ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 390

WBN-TS-02-13 - REVISE TS SECTION 5.9.5 TO INCORPORATE ANALYTICAL METHODS FOR ROBUST FUEL ASSEMBLY (RFA)-2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

NON-PROPRIETARY

Responses to the NRC Request for Additional Information on the RFA-2 Licensing Submittal (WBN-TS-02-13) for the Watts Bar Nuclear Plant

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Responses to the NRC Request for Additional Information on the RFA-2 Licensing Submittal (WBN-TS-02-13) for the Watts Bar Nuclear Plant

1. **Provide seismic and LOCA loading results using the SRSS method for the most limiting mixed core condition of RFA-2 and V+/P+ fuel assemblies.**

   The fuel assembly grid impact forces were obtained using a reactor finite element model consisting of fuel assemblies arranged in a planar array. For the analysis of Watts Bar Nuclear Plant, arrays of fifteen, thirteen, eleven and seven fuel assemblies were used in the model. Each fuel assembly is simplified to a lumped mass-spring model. The methodology used for this analysis is based on the approved methodology in WCAP-9401-P-A (References 1 and 2).

   Two limiting mixed core configurations of RFA-2 and V+/P+ fuel assemblies were evaluated with the planar arrays for Watts Bar. The first configuration consisted of arrays with RFA-2 fuel assemblies in the peripheral locations and V+/P+ fuel assemblies in the remainder of the array. The other configuration consisted of arrays with V+/P+ fuel assemblies in the peripheral locations and RFA-2 fuel assemblies in the remainder of the array.

   The calculation of the maximum LOCA and Seismic grid impact forces were combined using the square root sum of the squares method (in accordance with NUREG 0800, Section 4.2, Appendix A). The maximum SRSS grid impact forces for the mixed core configurations were [ ]$^{a, b, c}$ on the RFA-2 mid-grid and [ ]$^{a, b, c}$ on the V+/P+ mid-grid. The maximum grid impact force in the RFA-2 homogeneous core is [ ]$^{a, b, c}$ for the RFA-2 mid-grid. Since all grid impact forces (homogenous core and mixed cores) are well below the grid allowable limitation, which is greater than [ ],$^{a, b, c}$ coolable geometry is maintained with no grid deformation predicted.

2. **As stated in Section 4.2 of the Attachment to your submittal, the WRB-1 DNB correlation will continue to be used for the V+/P+ fuel. The W-3 correlation is used when conditions are outside the range of the WRB-1 or WRB-2M correlations. The WRB-1 and W-3 correlation topicals are not listed in the COLR documents list. Please provide a reference for their use.**

   The use of the WRB-1 correlation and the W-3 correlation is addressed in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," which is the first reference in the COLR documents list (Technical Specification 5.9.5.b). In addition, the WRB-1 and W-3 correlations are referenced in the current existing WBN TS Bases on Page B 2.0-2 and Page B 2.0-6, as follows:

   WRB-1 (Ref. 7) - (WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.)

W-3 (Ref. 8 and 9); (Ref. 8 - Tong, L. S. "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.)

(Ref. 9 - Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pages 31 through 41, American Society of Mechanical Engineers, New York, 1969.)

Note that References 8 and 9 were renumbered as 9 and 10 in TVA's Proposed TS Bases changes provided February 14, 2003.

3. **The licensee has included a section in Attachment 8 of the submittal entitled, "Responses to NRC Staff Questions on the Proposed Use of WRB-2M." RFA-2 fuel is stated to be a minor modification of the design addressed in WCAP-15025. Please explain what minor means. It appears that pressure increased across the assembly and caused an increase in core bypass flow. How specifically did you account for these differences between fuels?**

In the response to WCAP-15025-P-A SER Condition 1, it was stated that the structural mid-grid design used in the RFA-2 fuel assembly is a minor modification of the Modified Low Pressure Drop mid-grid design that was addressed in WCAP-15025-P-A for use with the WRB-2M DNB correlation. The modified V5H (MV5H) mid-grid restored DNB margins and eliminated fuel assembly vibration associated with the original V5H mid-grid design. The RFA-2 mid-grid is a modification to the MV5H mid-grid to further improve its resistance to fuel rod fretting wear. The modification [ ].[a,c] This modification improves fretting wear, but does not significantly affect any other thermal-hydraulic or mechanical performance features. The RFA-2 mid-grid design has the same mixing vane shape and pattern as the mid-grid design that was addressed in WCAP-15025-P-A. The RFA-2 Intermediate Flow Mixer (IFM) grids are the same design as the Modified IFM grids addressed in WCAP-15025-P-A.

As noted in the Attachment to the submittal, the RFA-2 mid-grid design was evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), Reference 3. The Westinghouse notification to the NRC of the RFA-2 mid-grid design modifications was provided in Reference 4. Reference 4 addressed the design categories and associated parameters in FCEP that were potentially impacted by the RFA-2 mid-grid design changes. It was demonstrated that the RFA-2 mid-grid design changes had an insignificant impact on these parameters except for the significant improvement in fretting wear resistance. As discussed in Reference 4, the minor changes associated with the RFA-2 mid-grid design did not change the hydraulic resistance.

The RFA-2 fuel assembly has an increased hydraulic resistance compared to the current V+/P+ fuel design in Watts Bar because the RFA-2 fuel assembly design includes the use of three Intermediate Flow Mixer (IFM) grids. The increased flow resistance of the RFA-2 fuel assembly caused by the use of IFM grids results in an increased core bypass flow from 9.0% to 9.6%. The core average and outlet temperatures increase slightly due to the increased bypass flow. These changes to the NSSS design parameters resulting from the use of IFMs, as identified in Table 1.1 of the Attachment to the submittal, were evaluated in all aspects of the analyses discussed in Sections 1 through 7 of the Attachment to the submittal.

4. **Page B 2.0-4 of Enclosure 3 to your submittal specifically removes an additional 10% DNBR margin. Because it is unclear how you are certain that the margin was offset, please provide a roadmap of how you arrived at the removal of this 10% margin.**

The DNBR limits listed on Page B 2.0-4 of Enclosure 3 of the submittal are the values that satisfy the DNB design criterion for the DNB analyses performed with the Revised Thermal Design Procedure (RTDP), Reference 5. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least a 95% probability (at a 95% confidence level) that DNB will not occur on the limiting fuel rods for any Condition I or II event.

In addition to the above considerations for uncertainties, additional DNBR margin was maintained in the RTDP analyses by performing the safety analyses to selected DNBR limits which are higher than the design limit DNBR values. Sufficient DNBR margin was maintained in the safety analysis DNBR limits to offset the known DNBR penalties, e.g., rod bow. The net remaining DNBR margin, after consideration of the DNBR penalties, is available for operating and design flexibility issues, such as the cycle-specific DNBR penalty associated with a mixed core of RFA-2 and V+/P+ fuel assemblies

A summary of the design limit and safety analysis limit DNBR values as well as the DNBR margins and penalties for the first transition cycle is presented in Table 1. [

] **a, b, c**

Table 1 shows that there is sufficient DNBR margin for the first transition cycle to RFA-2 fuel. The margin assessment in Table 1 is based on 76 feed RFA-2 fuel assemblies in the first transition cycle.

The net DNBR margin can change on a reload-specific basis. Therefore, only the design limit DNBR values are listed in the proposed changes to the Bases of the Technical Specifications (as provided in Enclosure 3), since the design limit DNBR values are the values that have to be met to satisfy the DNB design criterion and will not change as a result of a reload cycle design.

References:

1. Davidson, S. L. (Ed.), "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A, August 1981.

2. Letter from C. O. Thomas to E. P. Rahe, "Supplemental Acceptance Number 1 for Referencing of Licensing Topical Report WCAP-9500A", dated November 12, 1982.

3. Davidson, S. L. (Ed.), "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

4. Letter from H. A. Sepp, (Westinghouse) to J. S. Wermiel (NRC), "Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design, Revision 1 (Proprietary)," LTR-NRC-02-55, November 13, 2002.

5. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

Table 1
DNBR Margin Summary for Watts Bar Nuclear Plant
First Transition Cycle

	V+/P+ (w/o IFMs)		RFA-2 (w/ IFMs)	
DNB Correlation	WRB-1		WRB-2M	
Cell type	Typical	Thimble	Typical	Thimble
DNBR Correlation Limit	1.17	1.17	1.14	1.14
DNBR Design Limit	1.25	1.24	1.23	1.23
DNBR Safety Analysis Limit				
DNBR Margin (between the Design Limit and the Safety Analysis Limit)				
DNBR Margin from $F_{\Delta H}$ reduction (from 1.65 to 1.62)				
Total DNBR Margin				
Total of DNBR penalties to address rod bow, transition core, and other design/operational margin assessments				
Available Unused DNBR Margin				

a, b, c