UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 2, 2010

Mr. Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development
and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSEE'S FINAL SAFETY ANALYSIS REPORT AMENDMENT RELATED TO MECHANICAL AND CIVIL ENGINEERING SYSTEMS (TAC NO. ME2731)

Dear Mr. Bhatnagar:

By letter dated January 11, 2010 (NRC Agencywide Document Access and Management System Accession No. ML100191686), to the U.S. Nuclear Regulatory Commission (NRC), the Tennessee Valley Authority provided an update (Amendment No. 97) to the Final Safety Analysis Report (FSAR) for the Watts Bar Nuclear Plant (WBN), Unit 2. That update contained changes to a number of sections of the WBN Unit 2 FSAR, including Sections 3.9.1, 3.9.2, 3.9.3, and 5.5.1. The NRC staff has reviewed these four sections and has identified additional information that is needed to complete the technical review of the operating license application.

A response is required within 30 days of receipt of this letter.

If you should have any questions, please contact me at 301-415-1457.

Sincerely,

Joel S. Wiebe

Joel S. Wiebe, Senior Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFROMATION

WATTS BAR NUCLEAR PLANT, UNIT 2

FINAL SAFETY ANALYSIS REPORT AMENDMENT NOS. 95, 96, AND 97

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

By letter dated January 11, 2010 (NRC Agencywide Document Access and Management System Accession No. ML100191686), to the U.S. Nuclear Regulatory Commission (NRC), the Tennessee Valley Authority (TVA) provided an update (Amendment 97) to the Final Safety Analysis Report (FSAR) for the Watts Bar Nuclear Plant (WBN), Unit 2. This update contained changes to a number of sections of the WBN Unit 2 FSAR, including Sections 3.9.1, 3.9.2, 3.9.3, and 5.5.1. The NRC staff has reviewed these four sections and has identified additional information that is needed to complete the technical review of the operating license application.

EMCB Request for Additional Information (RAI) 3.9-1

The NRC staff noted a number of instances in the review of Sections 3.9.1, 3.9.2, 3.9.3 and their corresponding tables and figures of Amendment No. 97 to the WBN Unit 2, Final Safety Analysis Report (FSAR) (Reference 1) where editorial modifications may be necessitated in subsequent revisions to the WBN Unit 2 FSAR. Please review the following NRC staff notations and rectify, as necessary.

1) On page 3.9-18 of Reference 1, continuing to page 3.9-19, the first two paragraphs of Section 3.9.2.5.6, "Results and Acceptance Criteria," are duplicates of the first two paragraphs of the following section (3.9.2.5.7), also titled "Results and Acceptance Criteria."

2) On page 3.9-36 of Reference 1, superfluous spaces exist between the word "Table" and "3.9-17."

3) On page 3.9-44 of Reference 1, the primary membrane plus primary bending stress limit should be "1.1 S" versus the current "1.1.S."

4) On page 3.9-63 of Reference 1, the title of Table 3.9-5 should be revised to state that the limits are "Maximum Deflections" versus the current wording of "Maximum Defections."

5) On page 3.9-63 of Reference 1, Note 1 references Westinghouse Commercial Atomic Power (WCAP)-5890 with a corresponding superscript of number 21, indicating that this refers to Reference 21. Page 3.9-58 of Reference 1 indicates that this WCAP report is Reference 22, not Reference 21. If this is not erroneous, please provide additional justification in conjunction with RAI 3.9.2-3 below.

6) On page 3.9-77 of Reference 1, the third note corresponding to Table 3.9-16 should be revised to correct the misspelling of "Non-pressure" and "other justifiable" versus the current wording of "Non-pressur" and "othe justifiable."

EMCB RAI 3.9.1-1

In Supplemental Safety Evaluation Report (SSER) 6 (Reference 3), the NRC staff noted that the licensee's piping evaluation for a postulated main feedwater header rupture transient, which results in a water hammer event due to a rapid check valve closure, included an assumption that certain feedwater piping system supports failed when the loads exceeded their calculated capacities; this was listed as an open item in SSER 6 (tracked as Outstanding Issue 20(a)). In SSER 13 (Reference 6), the staff noted that the analyses performed, which postulated pipe support failures, was acceptable based on the difficulty involved with making subsequent pipe support modifications and the low probabilistic nature involved with the water hammer transient. Additionally, as part of the closure of this open item, SSER 13 also included a copy of a report performed by Brookhaven National Laboratory (BNL) regarding this issue. BNL was contracted by the NRC to evaluate the licensee's piping analyses performed to demonstrate compliance with the criteria of Appendix F of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code. BNL concluded that the licensee's piping analyses performed for the feedwater loops inside containment were sufficient and demonstrated that the piping system would maintain its structural integrity when subjected to the dynamic loading associated with the water hammer event.

Please describe the applicability of the conclusions made by the NRC staff and the contractor (BNL) regarding the piping analyses described above as they relate to the current WBN Unit 2 refurbishment efforts. Please indicate whether the same issues exist with the inability to modify certain piping supports within containment and whether the piping analyses for the WBN Unit 2 feedwater loops are the same as those analyses performed in support of WBN Unit 1. If these analyses are dissimilar, please summarize and provide justification for any portions of the analyses that are not exactly the same and whether the results of these dissimilar analyses demonstrate that the feedwater piping loops meet the acceptance criteria of the code of record for this piping system.

EMCB RAI 3.9.2-1

In Section 3.9.2.3 of Reference 1, it is indicated that Sequoyah Nuclear Plant Unit 1 and Trojan Nuclear Power Plant (Trojan) "...have been instrumented to provide prototype data applicable to Watts Bar" for the purposes of evaluating the flow induced oscillatory pressure effects on the reactor vessel internals. Additionally, it is concluded, based on scale model test results and "...preliminary results from Trojan...," that plants with neutron shielding pads exhibit less core barrel vibration than plants with thermal shields. Based on the fact that Trojan ceased operations in the year 1992, please discuss the applicability of the statements above, which are currently included in Reference 1. If these data was captured during Trojan's operational state, please describe how this operating experience has been applied to the design or operational characteristics of any of the reactor vessel internals. Additionally, please indicate whether additional results, other than the "preliminary results" mentioned in Reference 1, were utilized to provide additional information regarding the comparison between plants with neutron shielding pads and plants with thermal shields as they relate to core barrel excitation.

EMCB RAI 3.9.2-2

The analyses methods described in Section 3.9.2.5 of Reference 1, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions," were approved for use by a previous license amendment request submitted for WBN Unit 1. These methods incorporate the use of the MULTIFLEX, LATFORCE, FORCE-2 and WECAN computer codes to model the complex, nonlinear thermal-hydraulic loadings induced on the reactor vessel internals under upset loading conditions. Please confirm that the inputs used to analyze these conditions for WBN Unit 2 are the same inputs as those used to analyze the loadings induced on the WBN Unit 1 reactor vessel internals. If any variances exist between the WBN Unit 1 and WBN Unit 2 inputs for these codes, including primary and secondary loadings, flow parameters, mass models, finite element formulations, or other input parameters, provide justification for the variation and its effects on the ability of the WBN Unit 2 reactor vessel internals to meet the acceptance criteria provided in Table 3.9-5. Additionally, please clarify whether the references to "Watts Bar Unit 1" on pages 3.9-15, 3.9-19, and 3.9-20 (2) are correctly referring to WBN Unit 1 for purposes of comparing analyses or whether these instances are incorrect (i.e., these references should state WBN Unit 2 and not WBN Unit 1).

EMCB RAI 3.9.2-3

Table 3.9-5 of Reference 1, "Maximum Def[l]lections Under Design Basis Event (in)," provides the maximum allowable and no loss-of-function limits for the reactor vessel internals under design basis loading conditions. Note 1 to Table 3.9-5 indicates that WCAP-5890 provides limiting criteria for internals deflection based on stress levels induced in the internals structures. Please discuss whether the acceptance criteria provided in Table 3.9-5 are based on WCAP-5890. If these criteria are based on this WCAP report, please provide the bases for the regulatory acceptance of this report. If these criteria are based on a methodology other than the WCAP report, please provide additional information regarding the development of these deflection limits and the bases for the regulatory acceptance of this alternate methodology.

EMCB RAI 3.9.2-4

Please provide justification for the variance between the WBN Unit 1 and WBN Unit 2 allowable and no loss-of-function deflection limits as this variance relates to the upper barrel expansion and compression limits and the no loss-of-function limit for the upper package axial deflection. This justification should include information regarding whether there are variations in the analyses methodologies for determining the WBN Units 1 and 2 reactor vessel internals faulted loads (as requested in RAI 3.9.2-2). Additionally, this justification should indicate whether there are variations in the acceptance criteria for the WBN Units 1 and 2 deflection limits.

EMCB RAI 3.9.3-1

In SSER 4 (Reference 2), the NRC staff noted that a sampling program was initiated by TVA to determine whether the compressive stresses imposed on short column pipe supports exceeded the buckling criteria margin established by the NRC. The NRC staff accepted the sampling program and determined that TVA had adequately addressed the NRC design criteria for Class 2 and 3 pipe supports; this resolved Outstanding Issue 2. Please confirm the applicability of the sampling program discussed in Reference 3 as it relates to Class 2 and 3 pipe supports at WBN Unit 2. If this sampling program was not used in support of the WBN Unit 2 refurbishment effort, please discuss the current criteria used for demonstrating that these pipe supports maintain sufficient margin against critical buckling of short column pipe supports.

EMCB RAI 3.9.3-2

In SSER 6 (Reference 3), the NRC staff noted its concerns regarding the licensee's use of earthquake experience data to seismically qualify Category I(L) piping and identified this concern as Outstanding Issue 19(h). In SSER 8 (Reference 5), the NRC staff noted that the licensee had developed screening criteria to identify items in Category I(L) piping systems that may require further evaluation based on this earthquake experience data. Additionally, the licensee indicated that bounding stress cases would be performed to demonstrate the conservatism of these screening criteria. The NRC staff found this screening criteria adequate for demonstrating the seismic ruggedness of Category I(L) piping. Please confirm that this screening has been performed for the WBN Unit 2 refurbishment efforts. If this screening method was not utilized in the seismic qualification of the WBN Unit 2 Category I(L) piping, please discuss the criteria that has been used to seismically qualify these piping systems and discuss the regulatory acceptance bases for this alternate criteria.

EMCB RAI 3.9.3-3

In addition to the screening methods used for Category I(L) piping systems described in RAI 3.9.3-2, SSER 8 also describes TVA's criteria used for the evaluation of Category I(L) piping supports. The NRC staff noted in SSER 8 that TVA had indicated it would utilize a factor of safety of three in their evaluation of concrete expansion anchor bolts for these pipe supports. The NRC staff accepted the use of this safety factor value for validating the existing design of concrete expansion anchors used in this piping system based on TVA's implementation of recommendations including additional concrete inspection, anchor spacing, and concrete edge distance in conjunction with the existing anchor bolts. The NRC staff also noted in SSER 8 that for future Category I(L) piping, the required safety factors for these piping systems found in the former Office of Inspection and Enforcement (IE) Bulletin 79-02, should be utilized. Please discuss whether the existing, applicable Category I(L) piping supports at WBN Unit 2 have been evaluated in the manner described in SSER 8. If these supports have been evaluated in a dissimilar manner, please provide justification for the departure from the methods described in Reference 4.

EMCB RAI 5.5.1-1

Please discuss whether TVA has committed to perform an augmented inservice inspection of the reactor coolant pump (RCP) flywheel. If no commitment has been made, please provide justification that the potential for excessive vibration on the reactor coolant pump flywheels will be adequately addressed to minimize the possibility of RCP shaft or flywheel failure.

References

1) Letter from M. D. Jesse, Exelon Generation Company, LLC, to NRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) – Unit 2 – Final Safety Analysis Report (FSAR), Amendment 97," dated January 11, 2010. (Accession Nos. ML100191421 (letter), ML100191684 (Section 3.8.5-3.11))

2) NUREG-0847, Supplement 4, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated March 31, 1985. (Accession No. ML072060524)

3) NUREG-0847, Supplement 6, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated April 30, 1991. (Accession No. ML072060464)

4) NUREG-0847, Supplement 7, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated September 30, 1991. (Accession No. ML072060471)

5) NUREG-0847, Supplement 8, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated January 31, 1992. (Accession No. ML072060478)

6) NUREG-0847, Supplement 13, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated April 30, 1994. (Accession No. L072060484)

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Request for Additional Information

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DISTRIBUTION:
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WJessup, DE/EMCB

ADAMS Accession No. ML101530474 *via memo

OFFICE	LPWB/PM	LPWB/LA	EMCB/BC	OGC-NLO	LPWB/BC
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