UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-259/84-01, 50-260/84-01, and 50-296/84-01

Licensee: Tennessee Valley Authority
500A Chestnut Street
Chattanooga, TN 37401

Docket Nos.: 50-259, 50-260 and 50-296

License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry

Inspection at TVA Engineering Design Office at Knoxville, Tennessee

Inspector: for W. P. Ang	Jan 20, 1984 Date Signed
Approved by: J. J. Blake, Section Chief, Engineering Program Branch, Division of Engineering and Operational Programs	1/20/84 Date Signed

SUMMARY

Inspection on January 4 - 6, 1984

Areas Inspected

This routine, announced inspection involved 17 inspector-hours at TVA Engineering Design Office at Knoxville, Tennessee in the area of design activities regarding licensee event report BFRO-50-296/83026 - Emergency Diesel Generator Coolers.

Results

Of the area inspected, one apparent violation was found (Incomplete Design Change Analysis, paragraph 5d).

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *D. W. Wilson, Head Nuclear Engineer, EN DES
   *D. L. Williams, Licensing Supervisor, Nuclear Engineering Branch-ENDES
   *J. M. Marshall, Jr., Civil Design Project Engineer, EN DES
   *N. R. Beasley, Mechanical Design Project Engineer, EN DES
   *F. E. Denny, Browns Ferry Section Supervisor, OQA
   *T. Barkalow, Nuclear Engineer, EN DES
   *S. Davidson, Mechanical Engineer, EN DES
   W. Joest, Metallurgical Engineer, EN DES
   J. Stellern, Mechanical Engineer, EN DES
   *R. Tucker, Mechanical Engineer, EN DES
   **J. Domer, Supervisor, BWR Section
   *J. Walcott, Supervisor, Nuclear Power

   *Attended exit interview
   **Participated in exit interview by telephone

2. Exit Interview

   The inspection scope and findings were summarized on January 6, 1984, with those persons indicated in paragraph 1 above. The inspection scope and findings were discussed with the licensee. The licensee acknowledged the inspection findings listed below.

   Violation 259, 260, 296/84-01-01, Incomplete Design Change Analysis, paragraph 5d.

3. Licensee Action on Previous Enforcement Matters

   Not inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Design Activities Regarding Licensee Event Report, (LER) BFRO-50-296/83026 Revision 4 - Emergency Diesel Generator Coolers

   On November 25, 1983, the licensee submitted Revision 4 of LER 83026. The licensee reported that on April 11, 1983, a diesel generator cooling water heat exchanger head was found to be cracked subsequent to modification work on the heat exchanger. The report stated that the head was replaced with a newly fabricated head, tested and returned to service. The licensee further reported that the 16 emergency diesel cooling water heat exchangers were manufactured by Young Radiator Company to standards for 75 psig design. The licensee reported that maximum operating pressure for the raw cooling water to the heat exchangers was 135 psig. The licensee reported that EECW side of the heat exchangers had been previously tested to 142-168 psig. The licensee further reported that a design analysis on the diesel generator cooler heat exchanger shell flanges, tube sheet, and tubes indicated rated pressures of 180 psig, 190 psig and 1200 psig respectively. Based on the analysis and hydrostatic test data the licensee concluded that there was no immediate operating concern.

An inspection of the licensee's design activities related to LER 83026R4 was performed to verify licensee compliance with NRC requirements and licensee commitments. LER 83026R4 was reviewed and discussed with the licensee.

a. Diesel Generator Cooling Water Heat Exchanger Calculations - Design engineers calculations to qualify the 75 psig design heat exchangers for potential service of 135 psig were reviewed. The following items were noted.

   (1) Shell and Bonnet (Head) flanges were qualified by comparing 12 inch nominal pipe size 150 pound ANSI and American Water Works Association flange thickness with the 13 inch I.D. cooler flanges. This was not a conservative comparison since the 13 inch I.D. cooler flange would require larger flange dimensions.

   (2) Shell and Bonnet flange bolting were similarly compared with bolts for 12" NPS flanges. The flange bolting pattern, bolt hole size, and outside diameter were not evaluated for adequate flange material.

   (3) Bonnet (Head) pressure boundary thickness was similarly compared with dimensions for 12" NPS flanges. No evaluation was performed of the total heat exchanger heads (Bonnets and Flanges as one piece).

   (4) Calculations (EN DES calculation 83 1116 301, were performed for the evaluation of the tube sheet using the Tubular Exchanger Manufacturers Association's (TEMA) standards. However, the tube to tube sheet rolled end joints were not evaluated.

   (5) The load on the shell to tube sheet flange weld transmitted through the bolts from the heads was not evaluated.

   (6) Nozzle loads were not evaluated for the existing heads. EN DES calculation BWP 83 1123 101 is based on two replacement heads.

   (7) Calculations (EN DES calculation MEB 83 0422 301) were performed for the replacement head. However, the welds for the inlet/outlet flow partition of the head were evaluated by engineering judgement.

b. The diesel generator cooling water heat exchanger design problem was identified on NCR BFNBWP 8311. A review of the NCR showed that other components were identified with similar design discrepancies. The NCR was reviewed and discussed with the licensee. The following items were noted.

(1) Only heat exchangers on the EECW system were evaluated. Other components and other systems were not evaluated.

(2) The NCR evaluated a cracked upper head on a RHR pump seal heat exchanger. Functional impairment was determined to be unlikely since adequate cooling would be provided even if the cooling water escaped through the upper head.

(3) The inspector determined from the licensee that the RHR pump seal water coolers are designed for 150 psig but could experience a maximum of 200 psig pressure during operation. This condition was not noted on the NCR 8311 evaluation. Furthermore, the heat exchanger bottom head and shell had not been analyzed. The inspector was informed that EN DES boiling water project recommended testing the heat exchangers to 200 psig and analyzing the heat exchangers for 200 psig. However, EN DES had not yet issued the memorandum to the Office of Nuclear Power with these recommendations.

(4) The licensee informed the inspector that the upper head on one of the RHR pump seal water coolers had cracked in 1978 and has still not been replaced. Replacement heat exchanger for the 12 Browns Ferry RHR Pump Seal Water coolers had been purchased but was also designed for 150 psig. No NCR had been written on this condition.

c. LER BFRO-50-296/83026R4 detailed report and analysis attributed the failure of a diesel generator heat exchanger bonnet (Head) to "bolt-up" stress and a possible manufacturing defect. During the inspection, the licensee reported the failure of a second bonnet (head) during reinstallation. No evaluations were available at EN DES for the potential "manufacturing defects" on the remaining 14 heat exchangers. No "bolt-up" stresses or other assembly stresses have been taken into account in the evaluation of both the existing heads and the new heads. No engineering limitations for torquing of the bolts or for fit-up of the heads have been generated by EN DES.

d. The evaluation of the EECW heat exchangers noted on NCR 8311 was performed using TVA EN DES Engineering Procedure (EP) 1.26 - Nonconformances - Reporting and Handling by EN DES. Subsequent to the evaluation, on December 12, 1983, EN DES EP 1.48 was issued to provide a procedure for the preparation of failure evaluations/engineering reports of deficient conditions for operating nuclear plants. The NCR 8311, EECW heat exchangers resolution included failure evaluations of the various heat exchangers similar to the failure evaluation process provided for by EP 1.48. A review of the EECW heat exchanger failure evaluations and EP 1.48 revealed the following items.

(1) Upon receipt of significant deficient condition report, EP 1.48 paragraph 2.4 requires the preparation of a failure evaluation/engineering report within ten working days. The report, Attachment 1 of EP 1.48 requires classification of the deficient condition in one of three categories - (I) Acceptable, (II) Not Acceptable for some design condition but functional impairment is not likely, or, (III) Unable to perform its design function without corrective action. EP 1.48 does not require notification of the Office of Nuclear Power or of the plant upon receipt of a significant nonconformance nor after determination of a Category (II) condition. Upon determination of a Category (II) condition, EP 1.48 paragraph 4.7.b allows an additional 20 working days for determining whether the condition is Category (I) or (III). At the end of the 30 working days, EP 1.48 requires notification of the Office of Nuclear Power. However, a note to EP 1.48 paragraph 4.7.b allows the condition to be classified as Category (II) indefinitely upon notification of the Office of Nuclear Power. The Diesel Generator Cooling Water Heat Exchangers and the RHR Pump Seal Coolers under pressure design condition were determined to be Category (II) conditions on June 28, 1983 and June 23, 1983, respectively, and are still classified as Category (II) conditions.

(2) EP 1.48 allows Category (II) conditions to be accepted without design control measures commensurate with those applied to the original design.

10 CFR 50 Appendix "B" Criterion III requires in part that design changes shall be subject to design control measures commensurate with those applied to the original design. Continued plant operation with components in a degraded design condition appears to be a design change for the affected components. EP 1.48 and the incomplete analysis of the diesel generator cooling water heat exchangers and RHR pump seal water heat exchangers appear to be in violation of 10 CFR 50 Appendix "B" Criterion III in that design control measures commensurate with the original design were not required by EP 1.48 nor were performed for the noted components. This shall be identified as violation 84-01-01 "Incomplete Design Change Analysis."