were based on the unirradiated state of the ADOPT fuel pellets. With
32 respect to the empirical database supporting the burnup-dependent irradiated material
33 properties and analytical models, the NRC staff compiled the following summary:
34
35   • Barseback 2 fuel rods irradiated up to 33.5 MWd/kgU segment average burnup:
36   o Data on rod profilometry used to validate fuel swelling model.
37   o Ramp and bump test data used to validate transient FGR and fuel thermal
38   expansion models.
39   • Oskarshamn 3 fuel rods irradiated up to 60 MWd/kgU rod average burnup:
40   o Destructive PIE data used to validate FGR model and characterize pellet
41   cracking, HBS formation, cladding stress (hydride orientation).
42   • SCIP II Ramp Testing conducted on fuel rods up to 54 MWd/kgU segment burnup:
43   o Ramp test data used to validate transient FGR and fuel thermal expansion
44   models.
45   • **[   ]^{d}** Commercial irradiation of fuel rods up to 72 MWd/kgU rod average
       burnup:

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

19

1   o   Data used to validate rod growth, profilometry, and FGR models.

2   •   Halden IFA-677 irradiation on fuel segments up to 26 ~~MWd/kgU~~ average burnup:

3   o   Data used to validate fuel temperature model.   [MWd/kgUO$_2$]

4   •   Section 5.2.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the empirical

5   database used to validate the fuel rod growth. The data shows that fuel rod growth is

6   well characterized and predictable up to 62 MWd/kgU rod average burnup.

7   •   Section 5.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes the empirical

8   database used to validate the FGR model. The data shows that steady-state FGR is

9   well characterized and predictable up to 62 MWd/kgU rod average burnup.

10

11   Based upon the extent of the empirical database information supporting the irradiated material

12   properties and analytical models described above, the NRC staff finds that ADOPT fuel is

13   acceptable for use up to a rod average burnup of 62 MWd/kgU. The additional information

14   needed to support higher burnups (mentioned in Section 1.3 of WCAP-18482-P/WCAP-18482-

15   NP, Revision 0, was not made available to the staff; therefore, the NRC staff was unable to

16   reach a finding as to the performance of ADOPT fuel beyond 62 MWd/kgU. Consequently, the

17   staff can only approve WCAP-18482-P/WCAP-18482-NP, Revision 0, with a limitation and

18   condition limiting ADOPT Pellet use to 62 MWd/kgU [   ]^{a,c} as Westinghouse

19   originally requested. Although Westinghouse's original proposed limitation and conditions did

20   require NRC approval of additional information before the NRC approval permitted the use of

21   ADOPT fuel at burnups higher than 62 MWd/kgU; the NRC staff does not consider the limitation

22   and condition sufficient as originally written because it leaves ambiguity as to exactly when

23   burnups higher than 62 MWd/kgU are permissible.

24

25   If Westinghouse decides to pursue higher fuel burnup, additional information will be required to

26   remove or alter this limitation and condition in accordance with the Fuel Design Constraints

27   listed in Section 1.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0.

28

29   **3.6   LICENSING CRITERIA ASSESSMENT**

30

31   3.6.1   Steady State and AOO Analysis

32

33   *Fuel Performance Models and Methods*

34

35   Westinghouse Performance Analysis and Design Model (PAD5) (Ref. 5) is the fuel rod design

36   tool which incorporates relevant fuel performance phenomena such as fuel thermal conductivity

37   degradation with fuel burnup (TCD), and FGR and swelling at high burnup. PAD5 calculates

38   fuel performance parameters such as, cladding stress, strain, oxidation and hydriding, fuel

39   temperatures and volume changes, and RIP.

40

41   It has been shown that the differences in ADOPT fuel and standard UO$_2$ fuel properties (Section

42   3.3 of SE) are negligible. The corrosion and creep data presented in Section 5.1 of the TR

43   shows that ADOPT fuel has improved resistance against post-failure degradation and increased

44   PCI margins in comparison to undoped UO$_2$ pellets. PAD5 modeling of ADOPT fuel has shown

45   that it has higher density and lower fuel densification which are analyzed by modifying existing

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

20

1 PAD5 input variables. The lower densification of ADOPT fuel can be explicitly modeled with
2 PAD5 densification model as described in Section 5.7.1 of Reference 5.
3
4 [
5
6
7
8
9
10
11
12
13
14
15
16
17
18   ]a,c
19
20 3.6.2   Fuel Rod Design Criteria
21
22 The fuel rod design criteria ensure the fuel rods perform their intended function throughout the
23 lifetime of the fuel. Section 7.4 of WCAP-17642-P-A TR (Ref. 5) provides key criteria that
24 impact the Westinghouse fuel performance:
25
26   • Clad stress
27   • Clad strain
28   • Fuel RIP
29   • Cladding fatigue
30   • Cladding oxidation
31   • Cladding hydrogen pickup
32   • Axial growth
33   • Cladding free standing
34   • Pellet overheating
35   • Pellet-cladding interaction
36   • Interface to other Safety analysis
37
38 Among the above listed fuel rod design criteria, clad oxidation, clad hydrogen pickup, and clad
39 free standing do not depend on pellet properties/models. The impact on the affected criteria is
40 discussed in this section.
41
42 [
43
44   ]a,c All other fuel performance models do not
45 require a change for the ADOPT fuel when using PAD5 for evaluating the fuel rod design criteria
46 described in Section 7.4 of Reference 5. Figures 6-4 and 6-5 in WCAP-18482-P/WCAP-18482-
47 NP, Revision 0, show fuel centerline temperatures for high burnup and twice-burned assembly

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

21

1 of an uprated 3-loop plant with 15x15 fuel. These figures show differences in temperature, and
2 consistently lower temperatures, for ADOPT fuel than that for standard UO_2 fuel. The lower
3 ADOPT fuel temperature is due to the ADOPT fuel closing the gap slightly early due to lower
4 densification relative to standard UO_2 fuel. Early pellet-clad contact improves heat transfer from
5 the fuel thereby lowering fuel temperatures.
6
7 3.6.2.1 Clad Stress
8
9 Section 7.4.1 of Ref. 5, stipulates the criteria that the fuel rod shall not be damaged due to
10 excessive fuel clad stress. The maximum cladding stress intensities ~~including~~ PCI induced

excluding

11 stress is evaluated based on American Society of Mechanical Engineers (ASME) Boiler and
12 Pressure Vessel Code (BPVC) guidelines. Stresses in the cladding are combined to calculate a
13 maximum stress intensity which is then compared to the criteria set forth in Section 7.4.1 of Ref.
14 5. The ASME BPVC limits are designed to protect pressure vessels from unconstrained
15 deformation due to high stresses.
16
17 The NRC staff concluded that the use of this approved methodology is acceptable to
18 demonstrate compliance with the acceptance limit for clad stress.
19
20 3.6.2.2 Clad Strain
21
22 The acceptance criteria limit for clad strain is that the total tensile strain, elastic plus plastic, due
23 to uniform cylindrical fuel pellet deformation during any single Condition I or II transient shall be
24 less than 1 percent from the pre-transient value. Transient clad strain is caused by a rapid
25 thermal expansion and fission gas swelling of the fuel pellet during a short-term overpower
26 event.
27
28 Section 4.1 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes a base irradiation and
29 subsequent ramp testing program conducted on two BWR, barrier lined, segmented rods
30 containing standard UO_2 fuel (D0) and two different variants of doped UO_2 fuel (D1 and D3).
31 Ceramography (D0, D1), FGR measurements (D0, D1, D3), and fuel volume change (D0, D1)
32 are presented. Based on the predicted volume change (D0 versus D1), Section 6.1.1 of WCAP-
33 18482-P/WCAP-18482-NP, Revision 0, concludes that applying the PAD5 fission gas swelling
34 model for ADOPT fuel will predict slightly larger pellet deformation and therefore is conservative
35 to the calculated cladding diameter change for transient strain analysis.
36
37 Section 4.3 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes ramp testing performed
38 on ADOPT fuel rod segments as part of the Studsvik SCIP-II program. Two ADOPT fuel rod
39 segments (WaL, WaH) and two standard UO2 fuel rod segments (WsL, WsH) were ramp
40 tested. [
41
42   ]a,c
43
44 Based on its review of two previous fuel designs involving large grain UO2 fuel pellets from
45 other fuel vendors, the NRC staff has acquired knowledge of fuel performance differences
46 between large grain UO2 fuel pellets and standard UO2 fuel pellets. Specifically, a large
47 empirical database of ramp tests on irradiated fuel rod segments exists and demonstrates that
48 the larger grains promote an increase in incremental diametral cladding strain relative to

22

1 standard UO2 fuel pellets. Due to the limited ramp data presented in WCAP-18482-P/WCAP-
2 18482-NP, Revision 0, the staff requested additional information to demonstrate the relative
3 performance of ADOPT fuel and PAD5's ability to predict its performance. In response to RAI
4 11 (Ref. 3), Westinghouse provided measured and predicted incremental diametral cladding
5 strain for the ramp tests described in Sections 4.1 and 4.3 of WCAP-18482-P/WCAP-18482-NP,
6 Revision 0. Below are the NRC staff's observations on this information:
7
8 D1 Step Ramp Test
9   • D1 does not fully represent ADOPT fuel pellet composition [
10
11   o
12   •
13   ]a,c
14   • Difference in ramp profile between D0 and D1. Due to a malfunction at the test facility,
15   D0 rod segment held for 12 hours, whereas D1 rod segment held for 7.7 hours.
16   • After accounting for initial conditions (i.e., pre-ramp, base irradiation fuel volume), the
17   calculated fuel volume change was larger for D0 relative to D1 (Figure 4-6 of
18   WCAP-18482-P/WCAP-18482-NP, Revision 0).
19   • [
20
21   o
22   ▪
23   o
24
25   o   ]a,c
26
27 D3 Bump Ramp Test
28   • D3 does not fully represent ADOPT fuel pellets composition [
29   ]a,c
30   o D3 contains [
31   ]a,c
32   • D3 segment burnup average of 30 - 33.5 GWd/MTU, never exceeding 8 KW/ft.
33   • No cladding profilometry or fuel volume change estimates provided in WCAP-18482-P/
34   WCAP-18482-NP, Revision 0.
35   • [
36
37   o
38   o
39
40   o   ]a,c

23

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).
- [

    o

    o

                                                                        ]a,c

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). [

        ] a,c

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

24

1 [
2

3
4   ]a,c
5
6 The PAD5 predictions of the D1 and D3 ramp tests appear well behaved as they follow trends in
7 measured cladding diameter change along the axial length of the test specimen. [
8
9
10
11
12
13
14
15   ]a,c
16
17 In a response to the request to clarify Westinghouse position on the ability of PAD5 to predict
18 ramp data (RAI 11, Ref. 4), Westinghouse provided measured versus predicted cladding strain
19 for the entire ramp database, including the doped fuel data. [
20
21
22   ]a,c Based on
23 the information provided in response to the RAI 11, the NRC staff finds the PAD5 code

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

25

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   typo

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.

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

26

3.6.2.4 Clad Fatigue

Reference 1 or 5

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 2)~~. 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

Section 6.1.2.5

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 ~~5.2.2~~ 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.

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

27

1   In Section 5.2.1 of the TR, Westinghouse states that during the manufacturing process, the
2   pellets are checked to ensure they are compliant with the material specification. A re-sintering
3   test was performed for 24 hours at 1700°C to check the thermal stability, a measurement of the
4   pellet's expected densification behavior during irradiation. Westinghouse performed a
5   manufacturing analysis on all ADOPT pellets manufactured at the Westinghouse fuel facility
6   over a two-year period, totaling **[   ]a,c** ADOPT pellets tested. The normally and non-normally
7   distributed data obtained from the manufacturing analysis was analyzed using methods
8   specified in NRC RG 1.126 (Ref. 47). The results from this analysis show **[**
9
10   **]a,c**
11
12   The NRC staff reviewed the controlling processes of fuel manufacturing and found that there are
13   no axial gaps large enough to allow clad flattening and that ADOPT fuel is acceptable with
14   respect to the clad fattening. The NRC staff also found that the advantage of ADOPT fuel in
15   densification has an impact on the allowable plastic strain criteria of cladding. The NRC staff
16   determined that the reduction in densification due to the ADOPT fuel material achieving higher
17   density during sintering processes gives ADOPT fuel a clear advantage over undoped UO_2 fuel.
18
19   3.6.2.7 Fuel Pellet Overheating (Power-to-Melt)
20
21   The acceptance limit for fuel pellet overheating is that the fuel rod centerline temperature will not
22   exceed the fuel melt temperature during Condition I and II operation, accounting for degradation
23   of the melt temperature due to burnup and the addition of integral burnable absorbers.
24   Section 3.2.3 of the TR and Section 3.3.3 of this SE concluded that there is no difference in the
25   melting point of standard UO_2 fuel and ADOPT fuel. Figure 6-4 of Reference 1 shows that the
26   fuel centerline temperature for ADOPT fuel is slightly lower relative to standard UO_2 fuel.
27
28   The NRC staff concluded that, since the design limit for the centerline melt for ADOPT fuel is
29   the same and calculated centerline temperatures are lower, the power-to-melt limit for ADOPT
30   fuel is as conservative as standard UO_2 fuel.
31
32   3.6.2.8 Pellet-Clad Interaction
33
34   The NRC SRP does not recommend a specific design criterion for PCI or PCMI. Rather, two
35   existing design limits, one percent transient clad strain and no fuel centerline melt, should be
36   satisfied to provide protection against PCI or PCMI fuel failure. PCI addresses stress-corrosion
37   cracking mechanisms due to fission product embrittlement of the cladding, while PCMI is a
38   stress driven failure mechanism. The one percent uniform clad strain criterion limits the clad
39   strain during a transient to a range where the cladding has sufficient ductility to preclude strain
40   related fuel failures. The fuel pellet overheating criterion precludes fuel melting and the
41   associated large volume increase in the fuel due to the phase change that results in excessive
42   cladding stresses and strain.
43
44   The NRC staff concluded that since the ADOPT fuel design meets the design limits, no
45   additional PCI calculations are required.
46
47   The NRC staff reviewed all the fuel design criteria as identified in WCAP-18482-P/WCAP-
48   18482-NP, Revision 0, and determined that the FRD design limits and the upper and lower

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

28

1   bounds calculated with all relevant uncertainties are applied to the ADOPT fuel design. The
2   NRC staff found FRD design limits, and the upper and lower bounds based on all relevant
3   uncertainties acceptable.
4
5   3.6.2.9 Pellet-to-Cladding Interaction Stress-Corrosion Cracking Plant Maneuvering Guidelines
6
7   While no analytical acceptance criterion exists for PCI stress-corrosion cracking (PCI/SCC)
8   cladding failure, licensees develop barriers to prevent cladding failure under normal operations.
9   One such barrier to PCI/SCC cladding failure during normal operations is plant maneuvering
10 and fuel pre-conditioning guidelines.
11
12 Section 3.3.2 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes fuel creep and testing
13 done on ADOPT fuel to characterize performance. It concludes:
14
15   It is clear that at temperatures in excess of 1500°C, the creep rate difference
16   between ADOPT and standard UO_2 increases. This can be attributed to the
17   enlarged grain size. In this temperature regime, the viscoplastic behavior of
18   ADOPT fuel should provide a pellet-clad interaction (PCI) benefit, as the pellet
19   deforms under its own internal stresses and fills in as-manufactured dimples.
20   However, in steady state operations, there is no appreciable difference in the
21   creep behavior of conventional UO_2 and ADOPT fuel.
22
23 In response to the RAI 12 regarding PCI/SCC plant maneuvering guidance (Ref. 2),
24 Westinghouse stated that based on the data presented in WCAP-18482-P/WCAP-18482-NP,
25 Revision 0, it is [
26
27   ]a,c Westinghouse goes on to state that [
28
29   ]a,c
30
31 Based on the information presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, and in
32 response to the RAI 12, the NRC staff finds the proposed strategy being employed to avoid
33 PCI/SCC cladding failure acceptable.

    [Suggested title: "3.6.3.1 FULL SPECTRUM™ Loss-of-Coolant Accident Methodology and NOTRUMP Evaluation Model for ADOPT Fuel"]

34
35 3.6.3   Safety Analyses
36
37 3.6.3.1 ~~Full Spectrum™ Loss-of-Coolant Accident Methodology for ADOPT Fuel~~

    [FULL SPECTRUM]

38
39 Design-basis LOCA analyses is performed to demonstrate that the emergency core cooling
40 system (ECCS) meets requirement of 10 CFR 50.46. ADOPT fuel design does not affect the
41 overall goal of the LOCA analysis. However, it introduces potentially different physical effects
42 which can change the results. This section describes how Westinghouse ~~Full Spectrum~~ LOCA
43 (FSLOCA™) methodology as described in WCAP-16996-P-A (Ref. 32) and the NOTRUMP
44 evaluation model described in WCAP-10054-P-A (Ref. 33) and WCAP-10079-P-A (Ref. 34) are
45 applied to a core with ADOPT fuel. The FSLOCA best-estimate EM is applicable to all PWR
46 fuel designs with Zirconium alloy cladding for a full spectrum of pipe breaks for LOCA analysis.
47 ~~NOTRUMP EM was approved for use in the NOTRUMP EM to calculate the peak cladding~~
48 ~~temperature in the core during a small break LOCA transient.~~

    [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.]

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

29

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. [

]a,c

[

]a,c

Decay Heat

[

]a,c

Clad Deformation-Burst Strain, Relocation

[

]a,c

~~The NRC staff reviewed the physical effects of FSLOCA methodology 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 on ADOPT fuel is acceptable.~~

3.6.3.2 Full Spectrum Evaluation Model Thermal Properties

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

30

1　This section describes the aspects of the FSLOCA evaluation model (EM) (Ref. 32) that could
2　be affected by the ADOPT fuel pellet.
3
4　*Density*
5
6　The chromia and alumina additives adjust the TD of ADOPT fuel downward from approximately
7　10.96 g/cm^3 to [   ]^{a,c} of the TD of UO$_2$. The TD of 684.86 lbm/ft3 as
8　assumed in the FSLOCA EM therefore remains applicable for ADOPT fuel pellets. This value of
9　increased fraction of TD is modeled through user input to FSLOCA analysis.
10
11　*Thermal Conductivity*
12
13　WCOBRA/TRAC-TF2 as used in the FSLOCA EM uses the modified Nuclear Fuels Industries
14　(NFI) model to account for the effects of fuel burnup on pellet thermal conductivity. As
15　discussed in Section 11.4.1 of Reference 32, the modified NFI model represents the thermal
16　conductivity for as-fabricated density of 95 percent of TD, and an adjustment is made to account
17　for as-fabricated fractions other than 95 percent. As mentioned in Section 2.0 of the SE,
18　ADOPT fuel pellets have a [
19
20
21
22
23
24
25
26
27
28
29
30　　　　　　　　　　]^{a,c}
31
32　*Specific Heat*
33
34　No change is necessary to the models used for standard UO$_2$ fuel specific heat when modeling
35　ADOPT fuel pellets.
36
37　*Thermal Expansion*
38
39　There is negligible difference in thermal expansion between standard UO$_2$ fuel and ADOPT fuel
40　pellets. As such, the model described in Section 8.4.1 of Reference 32 remains applicable for
41　ADOPT fuel pellets.
42
43　*Thermal Conductivity of Relocated Fuel*
44
45　Section 8.6.2 of WCAP-16996-P-A (Ref. 32) describes the model used to represent relocated
46　fuel (fuel fragments axially relocated within the location of a rupture). [
47
48　　　　　　　　　　　　　　　　　　　　　　]^{a,c}

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

31

1 [
2
3
4
5
6   ]a,c
7
8   In section 3.3.2 o~~f~~ this SE, the NRC staff concluded that the models in
9   the thermal cond

[Annotation: 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."]

10
11 3.6.3.3 NOTRUM
12
13 This section addresses the impact of the ADOPT fuel pellets on the NRC-approved NOTRUMP
14 EM as described in References 33 and 34 for use in the NOTRUMP EM to calculate the peak
15 cladding temperature in the core during a SBLOCA transient. The models and correlations
16 used in the NOTRUMP EM [   ]a,c as
17 discussed in the following subsections.
18
19 *Density*
20

[Annotation: prop markings not necessary]

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

[Annotation: period]

28
29 *Thermal Conductivity*

[Annotation: 3.3.2]

30
31 Section ~~3.3.4~~ of this SE indicates that the standard Westinghouse methodology for standard
32 UO_2 fuel can be used to calculate the thermal conductivity for ADOPT fuel. The modified NFI
33 model is used in the NOTRUMP EM version of the LOCTA-IV code to account for the effects of
34 fuel burnup on pellet thermal conductivity predicted by the PAD5 fuel performance code.
35
36 *Specific Heat*
37
38 No change is necessary to the models used for standard UO_2 fuel specific heat when modeling
39 ADOPT fuel pellets.
40

[Annotation: Reference 34]

41 *Thermal Expansion*
42
43 Appendix T of ~~References 33~~ describes the thermal expansion of the fuel pellet in the
44 NOTRUMP EM. There is a slight difference in thermal expansion between standard UO_2 fuel
45 and ADOPT fuel. In view of this, the model described in Appendix T of Reference 33 is
46 sufficient for thermal expansion of ADOPT fuel.
47
48 3.6.3.4 Radiological Consequence Analyses

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

32

1 Section 6.2.4 of WCAP-18482-P/WCAP-18482-NP, Revision 0, describes potential impacts of
2 ADOPT fuel pellets on radiological consequence analyses and the applicability of established
3 guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis
4 Accidents at Nuclear Power Reactors," (Ref. 48) and RG 1.195, "Methods and Assumptions for
5 Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear
6 Power Reactors" (Ref. 49). In addition, Westinghouse provided supplemental information in
7 response to an RAI (RAI 13, Ref. 2). With respect to fuel rod performance and predicting the
8 number of failed fuel rods under accident conditions, changes in material properties and
9 performance characteristics between standard UO_2 fuel and ADOPT fuel have been identified
10 and are accounted for, as necessary, in safety analysis models and methods.
11
12 With respect to the Maximum Hypothetical Accident (MHA) (a.k.a. LOCA) radionuclide release
13 fractions, timing of releases, and elemental composition of releases, Westinghouse states that
14 the release fractions are based on accident scenarios involving significant core melt, not
15 impacted by ADOPT fuel pellets, and independent of ECCS performance demonstrations (i.e.,
16 10 CFR 50.46 LOCA evaluations). [
17   ]**a,c** which could impact the timing of releases, are not significantly impacted. In
18 response to the RAI 13, Westinghouse provided information as to why the addition of dopants
19 does not significantly change the chemistry of the fission products released during a core melt
20 accident. An independent NRC assessment of the impacts of large grain doped UO_2 fuel pellets
21 on the MHA-LOCA source term also concluded that any potential impacts were insignificant
22 (Ref. 28).
23
24 With respect to steady-state radionuclide release fractions (i.e., Table 3 gap fractions in RG
25 1.183 (Ref. 48)) used in the non-LOCA dose assessments, Westinghouse stated that given
26 there is **[**   **]a,c**
27
28 Based on the information presented in WCAP-18482-P/WCAP-18482-NP, Revision 0, and in
29 response to the RAI 13 (Ref. 2), the NRC staff found the **[**
30   **]a,c** for radiological consequence analyses acceptable.
31
32 3.6.4   Non-LOCA Transient Analyses
33
34 This section documents the non-LOCA transient analyses evaluation and the ADOPT fuel
35 non-LOCA input to transient analysis.
36
37 *Acceptance Criteria*
38
39 Non-LOCA analyses are performed to satisfy the acceptance criteria for fuel rod failure and
40 coolability. Westinghouse defines two categories of non-LOCA events that need to be
41 considered due to the change in fuel pellet makeup: (1) events that are dependent on core-
42 average effects, and (2) events analyzed to address local effects in the fuel rods.
43
44 For category 1, the non-LOCA events are analyzed to address gross plant criteria, such as loss
45 of shutdown margin, margin to hot leg saturation, overpressurization of the reactor coolant
46 system, overpressurization of the secondary system, or overfilling of the pressurizer. **[**
47

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

33

1
2   ]^{a,c}
3
4   For category 2 analysis to address local effects, the analyses are performed in two steps: 1)
5   predictions of average core response to an initiating event, and 2) such local effects as fuel
6   enthalpy, minimum DNB Ratio (DNBR), fuel melting, and peak cladding temperature (PCT).
7   [
8   ]^{a,c} Section 6.4 of the TR indicates that
9   ADOPT fuel pellets do not affect the fuel cladding DNB performance as determined from DNB
10   experiments or its method of determining the DNBR. Section 3.2.3 of the TR concludes that
11   there is no appreciable difference in the melting point of standard UO_2 fuel and ADOPT fuel
12   pellets, and the fuel centerline temperature for ADOPT fuel is slightly lower relative to standard
13   UO_2 fuel. Since the ADOPT fuel pellets do not impact the properties of the fuel rod cladding,
14   there is no impact on the PCT limits.
15
16   The NRC staff, upon review of the acceptance criteria, has determined that the existing
17   non-LOCA acceptance criteria remain applicable to ADOPT fuel design.
18
19   In summary, the computer codes and methods used in the analysis of the non-LOCA licensing
20   basis events remain applicable for the ADOPT fuel pellet design. The non-LOCA accident
21   acceptance criteria continue to be applicable for the ADOPT fuel pellet design.
22
23   3.6.5   Containment Integrity Analyses
24
25   This section discusses the effect of the ADOPT fuel pellet design on the containment integrity
26   analyses. Any impact would be the result of changes in the mass and energy (M&E) released
27   to containment due to a pipe rupture accident because the containment integrity analyses
28   themselves do not model the fuel. Containment integrity analysis considers M&E released to
29   containment from LOCA or a steam line break (SLB) event.
30
31   LOCA M&E release can be short term or long term. The short-term LOCA M&E releases are
32   used to determine the maximum differential pressure for structural analyses within sub-
33   compartments inside the containment building resulting from postulated pipe ruptures in the
34   primary system piping. Since the parameters that influence short-term M&E releases are break
35   location, temperature of the fluid in the broken pipe, size of the break, and initial reactor coolant
36   pressure, the fuel product and its performance do not influence the short-term M&E. The NRC
37   staff concluded that any change in the fuel materials would not impact the short-term LOCA
38   M&E releases.
39
40   For long-term LOCA M&E release calculations, Westinghouse has three licensed
41   methodologies used for containment integrity, maximum sump temperature, and equipment
42   qualification for Westinghouse and CE designs. The licensed/approved methodologies are:
43
44   •   WCAP-10325-P-A (Ref. 35)
45   •   WCAP-17721-P-A (Ref. 36)
46   •   CENPD-132D (Ref. 37, 38)
47

34

1 The NRC staff reviewed the methodologies used for short-term and long-term LOCA M&E
2 releases and determined that no methodological changes will be required for a full core ADOPT
3 fuel design.
4
5 3.6.6   Short-Term and Long-Term Steam Line Break M&E Releases
6
7 The short-term SLB M&E releases are used to determine the short-term pressure increase
8 transients for structural analyses within sub-compartments inside or outside the containment
9 building resulting from postulated secondary-side pipe ruptures. The transients are performed
10 (typically 1 to 10 seconds duration) and are governed by the mass flux at the break location.
11 Therefore, the parameters that influence the short-term SLB M&E releases are the break
12 location corresponding to the initial secondary system pressure, temperature and quality of the
13 fluid in the postulated ruptured pipe, and the size of the break. Since these transients are of
14 short duration, they are influenced only by the mass flux at the break location. Therefore, the
15 parameters that influence the short-term LOCA M&E releases are the break location, the
16 corresponding temperature of the fluid in the postulated ruptured pipe, the size of the break, and
17 the initial reactor coolant system pressure. This means that any change in fuel pellet materials
18 have no impact on the short-term SLB M&E releases.
19
20 Long-term SLB M&E release analyses use methods and models similar to those for non-LOCA
21 analyses as described in Section 3.6.4 of the SE and remain valid for ADOPT fuel pellet design
22 which is characterized by increased density and enlarged grain size. For the long-term SLB
23 M&E analyses, there are three NRC-approved methodologies:
24
25   •   LOFTRAN (Refs. 39, 40)
26   •   RETRAN (Ref. 41)
27   •   SGNIII (Ref. 42)
28
29 In summary, the NRC staff recognizes the computer codes, methods, and methodologies used
30 for LOCA and SLB M&E releases for containment integrity analysis have been identified. These
31 methodologies and codes were previously approved by the NRC staff. Therefore, the NRC staff
32 determined that the above-mentioned containment integrity analyses methodologies are valid
33 for ADOPT fuel design and are acceptable.
34
35 3.6.7   Nuclear Design Requirements
36
37 The ADOPT fuel characteristics of density, doping materials, and fuel temperature are inputs
38 into the nuclear design methodology based on previously NRC-approved TRs assessing
39 neutronics and nuclear design. The concentration of doping material is sufficiently low so that
40 the doping materials have minimal impact on core reactivity due to its relatively low absorption
41 cross sections. Since fuel temperature of ADOPT fuel is comparable to standard UO_2 fuel, fuel
42 temperature changes will have minimal impact on neutronic behavior of ADOPT fuel. The
43 pertinent ADOPT fuel characteristic which benefits nuclear design is the higher nominal density
44 of [   ]^{a,c} in comparison to the current nominal density of [   ]^{a,c} The higher density of
45 ADOPT fuel can potentially reduce up to four assemblies per cycle due to increased fissile
46 material.
47

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

35

The NRC staff concluded that ADOPT fuel will be explicitly modeled using currently approved Westinghouse nuclear design methods (~~Ref. 43~~) and finds this approach acceptable.

[Annotation: Suggest deleting reference 43 and leaving this statement generic to be consistent with Sections 2.2.4 and 6.3 of the TR.]

3.6.8   Thermal-Hydraulic Design Methods

Westinghouse states that implementation for ADOP[...] or update to any previously NRC-approved methods a[...] ulic analyses. The thermal-hydraulic methods applied to PWR DNB consists of a DNB correlation such as WRB-1 (Ref. 45), WRB-2 M (Ref. 16) and ~~WNG~~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. 45) and the Westinghouse Thermal Design Procedure (Ref. 46). [Annotation: delete]

The ADOPT fuel does not ~~affect the fuel cladding DNB performance~~ as determined from DNB experiments, or the method[...] [Annotation: Suggested wording: "...approved CHF correlations such as the ones listed above..."] lation. The VIPRE-W code can perform steady-s[...] [...]on-LOCA post-critical heat flux (CHF) fuel rod tra[...]

Based on a review the methodologies, the NRC staff ~~determined~~ that the existing Westinghouse thermal-hydraulic design methods and codes and ~~the CHF correlations 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.

36

Suggested wording: "With or without annular pellets and application of ZrB$_2$ integral fuel burnable absorber (IFBA) coating consistent with the defined IFBA parameters in applicable NRC-approved fuel performance or product topical reports."

with correction

1 • AD[   ] materials, such
2   as
3
4 *3   Fuel Limitations*
5
6 • Fuel burnup shall be limited to 62 GWd/MTU peak rod average for all cladding types.
7 • Nominal pellet density range will be [   ]^{a,c}
8 • Fuel grain size range will be [   ]^{a,c} as measured according to ASTM E112 as linear
9   intercept without correction factor, which corresponds to [   ]^{a,c}
10 • Cr range from [   ]^{a,c} which corresponds to inclusion of Cr$_2$O$_3$ ranging from [
11   ]^{a,c}
12 • Al ranging from [   ]^{a,c} which corresponds to inclusion of Al$_2$O$_3$ ranging from [
13   ]^{a,c}
14
15 **5.0   CONCLUSIONS**

delete -P

16
17 The NRC staff has reviewed the Westinghouse's ADOPT fuel TR, WCAP-18482-P/WCAP-
18 18482-P-NP, Revision 0, for direct replacement for standard UO$_2$ fuel. ADOPT fuel is a
19 modified UO$_2$ pellet doped with small amounts of chromia and alumina that results in higher
20 density and enlarged grain size compared to undoped UO$_2$ fuel. The NRC staff's review of the
21 TR has identified and confirmed the ADOPT fuel design has [
22   ]^{a,c}
23
24 The NRC staff's extensive review of the TR consisted of a prolonged virtual audit of supporting
25 documents, requests for additional information, and review of the responses to RAIs. The
26 review consisted of [
27
28
29
30
31   ]^{a,c}
32
33 The NRC staff completed its review of Westinghouse TR titled WCAP-18482-P/WCAP-18482-
34 NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel," and
35 found that WCAP-18482-P/WCAP-18482-NP, Revision 0, is acceptable for referencing in
36 licensing applications to the extent specified and under the limitations delineated in the TR and
37 Section 4.0 of the NRC staff's SE.

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

37

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™ 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 Package Accession No. ML21316A137).

5. Letter from K. Hosack, Westinghouse, to NRC, DCD, "Submittal of WCAP-17642-P-A, Revision 1, 'Westinghouse Performance Analysis and Design Model (PAD5)'," LTR-NRC-17-75, November 27, 2017, (ADAMS Package Accession No. ML17335A334.

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).

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

38

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 E831-19~~, "Standard Test Method for Linear Thermal Expansion of Solid Materials by Thermomechanical Analy ASTM-E831-86 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

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

39

TR WCAP-18482-P/WCAP-18482-NP, Revision 0, 'Westinghouse Advanced Doped Pellet Technology (ADOPT™)'," February 9, 2021 (ADAMS Accession No. ML21047A466).

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

27. **[**

          ]

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

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

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

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

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 Local~~ Methodology)'," LTR-NRC-16-75, November 30, 2016 (ADAMS Accession No. ML16343A232). FULL SPECTRUM LOCA

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
~~OFFICIAL USE ONLY – PROPRIETARY INFORMATION~~

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

40

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~~ ion 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)).

Non-publicly available, Proprietary

40. WCAP-8822-S1-P-A, "Mass a~~nd energy~~ n 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-16259-P-A, "Westinghouse Methodology for Application of 3-D Transient Neutronics to Non-LOCA Accident Analysis," August 31, 2006 (ADAMS Accession No. ML062580357 (Non-publicly available, Proprietary)).~~

Suggest deleting reference 43 and leaving this statement generic to be consistent with Sections 2.2.4 and 6.3 of the TR.

44. WCAP-8762-P-A, "New Westinghouse Corr~~elation~~ dicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 3~~0~~ ~~Acc~~ession No. ML080630433 (Non-publicly available, Proprietary)).

45. WCAP-11397-P-A, "Revised Thermal Desig~~n~~ 89 (ADAMS Accession No. ML080630437 (Non-publicly available, Proprietary).

46. 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).

47. 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).

OFFICIAL USE ONLY – PROPRIETARY INFORMATION

41

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

49. 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).

50. 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.

51. 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.

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

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

54. "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.

Principal Contributors: Mathew Panicker, NRR/DSS/SFNB
                          Paul Clifford, NRR/DSS
Date: March 8, 2022