August 5, 2004

10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 LICENSE RENEWAL APPLICATION – UPDATE OF APPLICATION SECTIONS 4.2 and 4.3 TO REFLECT EXTENDED POWER UPRATE CONDITIONS – SUPPLEMENTAL INFORMATION

By letter dated May 28, 2004, TVA submitted revised pages for Sections 4.2 and 4.3 of the BFN License Renewal application that reflect reactor pressure vessel fatigue reanalyses at 122 percent of original licensed thermal power (pages 4.2-3, 4.2-8, 4.2-9, 4.2-13, 4.3-1, 4.3-2, and 4.3-3). While the revised pages were technically correct the page numbers for pages 4.3-1, 4.3-2, and 4.3-3 were later found to be incorrect in the NRC Agencywide Documents Access and Management System (ADAMS) copy.

The enclosure to this letter contains replacement pages 4.3-1, 4.3-2, and 4.3-3 of the License Renewal application revised to show the correct page numbers. Please replace the last two pages (4.2-1, and 4.2-2) of Enclosure 1 of TVA’s May 28, 2004, letter with the three pages enclosed.

This letter contains no new regulatory commitments.

If you have any questions about this information, please contact Ken Brune, Browns Ferry License Renewal Project Manager, at (423) 751-8421.

I declare under penalty of perjury that the forgoing is true and correct. Executed on this fifth day of August, 2004.

Sincerely,

Original signed by:

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Enclosure:
cc: See page 3

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, AND 3

LICENSE RENEWAL APPLICATION,

UPDATE OF APPLICATION SECTIONS 4.2 and 4.3 TO REFLECT

EXTENDED POWER UPRATE (EPU) CONDITIONS,

SUPPLEMENTAL INFORMATION FOR SECTION 4.3

(SEE ATTACHED)

BFN License Renewal Application

Replacement Pages (4.3-1, 4.3-2, and 4.3-3)

4.3 METAL FATIGUE

A cyclically loaded metal component may fail because of fatigue even though the cyclic stresses are considerably less than the static design limit. Some design codes (such as the ASME Boiler and Pressure Vessel Code and the ANSI piping codes) therefore contain explicit metal fatigue calculations or design limits. Cyclic or fatigue design of other components may not be to these codes, but may use similar methods. These analyses, calculations, and designs to cycle count limits or to fatigue usage factor limits may be TLAAs.

BFN Fatigue analyses are presented in the following groupings:

- Reactor Vessel Fatigue Analyses
- Fatigue Analysis of Reactor Vessel Internals
- Piping and Component Fatigue Analysis
- Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

4.3.1 Reactor Vessel Fatigue Analyses

Summary Description

Reactor vessel fatigue analyses of the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs depend on assumed numbers and severity of normal and upset-event pressure and thermal operating cycles to predict end-of-life fatigue usage factors.

These assumed cycle counts and fatigue usage factors are based on 40 years of operation. Calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations. The reactor vessel fatigue analyses are TLAAs.

Analysis

The original reactor pressure vessel report included a fatigue analysis for the reactor vessel components based on a set of design basis duty cycles. These duty cycles are listed in Section 4.2.5 of the BFN UFSAR. The original 40-year analyses demonstrated that the cumulative usage factors (CUF) for the critical components would remain below the ASME Code Section III allowable value of 1.0.

A reanalysis was performed for reactor vessel cumulative fatigue usage factors for Extended Power Uprate (EPU) and Maximum Extended Load Line Limit Analysis (MELLLA+) conditions. A subset of the bounding reactor vessel components was evaluated as a part of this analysis. The resulting fatigue CUFs for these limiting components supersede the values determined in the original reactor vessel analyses.

The original code analysis of the reactor vessel included fatigue analysis of the Feedwater (FW) and control rod drive (CRD) hydraulic system return line nozzles. After several years of operation, it was discovered that both the CRD hydraulic system return line nozzles and the FW nozzles were subject to cracking caused by a number of factors including rapid thermal cycling. Consequently, the CRD hydraulic system return line nozzles were capped and removed from service. As such, they are no longer subject to rapid thermal aging. A reanalysis was later performed on the FW nozzles along with modifications to reduce or eliminate the causes. This revised analysis did not include the effects from rapid thermal cycling as the FW System design and operation is bounded by a generic BWR Owners Group guidance. BFN follows the improved BWR Owners Group inspection and management methods.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation; and 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

For the period of extended operation, the fatigue usage factors for the limiting components have been reevaluated based on EPU and MELLLA+ conditions. Several components have 60-year CUFs greater than the ASME Code allowable of 1.0. The results of the evaluation are shown in Table 4.3.1.1.

Table 4.3.1.1: Fatigue Evaluation Results (Note 1)

Component	Computed Fatigue Usage Factor for 60 years (Note 2)	Included in Fatigue Monitoring Program (B.3.2) (Note 3)
Recirculation Outlet Nozzle	1.17	Y (NUREG/CR-6260 component)
Recirculation Inlet Nozzle	0.64	Y (NUREG/CR-6260 component)
Feedwater Nozzle	1.50	Y (NUREG/CR-6260 component)
Core Spray Nozzle	0.11	Y (NUREG/CR-6260 component)
Support Skirt	1.36	Y
Closure Stud Bolts	1.14	Y
Vessel Shell	0.048	Y (NUREG/CR-6260 component)

Notes:

1. These results do not account for environmental fatigue effects.
2. The usage factors are bounding for BFN Units 1, 2, and 3
3. The components listed as a “NUREG/CR-6260 component” will be monitored for GSI –190. (Section 4.3.4).

4.3.2 Fatigue Analysis of Reactor Vessel Internals

Summary Description

The original fatigue analysis of the reactor internals was performed using the ASME Boiler and Pressure Vessel Code, Section III, as a guide. The method of analysis used to determine the cumulative fatigue usage is described in [8], which determined that the most significant fatigue loading occurs at the jet pump diffuser to baffle plate weld location; this was the only fatigue analysis performed. The original 40 year calculation showed a CUF of 0.35, less than the ASME allowable of 1.0. Since this analysis used a number of cycles for a 40 year life, it is considered a TLAA. In addition, BFN Unit 3 installed a repair at the T-box location to address cracking, as well as a lower sectional replacement in the core spray line. Fatigue calculations were performed for several components using ASME Section III as a guide, since the core spray line is not a ASME Section III component. Since these analyses were based on a 40 year life, they are considered TLAAs.