QUESTION - 1

It appears that the OFA is lighter than the standard 17x17 Fuel Assembly. With the lighter fuel assembly is the turbine overspeed event the only event that predicts lift off. For the turbine overspeed event, in the case of a mixed core, will the adjacent fuel assemblies, i.e., one OFA and one standard, lift off differently, thus causing grid to grid mechanical interaction? Please discuss.

RESPONSE

Even though the Optimized Fuel Assembly (OFA) is lighter than the standard 17x17 fuel assembly, and will lift off sooner, the OFA holddown springs are designed to keep the fuel assembly in contact with the lower core plate under all Condition I and II events with the exception of the turbine overspeed event associated with a loss of external load.*

The holddown springs are identical to the 17x17 standard fuel assembly springs, and are designed to accommodate the possibility of an over-deflection associated with fuel assembly lift off for the turbine over-speed event. Following this transient, the spring holddown force will assure full fuel assembly contact and proper alignment with the lower core plate without experiencing any permanent spring deformation.

For mixed cores, no grid damage is expected, even for the postulated case of an OFA fully lifted adjacent to a standard fuel assembly fully seated. For this case, the lift off does not translate to an axial distance beyond the adjacent grid height; thus the potential for grid-to-grid "hangup" is precluded. In summary, the momentary lift off will result in no adverse effects on the fuel assembly performance.

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*i.e., the same event as for standard 17x17 fuel (ref. RESAR-3S, p. 4.4-32)

QUESTION - 2

(a) Clarify the Burnable Poison Design bases for Condition I and II events given in part (b) of question 231.8 (b) Provide internal pressure limits for all core components. (c) Summarize for all core components the clad stress, strain limits, citing typical values for each. Also provide expected lifetimes for each core component.

RESPONSE

(a) The mechanical design basis for the burnable poison rods and for all core component rods, is that dimensional stability and cladding integrity shall be maintained during Condition I and II events. (b) Core component rod internal pressures are stated in the response to Question 231.19 with the exception of the Ag-In-Cd absorber rod which is [ ]+ psi and the B_4C absorber rod which is approxi- (a,c)
mately [ ]+ psi. (c) The table below summarizes, the requested clad (a,c)
stress, strain limits, typical values and design lifetimes for all core components.

CONDITION I & II CORE COMPONENT CLADDING DESIGN VALUES

Component	Yield Strength #/in^2	Stress Limit (S_m) #/in^2	Typical Stress Values #/in^2	Strain Limit Percent	Typical Strain Values Percent	Design Life	
RCC (Ag-In-Cd)	62,000	41,333	[ ]+	1	[ ]+	15 yrs.	(a,c)
RCC (B_4C)	62,000	41,333	[ ]+	1	[ ]+	15 yrs	(a,c)
Burnable Poison	62,000	41,333	[ ]+	1	[ ]+	3 yrs	(a,c)
Primary Source	62,000	41,333	[ ]+	1	[ ]+	1 yr	(a,c)
Secondary Source	62,000	41,333	[ ]+	1	[ ]+	15 yrs	(a,c)

QUESTION - 3

No design basis or analysis are given in the WCAP-9500 text for guide thimble wear. Consider updating the text in WCAP-9500 or amending your response to Question 231.41 with a more appropriate response.

RESPONSE

The response to WCAP-9500 Question 231.41 can be amended by including the following information in SRP format:

Design Basis - The design basis for the OFA guide thimble wear is that the remaining guide thimble tube shall provide structural integrity of the fuel assembly and functionability of the guide thimble tube.

Design Limit - Stress analysis has shown that the fuel handling design criterion of 6g is the most limiting load on the fuel assembly. The OFA design is limited to operate under an RCC location with control rods in the parked position for [ ]+, without exceeding the 6g load criterion. The guide thimbles remain structurally adequate to preclude buckling due to axial loads, and to limit tensile and compressive stresses to within acceptable limits. Normal present day fuel shuffle planning prevents locating a fuel assembly in an RCC location for more than three cycles of operation. (a,b,c)

Design Evaluation - A detailed review of all available information on thimble wear in the vicinity of the top grid results in the conclusion that wear is primary due to a combination of flow induced rodlet vibration and mechanical misalignment. Thimble wear

RESPONSE (Con't)

information obtained from hydraulic tests, hot cell examinations, on-site measurements, and mechanical design analysis has indicated that guide thimble thinning is acceptable and is not a safety problem. Specifically, the fuel assembly structural integrity is maintained when functioning with an RCC assembly in the fully withdrawn position for a minimum of [ ]+ (a,b,c)

QUESTION - 4

Provide a discussion of the effects of irradiation growth of the SS clad for both BPs and control rods.

RESPONSE

The stainless steel clad for both the BPs and the control rcds is an isotropic material. The irradiation growth of this material based on experimental information[1] is expected to be negligible.

1. Foster, J. P. and Strain, R. V."Empirical Swelling Equations for Solution-Annealed Type 304 Stainless Steel". Nuclear Technology, Volume 24 - October 1974, p. 93

QUESTION - 5

Confirm that the fuel moisture level is less than that specified in SRP-4.2 Rev. 1 p. 4.2-5, ite. K.

RESPONSE

Per the SRP 4.2: "Hydriding as a cause of failure is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM Specifications for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources." Westinghouse manufacturing specifications ensure that the above SRP hydriding requirements are met.

QUESTION - 6

Provide a more quantified response to the fuel assembly growth discussion given for Question 231.40. The design basis, design limit, design evaluation, and typical values of fuel assembly growth need to be clarified.

RESPONSE

Design Basis - The fuel assembly design shall accommodate differential movement of the fuel rods with respect to the fuel assembly structure caused by temperature and irradiation. Specifically, the fuel rod design must be such that no axial interference occurs between the fuel rod and the top and bottom nozzles. Further, there shall be no axial interference between the fuel assembly and upper and lower core plates caused by temperature or irradiation.

Design Limit - The minimum gap which allows for differential growth between the fuel rods and the fuel assembly is slightly greater than [ ]$^{+}$ of the fuel rod length. The minimum gap between the fuel assembly and reactor internals is [ ]$^{+}$ of the fuel assembly length. (a,c) (a,c)

Design Evaluation - Fuel assembly growth appears to be primarily due to irradiation growth of stress free guide thimbles, and is a result of the Zircaloy's tubing anisotropic texture. In addition to this irradiation-induced growth, irradiation induced creep of the guide thimbles affects fuel assembly growth to a lesser extent. Creep occurs as a result of the stresses in guide thimbles due to holddown spring forces acting to compress the guide thimbles, and due to fuel

RESPONSE (Con't)

rod expansion forces acting through the grid springs tending to lengthen the guide thimbles. The balance between these two effects changes with burnup as grid springs relax and holddown forces change. As a result, the creep contribution to fuel assembly growth can be either positive or negative, and can change with residency time.

Fuel rod growth has been represented by an empirical equation with error bands obtained from post irradiation fuel rod growth measurement data.

[ ]$^+$ (b,c)

Worst case growth calculations are combined with worst case fabrication tolerances to determine the acceptability of the fuel rod and fuel assembly design.

TYPICAL VALUES

FUEL ROD GROWTH VALUES

Fuel Rod Length	Growth Gap Provided	EOL Clearance Remaining	
151.6 inches	[ ]$^+$ inch min	[ ]$^+$ inch	(a,c)

FUEL ASSEMBLY GROWTH VALUES

Fuel Assembly Effective Length	Growth Gap Provided	EOL Clearance	
[153.4]$^+$ inches	[ ]$^+$ inch min.	[ ]$^+$ inch	(a,c)

The following is a set of recommended text changes to WCAP-9500 recently requested by the NRC.

1. Add a sentence to correct paragraph (a) on page 4.2.1. The Reg. Guide should be 10CFR50.46 instead of 10CFR100 for the dose limits.

2. Change "Appendix K" to 10CFR50.46 in your response to Question 231.2. The design basis for rod bursting will now read: "The design basis for fuel rod bursting is to maintain compliance with 10CFR50.46."

3. Per Question 231.10, add "A" version to references in WCAP-9500 when appropriate (e.g. WCAPS 8963/9401). This will be done when appropriate; "A" versions of these topicals do not presently exist.

4. Per Question 231.12, cite Reference 7 on page 4.2.3 of the WCAP-9500 text. Reference will be made appropriately.

5. Per Question 231.13, cite reference 7 on page 4.2.5 of the WCAP-9500 text. Reference will be made appropriately.

6. Per Question 231.22, cite the 6 new references on page 4.2-26 of the WCAP-9500 text. Westinghouse recommends that to support the statement in the text concerning fuel clad fatigue that reference 1 only be cited, since it fully supports the experimental data. The remaining references were previously cited as peripheral information.

7. Per Question 231.24, cite the 3 new references on page 4.2-34 of the WCAP-9500 text. References will be made appropriately.

8. Per Question 231.26, remove confusion on page 4.2-39 by writing "Normal transient, and accident . . . " rather than ". . . Normal accident and transient . . ." The text on this page will be modified accordingly.

9. Per Question 231.27, fix typo (30% instead of 20%) on page 4.2-39. This correction will be made.

10. Per Question 231.30, fix typo (5% instead of 1%) on Page 15.4.21. This correction will be made.

11. Per Question 231.30, correct incorrect statement. The text will contain the corrected statement. The radiological consequences from this event would be less than those from the locked reactor coolant pump rotor accident analyzed in subsection 15.3.3.3.

12. Per Question 231.35, Change "all" to "most" on page 4.4-19 of the WCAP-9500 text or make some other appropriate change. Westinghouse recommends that "all" remain in the text, since some uncertainty is used in all fuel temperature evaluations.

13. The results given in WCAP-9401, Section 3.0, only apply to certain plants. Therefore, add a statement to page 4.2-37 in WCAP-9500 to clarify this. The phrase, "For the plants encompassed by," will be added to the beginning of the first sentence of item (2) in the WCAP-9500 text. page 4.2-37.