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APR 08 1993

O. J. "Ike" Zeringue
Vice President, Browns Ferry Nuclear Plant

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter Of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI INSERVICE SYSTEM PRESSURE TEST PROGRAM

In accordance with 10 CFR 50.55a(g)(iii), BFN submits the attached Request for Relief from the specified Section XI system pressure testing requirements of the 1986 Edition of the ASME Boiler and Pressure Code for NRC review.

The impractical aspect of the specified ASME Section XI requirements was identified during the recent preparations to perform the code Class 1 leakage test at the completion of the current Unit 2, cycle 6 refueling outage. With Unit 2 scheduled to restart on May 8, 1993, we request an expeditious review and approval of the attached Request for Relief.

If you have any questions, please telephone G. D. Pierce, Interim Manager of Site Licensing, at (205) 729-7566.

Sincerely,

O. J. Zeringue

Enclosure
cc: See page 2

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Enclosure
cc (Enclosure):
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REQUEST FOR RELIEF SPT-4

System:	Control Rod Drive Hydraulic (85)
Drawing:	47E820-6
Components	Control rod drive housing cap screws (185 CRDs per unit, 8 cap screws per housing-to-flange connection)
Class:	1
Function:	Connects CRD housing to the reactor pressure vessel CRD nozzle flange
Impractical Test Requirements:	IWA-5250(a) - The source of leakage detected during the conduct of a system pressure test shall be located and evaluated by the Owner for corrective measures as follows: (2) if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.
Basis for Relief:	In accordance with the requirements of Table IWB-2500-1, Examination Category B-P, Item No. B15.10, a leakage test of the reactor pressure vessel pressure retaining boundary is conducted prior to plant startup following each reactor refueling outage. The leakage test is conducted at nominal system pressure (1005 psig at the RPV dome) immediately prior to the startup of the unit. This examination includes the 185 CRD housing connections located on the bottom of the reactor pressure vessel. During repressurization following unit refueling, it is not uncommon to have small amounts of leakage at some of the CRD housing connections. Compliance with subparagraph IWA-5250(a)(2) in the event of leakage at a CRD housing connection would result in an extreme hardship which is not commensurate with the increased level of safety that would be achieved. The hardship is due to the requirement that for any CRD connectors where leakage is detected during the pressure test the housing connector cap screws be removed and examined. This requires that the reactor pressure vessel (RPV) be depressurized and the CRD housing restraint be removed to permit removal of the CRD. Due to the torquing sequence requirements the cap screws cannot be removed individually for examination.

The NSSS supplier (General Electric) has informed BWR owners that leakage from these cap screw connections is a common occurrence, and in most instances leakage stops within 8 hours of the connection being pressurized to 1000 psig.

Industry experience as documented in GE Nuclear Energy Services Information Letter No. 483, Revision 2, dated August 5, 1992, has shown that these CRD cap screws are susceptible to stress corrosion cracking. Due to this susceptibility, GE has recommended the replacement of these cap screws with a new design and higher strength material cap screw, and a new design washer to facilitate drainage.

This new design is being incorporated at Browns Ferry. The cap screws for 26 CRD housing units have been changed during the Unit 2 cycle 6 refueling outage. The remaining cap screws will be changed as the CRDs are replaced or maintenance is performed. The cap screws which are removed are examined in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. None of the cap screws thus far examined have exhibited indications of stress corrosion cracking. General Electric has determined that based on the evaluation of crack data, structural integrity and plant safety are not effected by this situation. This evaluation is based in part on the following:

1) three uniformly distributed uncracked cap screws are capable of supporting the CRD loads, and the probability that through-wall cracks will occur in five or more cap screws on a single CRD housing is extremely small;

2) if such a failure were to occur leakage at the connection would precede failure, and the leak detection system and drywell temperature monitoring system would detect this leakage;

3) the CRD support structure under the reactor vessel would allow the CRD to drop a maximum of one inch;

4) the evaluation of the loss of one CRD has been considered in the plant safety analysis report.

Alternate Testing:

During the Class 1 component leakage test following refueling, all leakage from the CRD housing connections will be documented and evaluated based on the General Electric Company recommendations.

The VT-3 examination of all eight cap screws at CRD housing connections where leakage is detected during the leakage test will be deferred to the next refueling outage.