Enclosure

Response to NRC Request for Additional Information

WCAP-16072-P & -NP

"Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

RAI No. 1a:

Section 5 best describes the ZrB_2 IFBA fuel rod design as consisting of "...a ZrB_2 coating on the outer diameter of UO_2 fuel pellets over the center region of the fuel rod with cutback regions (regions without ZrB_2 coating) on both ends of the fuel rod". The description continues, "Lower enrichment fuel pellets may also be used in a portion of the cutback region....The cutback regions may consist of solid, annular, or a solid and annular fuel pellet combination...".

Evaluations credit the location of annular fuel pellets in the lower power ends of the fuel stack. Yet, the topical does not provide any limitation on the axial length of the annular pellet cutback regions. Please provide the supporting technical basis for your conclusion.

Response 1a:

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]^{a,c} Although the exact length may vary depending on what is required to provide optimal peaking for individual plants or cycles, it is anticipated that [

]^{a,c} No specific limitation on the size of this region is necessary since core design guidelines and cycle specific calculations will explicitly verify that the required power margin in the annular pellet region is maintained.

RAI No. 1b:

The topical states that the ZrB_2 IFBA coating may be natural or enriched with the B^{10} isotope. Yet, the topical does not provide any limitation on the extend of B^{10} enrichment or its potential impact on core physics predictions. Please provide the supporting technical basis for your conclusion.

Response 1b:

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RAI No. 1c:

Evaluations credit the narrow width of the ZrB_2 coating. The topical states that the coating thickness may vary within a specified range. Please describe the impact of coating thickness on core physics predictions.

Response 1c:

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RAI No. 1d:

The IFBA fuel rod design includes lower U^{235} enrichment axial blanket regions. The topical does not provide any detail on limitations on the axial length or the enrichment split of these blanket regions. Please provide additional information on these axial blanket regions.

Response 1d:

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As stated above, the cycle specific reload calculations will verify that the required power margin in the annular pellet region is maintained even in cases where the blankets are fully enriched.

RAI No. 2:

The topical does not provide any information on the impact of U^{235} enrichment axial blanket regions on core physics predictions and safety analyses. Please provide the analyses that demonstrate that the impact of these axial blanket regions are acceptable.

Response 2:

Low enriched blankets have been used extensively in PWR plants throughout the US. Most of the Westinghouse plants currently employ low enriched axial blankets. The Westinghouse physics code ANC has been extensively benchmarked to cycles containing low enriched axial blankets. The PHEONIX/ANC code package is currently being used to perform reload analysis for St. Lucie 2 core, a CE plant that contains low enriched axial blankets. Axial blankets have been used in the St. Lucie 2 core for three (3) cycles. The St. Lucie 2 UFSAR was appropriately updated. Other CE plants have indicated an interest in axial blankets as well.

The DIT/ROCS computer code systems has been used to analyze several of these cycles. Topical report CENPD-275-P-A contains results of benchmarks of the DIT/ROCS computer codes on St. Lucie Unit 1 Cycle 7 and St. Lucie Unit 2 Cycle 3 that contained several LTAs with low enriched axial blankets. In all of these cases, no significant impact on the accuracy of the physics predictions was observed.

The impact of low enriched axial blankets will be explicitly considered in the plant specific safety analysis. The physics analysis that will be performed to support the implementation of the axial blankets will explicitly model the low enriched axial blankets. The impact of these blanket regions on parameters important to safety will be calculated and used to revise the safety analysis where necessary.

RAI No. 3:

The topical does not provide any information on the impact of biases/uncertainties and manufacturing tolerances on the ZrB_2 coating thickness and B^{10} enrichment on core physics predictions and safety analyses. Please provide the analyses that demonstrate that these uncertainties and tolerances have been properly accounted for.

Response 3:

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The impact of uncertainty in B^{10} loading on helium release and internal pressure is conservatively accounted for in a manner as discussed in Response 7.

RAI No. 4:

Section 1.2 states, "...Westinghouse has had considerable fabrication and operational experience with the ZrB_2 Integral Fuel Burnable Absorber". Section 1.3 states, "Post-irradiation examinations of ZrB_2 IFBA test rods revealed no profilometry anomalies in the coated fuel pellet zone, no chemical interaction between the coating and fuel rod cladding, no incipient cracks in the cladding inner diameter, no excessive fuel pellet cracking, nor any anomalies in the fuel structure".

4 Please provide details of Westinghouse's fabrication and operational experience with annular fuel pellets.

5 Please provide details of Westinghouse's fabrication and operational experience with U^{235} enrichment axial blanket regions.

6 Please provide details of post-irradiation examinations of annular fuel pellets.

Response 4:

Summation of IFBA Rods, Axial Blanket (AB) Pellet Rods and Annular Axial Blanket (AAB) Pellet Rods per Year

Year	# of IFBA Rods	# of Rods with AB		# of Rods with AAB	
		Natural	Mid-enriched	Natural	Mid-enriched or Fully-enriched

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RAI No. 5a:

With regard to the modeling capability of the CE design analyses:

Is FATES3B capable of specifically modeling the different axial regions of the IFBA fuel rod (e.g. annular vs solid pellet region, enrichment blankets, ZrB_2 coating and cutback regions)?

Response 5a:

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RAI No. 5b:

Will the axial nodes be aligned in such a way as to avoid splitting these different axial regions?

Response 5b:

The actual lengths of annular pellets and ZrB_2 coatings are expected to coincide closely with FATES3B axial node lengths. However, if they do not, the input and FATES3B adjustments are described in Response 5a.

RAI No. 5c:

What are the limitations of STRIKIN-II with respect to calculating radial power distribution in an annular pellet?

Response 5c:

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]^{a,c} Solid and annular pellet radial power distributions are described in approved FATES3B reports, Reference 38 and References 2 and 3, respectively, of WCAP-16072-P.

RAI No. 5d:

In the annulus region, how does FATES3B model relocation, thermal expansion, and swelling? Is there an experience database available to validate these models specifically for annular fuel?

Response 5d:

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RAI No. 6:

In order to compensate for the helium production associated with the B^{10} depletion, the initial helium fill pressure will be reduced. One consequence of this change would be a lower BOC gap conductivity. Does this reduced gap conductivity remain above the minimum gap conductivity assumed in the UFSAR safety analyses for each of the CE plants?

Response 6:

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RAI No. 7:

Section 2.2 states, "...the maximum and minimum ZrB_2 IFBA helium release will be applied deterministically consistent with the specific applications". Please provide the values and bases for the minimum and maximum helium release fraction.

Response 7:

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RAI No. 8:

Section 4.2.2.4 states, "The evaluations of cladding collapse in the plenum region of the rods demonstrated that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring is factored into the analysis". Is credit for the spring required to compensate for attributes of the IFBA fuel rod design?

Response 8:

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RAI No. 9:

Section 4.2.3 states, "...the post-LOCA Long Term Cooling EM is unaffected and, therefore, not addressed herein". IFBA has the potential to influence the initial critical boron concentration of the RCS which in turn may impact the timing and magnitude of boron precipitation in the LTC Analysis. Please provide the supporting technical basis for your conclusion.

Response 9:

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The maximum RCS boron concentration is determined by the cycle length and the burnable absorber worth loaded into the core. The required burnable absorber worth for CE plants is typically set by number of burnable absorber rods required to reduce the RCS boron concentration to a value low enough to assure that the most positive MTC Tech Spec limit will not be exceeded. [ ]a,c

RAI No. 10:

Section 4.2.4.1 discusses the impact of ZrB_2 IFBA on core peaking. Does the rapid depletion of B-10 result in an increasing radial peaking factors in the beginning of cycle (e.g. Fr increases with burnup initially, then burns down)?

Response 10:

Cores containing ZrB_2 IFBA often experience the highest radial peak sometime after BOC. This is also true for most cores containing gadolinia burnable absorber and for some cores containing the erbia burnable absorber. This behavior is thus not unusual and is not anticipated to cause any problems. The current CE safety analysis methodology does not assume or require that the power peaking be monotonically decreasing.

RAI No. 11a:

Section 4.2.4.1 states, "peak soluble boron concentration may occur sometime after beginning-of-cycle".

Will Plant Operations staff be trained and will Operating Procedures be updated to reflect this new operating scheme?

Response 11a:

This is a plant specific implementation issue, but will be recommended to the utility. Some cycle designs may show a slow and modest increase in the RCS critical boron concentration over the first third of the cycle. When ZrB_2 IFBA is implemented in a specific plant, the utility may decide that some training is necessary in order to alert the plant operations staff to the possibility of this behavior for some cycles. The utility may decide to review and update plant operating procedures to accommodate potential differences in core behavior between ZrB_2 and erbia IFBAs.

RAI No. 11b:

Plant Technical Specifications require MTC surveillance tests to validate the physics predictions (and safety analyses) and ensure that plant operations remain within Technical Specification limits. Please justify how TS SR 3.1.3.1 and SR 3.1.3.2 perform their intended purpose in the presence of an increasing MTC at startup.

Response 11b:

This is a plant specific implementation issue. The Tech Specs currently require that the MTC be measured at HZP, at HFP near BOC (for some plants this has been described as prior to reaching 800 ppm RCS boron), and prior to reaching two thirds of the cycle length. Note that currently only the HZP BOC MTC measurement is used to confirm that the MTC is less than the most positive limit. Thus no explicit changes to the Tech Specs are necessary. However Westinghouse will recommend that procedures be implemented to confirm that the MTC (either by direct measurement or by extrapolation from other cycle specific measurements) is within its limits at the highest RCS boron concentration expected during the cycle.

RAI No. 12:

Figures 4.2-3 and 4.2-4 illustrate higher rod internal pressures with ZrB_2 IFBA. These higher internal rod pressures, especially at lower burnup, translate into a greater challenge to DNB Propagation. With the current methodology, a greater number of pins is likely to balloon and even rupture during Non-LOCA events which experience DNB. Please provide the analyses that demonstrate that this further challenge to DNB Propagation is acceptable from a radiological dose perspective, core coolable geometry perspective, and fuel relocation perspective.

Response 12:

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RAI No. 13:

The annulus region of an IFBA fuel rod (with annular fuel pellets) may potentially be filled with moderator as the result of a clad failure. Has any evaluations been completed to assess the neutronic, thermal, and mechanical behavior of a flawed IFBA fuel rod under normal, transient, and shutdown (including spent fuel pool) conditions?

Response 13:

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The current methodology used for spent fuel pool criticality analysis conservatively assumes that the annular pellet regions are comprised of solid, full diameter pellets. This is conservative since it results in the highest K_{eff} for the fuel storage configuration. The reactivity calculated for this configuration would bound the reactivity associated with a fuel rod comprised of annular pellets containing water in the annulus.

Westinghouse's operating experience with behavior of ZrB_2 IBFA fuel rods does not indicate any difference in behavior from non-IFBA fuel rods. Furthermore, operation with potentially flawed or failed fuel rods is a very low probability. No requirement exists to specifically model such rods. Since the probability of the existence of such flawed/failed fuel is small, e.g., 1 or 2 rods at most in the core, the impact on normal, transient, or shutdown conditions would be insignificant. Consequently, no specific evaluations have been performed and Westinghouse believes that such calculations are not needed.

RAI No. 14:

Section 4.2.3.2 states, "the pool-boiling hot rod heatup calculation for the limiting break of the break spectrum is reanalyzed over the range of rod internal pressures identified by the hot rod fuel performance analysis. A sufficient number of rod internal pressures is analyzed...". Please quantify the range of pressures evaluated and the technical basis of this range.

Response 14:

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RAI No. 15:

As a follow on to RAIs 4c and 5d, please provide analyses that demonstrate that the fuel relocation models are capable of accurately accounting for cladding ruptures within or just below the annular fuel region.

Response 15:

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Additional Information Concerning Dose Calculations

The Westinghouse safety analysis provided in WCAP-16072-P, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," addresses the impact on fuel rod design and safety analyses in the areas of fuel performance, fuel mechanical design, ECCS performance evaluations, non-LOCH (sic) transient safety analyses, and neutronic design. The safety analysis does not address the impact of the proposed changes (i.e., increased fuel rod pressure, annular fuel pellets, and/or axial blankets) on the generation of fission products and the transport of fission products released during LOCA and non-LOCA design basis accidents. These considerations need to be addressed in order for the staff to make a finding that the use of the proposed fuel will not impact assumptions used in the current licensing basis analysis for demonstrating compliance with 10 CFR Part 100, or 10 CFR 50.67, as applicable. For example:

Dose Question 1:

Does the addition of the zirconium diboride coating alter the source term assumptions provided in NUREG-1465 with regard to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release? In particular, does the added zirconium diborate upset current conclusions regarding the radiochemistry of iodines (e.g., does the increased mass of Zr shift the chemical equilibrium of reactions that yield CsI as opposed to ZrI)

Dose Response 1:

The application of ZrB_2 to the CE fleet is basically the same as that implemented for the Westinghouse fleet. The Vantage-5 Fuel Assembly design topical report WCAP-10444, Addendum 1, was reviewed and accepted by NRC on March 12, 1986; where Addendum 1 reported information relative to the use of ZrB_2 in the fuel rod design. Westinghouse has subsequently accrued more than 15 years of satisfactory performance of ZrB_2 design. Based on this experience base and the research and testing that preceded its application in operating plants, [

]^{a,c} Thus, the use of the ZrB_2 fuel will not impact assumptions used in the current licensing basis analysis for demonstrating compliance with 10 CFR Part 100, or 10 CFR 50.67, as applicable.

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[3] WCAP-12921 "Chalk River Irradiation Test of Enriched ZRBs Coatings", February 1991, (Westinghouse Proprietary Class 2)

[4] NUREG-1465

Dose Question 2:

Does the increased helium gas pressure in the fuel pins invalidate assumptions regarding fuel handling accident fission product release iodine scavenging in the spent fuel pool or reactor cavity. Note that the pool decontamination factors in Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," are predicated on a maximum fuel rod pressurization of 1200 psig. (The assumptions in this safety guide were largely developed from the results of experiments performed by Westinghouse as reported in WCAP-7518-L.)

Dose Response 2:

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Safety Guide 25 identifies a rod internal pressure of 1200 psig as being associated with the determination of the pool scrubbing DF provided by the pool of water and states that the DF will be lower for fuel rod pressures greater than 1200 psig. The Safety Guide also states that with pressures >1200 psig the DF is to be calculated on an individual basis using assumptions comparable in conservatism to those used in the Safety Guide. [

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Dose Question 3:

Is the fission product migration in an annular pellet different from that of a solid pellet in a manner that would affect previous analysis assumptions related to the fraction of core inventory in the fuel gap, or on the timing of fission product releases following a transient? Is the correlation between burnup and fission product diffusion different for an annular pellet than a solid pellet?

Dose Response 3:

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These three questions are offered only as examples. The staff expects Westinghouse to provide a full evaluation of the impact of zirconium diboride on the generation and transport of fission products as currently analyzed in DBA radiological consequence analyses.