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

ENCLOSURE

SAFETY EVALUATION REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CLOSEOUT OF NRC BULLETIN 88-02 ISSUES

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-~~237~~ AND 50-~~238~~
50-327 & 50-328

1.0 INTRODUCTION

The Tennessee Valley Authority (the licensee) submitted its response to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes" by letters dated March 1, March 31, and August 2, 1988 and by letters dated July 5, 1989 and January 24, 1990. Bulletin 88-02 requested that licensees for plants with Westinghouse steam generators employing carbon steel support plates take certain actions specified in the bulletin to minimize the potential for a steam generator tube rupture event caused by a rapidly propagating fatigue crack such as occurred at North Anna Unit 1 on July 15, 1987.

2.0 DISCUSSION

The licensee reports that the Sequoyah Units 1 and 2 steam generators exhibit indications of corrosion with magnetite at the uppermost support plate although these indications are only minimal in the case of Unit 2. Accordingly, Items C.1 and C.2 of the bulletin are applicable to both Sequoyah Units 1 and 2.

In accordance with Item C.1 of the bulletin, the licensee has implemented an enhanced primary-to-secondary leak rate monitoring program which is described in the licensee's March 1 and March 31, 1988 submittals. This enhanced leak rate monitoring program is an interim compensatory measure pending completion of the actions requested in Item C.2 of the bulletin and NRC staff review and approval of these actions. This safety evaluation report is the staff's evaluation of those actions.

The licensee has implemented the generic program developed by Westinghouse to resolve Item C.2 of the bulletin. The licensee's implementation of this program is described in Westinghouse reports WCAP-12289 (Proprietary Version) and WCAP-12290 (Non-Proprietary version) which was submitted with the licensee's letter dated July 5, 1989. This report describes the analyses which were conducted to establish the susceptibility of the Sequoyah steam generator tubes to rapidly propagating fatigue cracks and to identify any needed corrective actions.

The staff has reviewed the Westinghouse generic program and documented its evaluation in the attached Safety Evaluation Report (SER) (Reference 1). The staff concluded in the SER that the Westinghouse program is an acceptable approach for resolving Item C.2 of the Bulletin. The staff further concluded that the Westinghouse program, if properly implemented, will provide reasonable assurance against further failures of the kind which occurred at North Anna Unit 1.

The analyses for the Sequoyah steam generators in WCAP-12289 and WCAP-12290 conservatively assumed that all unsupported tubes are dented at the uppermost support plate. In addition, the stress ratio and fatigue estimates were based on the assumption of a full mean stress effect (i.e, yield stress), consistent with staff findings in the attached SER.

Stability ratios for the Sequoyah steam generator tubes were determined from the FASTVIB computer code using thermal-hydraulic input from a 3-D ATHOS model for assumed reference operating conditions (e.g., steam pressure and flow, circulation ratio) which are conservative for current operating cycle parameters. Flow peaking factors were determined for the anti-vibration bar (AVB) geometry at Sequoyah Units 1 and 2 on the basis of results from air model tests. Staff questions regarding how placement of the AVBs in the tube bundle was determined and the conservatism of the assumed flow peaking factors were addressed in an acceptable manner in the licensee's letter dated January 24, 1990.

The analyses documented in WCAP-12289 show that one unsupported tube in Unit 1 and two tubes in Unit 2 failed to satisfy the Westinghouse stress ratio criterion. These three tubes have been stabilized and plugged. All other unsupported tubes in the Sequoyah steam generators satisfy the Westinghouse stress ratio criterion and are acceptable for continued service. The fatigue usage factor for the most limiting tube remaining in service is calculated to be 0.83 for a 40 year operating period with the reference fuel cycle parameters.

3.0 CONCLUSION

The staff concludes that the actions taken by the licensee resolve the issues identified in Bulletin 88-02 and are, therefore, acceptable. Consistent with staff finding No. 11 in the attached SER, the above findings are subject to the development of administrative controls by the licensee to ensure that updated stress ratio and fatigue usage calculations are performed in the event of any significant changes to the steam generator operating parameters (e.g, steam pressure and flow, circulation ratio) relative to the reference parameters assumed in the analyses for Sequoyah Units 1 and 2 in WCAP-12289 and WCAP-12290. The licensee will update the Sequoyah Final Safety Analysis Report with sufficient information on its response to the bulletin to ensure changes to steam generators are properly reviewed. This is acceptable to the staff.

4.0 REFERENCE

1. Safety Evaluation Report, "Evaluation of Westinghouse Methodology to Address Item C.2 of NRC Bulletin 88-02" which was transmitted to Westinghouse by NRC letter dated October 2, 1989.

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Dated: February 15, 1990