May 28, 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

By letter dated December 31, 2003, Tennessee Valley Authority (TVA) submitted an application to renew the operating licenses for BFN Units 1, 2, and 3. Since TVA plans to submit license amendment requests for extended power uprate (120% of original licensed thermal power), the application was prepared conservatively assuming the extended power uprate for all three units. However, the fatigue analyses for the reactor vessel were performed at 120% rather than 122% as stipulated in Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants". As such, the application was submitted assuming a bounding power level to the current licensing basis. This was discussed with the NRC Staff in a meeting on January 28, 2004. In that meeting, TVA stated that the fatigue analyses would be performed at the extended power uprate power level and the application updated accordingly by June, 2004.

The fatigue analyses have been revised to assume the 122% reactor thermal power level as described above. The re-analyses did not change the Computed Fatigue Usage Factors that were reported in Table 4.3.1.1 for Units 2 and 3 in the December 31, 2003 application, and they confirmed that the Unit 2 and 3 bounding values also bound Unit 1.

Enclosure 1 contains the revised pages of Sections 4.2 and 4.3 of the application that reflect this change (pages 4.2-3, 4.2-8, 4.2-9, 4.2-13, 4.3-1, 4.3-2, and 4.3-3). It should be noted that the revision to Table 4.3.1.1 (page 4.3-3) deleted the values for Unit 1 that were included in the original application. The Unit 1 values were deleted since the values for Units 2 and 3 were confirmed to bound the Unit 1 values. The changes to Section 4.2 are editorial. Pages 4.2-3, 4.2-8 and 4.2-13 were revised to add the statement, "The fluence values are based on EPU conditions."

Enclosure 2 contains the marked up pages showing the revisions for ease of review.

This update makes all aspects of the license renewal application consistent with extended power uprate conditions for all three units. As a result, no additional reviews should be necessary if the license amendment requests for power uprates are approved prior to issuance of the renewed license.

This letter contains no new commitments.

If you have any questions about this information, please contact Gary Adkins, Browns Ferry License Renewal Project Manager, at (423) 751-4363.

I declare under penalty of perjury that the forgoing is true and correct. Executed on this 28th day of May, 2004.

Sincerely,

Original signed by:

T. E. Abney
Manager of Licensing
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Enclosures:
1. Revised Pages
2. Marked up Pages

cc: See page 3

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ENCLOSURE 1

Disposition: 10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

Fluences were calculated for the reactor vessels for the extended 60-year (54 EFPY (Effective Full-Power Year), for Unit 1; 52 EFPY for Units 2 and 3) licensed operating periods, using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation."[2] One bounding fluence calculation was performed for Units 1, 2 and 3. The fluence values are based on EPU conditions. Peak fluences were calculated at the vessel inner surface (inner diameter), for purposes of evaluating USE. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter using Equation 3 from Paragraph 1.1 of Regulatory Guide 1.99, Revision 2. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G Sub-article G-2120 as the maximum postulated defect depth.

The end-of-life USE was evaluated by an equivalent margin analysis using the 54 EFPY calculated fluence for Unit 1 and the 52 EFPY calculated fluence for Units 2 and 3. As described in the SER to Boiling Water Reactor Vessel and Internals Project (BWRVIP-74-A)3, the percent reduction in USE for the limiting BWR/3-6 plates and BWR/2-6 welds are 23.5% and 39% respectively. LRA summary Tables 4.2.1.1 through 4.2.1.6 provide results of the equivalent margin analysis for limiting welds and plates on the three BFN reactor vessels. The results show that the limiting USE EMA percent is less than the BWRVIP-74-A EMA percent acceptance criterion in all cases, and is therefore acceptable.

2 Approved by the NRC in letter, S.A. Richard, USNRC to J.F. Klapproth, GE-NE, "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)", MFN 01-050, September 14, 2001.

3 NRC letter (Accession No. ML012920549) to BWRVIP, "Acceptance Criteria for Referencing of EPRI Proprietary Report TR-113596, 'BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Inspection and Flaw Evaluation Guidelines (BWRVIP-74)' and Appendix A, 'Demonstration of Compliance with the Technical Information Requirements of the License Renewal rule (10 CFR 54.21),"' October 18, 2001

4.2.2 Adjusted Reference Temperature for Reactor Vessel Materials due to Neutron Embrittlement

Summary Description

The initial RT_{NDT}, nil-ductility reference temperature, is the temperature at which a non-irradiated metal (ferritic steel) changes its fracture characteristics from ductile to brittle behavior. The RT_{NDT} was evaluated according to the procedures in the ASME Code, Paragraph NB-2331. Neutron embrittlement raises the initial nil-ductility reference temperature. 10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the vessel. The shift to the initial nil-ductility reference temperature (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The adjusted reference temperature (ART) is defined as RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in Regulatory Guide 1.99, Revision 2. The P-T curves are developed from the ARTs for the vessel materials. These are determined by the unirradiated RT_{NDT} and by the ΔRT_{NDT} calculations for the licensed operating period. Regulatory Guide 1.99 defines the calculation methods for ΔRT_{NDT}, ART, and end-of-life USE.

The ΔRT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a). As such, they are TLAAs.

Analysis

As described in UFSAR Section 4.2, the reactor vessels were designed for a 40-year life with an assumed neutron exposure of less than 10^{19} n/cm^2 from energies exceeding 1 MeV. The current licensing basis calculations use realistic calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year term for all three units. The ΔRT_{NDT} values were determined using the embrittlement correlations defined in Regulatory Guide 1.99, Revision 2.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

Fluences were calculated for the reactor vessels for the extended 60-year (54 EFPY for (Unit 1); 52 EFPY for Units 2 and 3) licensed operating periods using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation".4 One bounding calculation was performed for the three BFN reactor vessels. The fluence values are based on EPU conditions. Peak fluences were calculated at the vessel inner surface (inner diameter), for purposes of evaluating USE and ART. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter using Equation 3 from Paragraph 1.1 of Regulatory Guide 1.99, Revision 2. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G Sub-article G-2120 as the maximum postulated defect depth.

The 54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3) ΔRT_{NDT} for beltline materials were calculated based on the embrittlement correlation found in Regulatory Guide 1.99, Revision 2. The peak fluence, $\Delta RT_{NDT,}$ and ART values for the 60 year (54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3)) license operating period are presented in LRA Table 4.2.2-1. This table shows that the limiting ARTs allow P-T limits that will provide reasonable operational flexibility.

4 Approved by the NRC in letter, S.A. Richard USNRC to J.F. Klapproth, GE-NE, "Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891)", MFN 01-050, September 14, 2001.

Table 4.2.2.1 60-Year Analysis Results for BFN Units 1, 2 & 3

Parameter	Unit 1 (54 EFPY)	Unit 2 (52 EFPY)	Unit 3 (52 EFPY)
Peak Surface Fluence (n/cm^2)	1.95 x 10^{18}	2.3 x 10^{18}	2.3 x 10^{18}
1/4T Fluence (n/cm^2)	1.35 x10^{18}	1.59 x10^{18}	1.59 x10^{18}
RT_{NDT} (°F)	88	73	73
ART (°F)	167.7	157	157

4.2.3 Reflood Thermal Shock Analysis of the Reactor Vessel

Summary Description

The UFSAR Section 3.3.5 includes an end-of-life thermal shock analysis performed on the reactor vessels for a design basis loss of coolant accident (LOCA) followed by a low-pressure coolant injection. The effects of embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. This analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

Analysis

For the current operating period, a thermal shock analysis was originally performed on the reactor vessel components. The analysis assumed a design basis LOCA followed by a low-pressure coolant injection accounting for the full effects of neutron embrittlement at the end of the current license term of 40 years. The analysis showed that the total maximum vessel irradiation (1MeV) at the mid-core inside of the vessel to be 2.4 x 10^{17} n/cm^2 which was below the threshold level of any nil-ductility temperature shift for the vessel material. As a result, it was concluded that the irradiation effects on all locations of the reactor vessels are not limiting. However, this analysis only bounded 40 years of operation.

Disposition: 10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation.

extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for the period of extended operation and approval by the NRC to extend this relief request.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities.9 Three key assumptions of the PFM analysis are: 1) the neutron fluence was the estimated end–of-life mean fluence, 2) the chemistry values are mean values based on vessel types, and 3) the potential for beyond-design-basis events is considered. LRA Table 4.2.6.1 provides a comparison of the BFN Units 2 and 3 reactor vessel limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 for the first two key assumptions. The fluence values are based on EPU conditions. Data provided in LRA Table 4.2.6.1 was supplied from Tables 2.6.4 and 2.6.5 of the Final Safety Evaluation of the BWRVIP-05 Report.

For Units 2 and 3, the fluence is equivalent to the NRC analysis. However, the BFN Units 2 and 3 weld materials have significantly lower copper values (0.09 vs. 0.31) than those used in the NRC analysis. Hence, there is a significantly smaller chemistry factor. As a result, the shifts in reference temperature for Units 2 and 3 are lower than the 64 EFPY shift from the NRC SER analysis. In addition, the unirradiated reference temperatures for both units are significantly lower. The combination of unirradiated reference temperature ($RT_{NDT(U)}$) and shift (ΔRT_{NDT} w/o margin) yields adjusted reference temperatures for Units 2 and 3 that are considerably lower than the NRC mean analysis values.

Therefore, the RPV shell weld embrittlement due to fluence has a negligible effect on the probabilities of RPV shell weld failure. The Mean RT_{NDT} values for Units 2 and 3 at 52 EFPY are bounded by the 64 EFPY Mean RT_{NDT} provided by the NRC. Although a conditional failure probability has not been calculated, the fact that the BFN values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the BFN RPV conditional failure probability is bounded by the NRC analysis.

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when BFN requested that the BWRVIP-05 technical alternative be used for the current term for Units 2 and 3.

An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation.

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9 NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, "Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report," (TAC No. M93925), July 28, 1998.

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.

ENCLOSURE 2