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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO CLOSEOUT OF BULLETIN 88-02 ISSUES (MPA B-099)

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (FARLEY)

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

Alabama Power Company (the licensee) submitted its response to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes" by letters dated March 23 and August 29, 1988. 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 Farley steam generators exhibit evidence of denting at the uppermost support plate. Accordingly, Items C.1 and C.2 of the bulletin are applicable to Farley.

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 23, 1988 letter. 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.

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-11875 (Proprietary Version) and WCAP-11876 (Non-Proprietary Version) which were submitted with the licensee's letter dated August 29, 1988. These reports describe the analyses which were conducted to establish the susceptibility of the Farley 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 Reference 1. The staff concluded in Reference 1 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 Safety Evaluation herein incorporates the staff's generic Reference 1 evaluation by reference.

The analyses for the Farley steam generators 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 Reference 1.

Stability ratios for the Farley steam generator tubes were determined from detailed analyses performed for another plant with Model 51 steam generators with very similar thermal-hydraulic conditions as Farley. These analyses included tube instability analyses performed with the FASTVIB computer code using thermal-hydraulic input from a 3-D ATHOS model. The stability ratio results were adjusted upward by 2.4% to reflect the slightly different thermal-hydraulic conditions at Farley. The adjustment factor was calculated on the basis of a comparison of stability ratio estimates for both plants determined by 1-D analysis.

The original anti-vibration bar (AVB) supports in the Farley steam generators were replaced with a modified design between 1985 and 1987. Evaluation of the eddy current data revealed the modified AVBs to have very uniform insertion depths (i.e., depth variations are typically within one tube pitch) with the bottom of the AVBs being located between rows 11 and 12. Because of the relatively uniform insertion depths, flow peaking factors for the Farley steam generator tubes were determined to be small compared to peaking factors which exist for certain tubes at other plants (including the tube that failed at North Anna) which continue to employ the original AVB design.

The analyses documented in WCAP-11875 show that all presently unsupported tubes in the Farley steam generators satisfy the Westinghouse stress ratio criterion. The fatigue usage factor for the most limiting tube is calculated to be 0.4 from the time of AVB replacement (1987) to the projected 40 year lifetime of the plant. Fatigue usage prior to AVB replacement was not calculated since the old AVB positions and associated peaking factors were not determined. Westinghouse notes, however, that experience at other plants indicates that few tubes in rows 10 and 11 were likely to have been not supported by AVB's prior to modification of the AVBs. Further, it has been Westinghouse's experience that only a small portion of unsupported tubes have exhibited significant flow peaking factors. Thus, Westinghouse concludes that the probability of Row 11 and smaller Farley tubes having a significant prior fatigue usage and/or a large crack is very small. Even if a crack were to initiate during subsequent operation, Westinghouse concludes that the crack growth rate would be insignificant since even a 30° through-wall circumferential crack would not result in crack tip stresses that are above the threshold for crack propagation.

Finally, the staff notes that an N-16 monitor has been installed on all three steam lines at Farley Unit 2 and that similar monitors were installed at Farley

Unit 1 during the 1989 refueling outage which began on September 23, 1989. The staff considers this to be a significant enhancement of the licensee's capability to detect and monitor rapidly changing leak rates. (Note that the N-16 monitor is not part of the interim measures implemented by the licensee in response to Item C.1 of Bulletin 88-02). Readings from the N-16 monitor are automatically converted to primary-to-secondary leakage rate which is continuously displayed in the control room. The leak rates from the N-16 monitor are logged each shift. In addition, alarms with setpoints corresponding to three different leak rate thresholds provide added assurance that the operators will be alerted to a rapidly increasing primary-to-secondary leak rate.

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 Reference 1, the licensee's staff has advised us that existing 10 CFR 50.59 administrative procedures would ensure that updated stress ratio and fatigue usage calculations would be performed in the event of any significant changes to the steam generator operating parameters (e.g., steam pressure, flow, and circulation ratio) relative to the reference parameters assumed in the Farley analyses.

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 letter dated October 2, 1987.

Principal Contributor: E. Murphy

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