November 8, 2004

10 CFR 50.55a(a)(3)(i)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-001

Gentlemen:

In the Matter of	)	Docket No. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI AND AUGMENTED INSPECTIONS – REVISION TO REQUEST FOR RELIEF, 1-ISI-19, REGARDING REACTOR PRESSURE VESSEL (RPV) CIRCUMFERENTIAL SHELL WELD EXAMINATIONS

In recent discussions, the NRC Staff requested that TVA revise BFN Unit 1 Relief Request 1-ISI-19, submitted by letter dated May 12, 2004 (Reference 1). Relief Request 1-ISI-19 requests relief, based on the guidance contained in NRC Generic Letter 98-05 (Reference 2), from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of the BFN Unit 1 reactor vessel circumferential shell welds.

Relief Request 1-ISI-19 was based in part on an end-of-license reactor vessel circumferential beltline weld (Weld C-1-2) nil-ductility transition temperature (RT_{NDT}) calculated at ¼T (one quarter thickness) of the reactor vessel wall. Per discussions with the NRC Staff, the NRC requires the end-of-license reactor vessel circumferential beltline weld inner surface RT_{NDT} (RT_{NDT} calculated based on the reactor vessel inner surface fluence) to complete its acceptance review. Accordingly, TVA has recalculated Weld C-1-2 RT_{NDT} at the reactor vessel inner surface. The information requested and its supporting assumptions are provided below.

The Weld C-1-2 RT_{NDT} reported in Reference 1 and in TVA's response to an NRC Staff Request for Additional Information

(Reference 3), was based on assumed BFN Unit 1 operation of 32 Effective Full Power Years (EFPY) at Extended Power Uprate (EPU) conditions. As discussed in Reference 1, these assumptions were very conservative because BFN Unit 1 is expected to accumulate less than 14 EFPY of reactor operation by the end of its current license due to its extended shutdown, with less than 8 of the 14 EFPY at EPU conditions. Calculation of Weld C-1-2 RT_{NDT} at the inner vessel surface assuming a fluence representing 32 EFPY of operation at EPU conditions would yield an RT_{NDT} that would exceed the NRC's criteria given in Table 2.6-4 of Reference 4.

Therefore, TVA has calculated the Weld C-1-2 inner surface RT_{NDT} assuming 16 EFPY of operation at EPU conditions. This assumption is still very conservative, and demonstrates that the associated NRC acceptance criteria for granting the requested relief are met. The results of this calculation, along with key inputs are provided below.

BFN Unit 1 Weld C-1-2 Mean ID RT_{NDT} at 16 EFPY	
RPV manufacturer	B&W
Existing vessel exposure	6.15 EFPY
Current license expiration	December 20, 2013
Assumed end-of-license exposure[1]	16 EFPY
RPV ID peak flux (>1.0 Mev)	1.40E9 n/cm^2-sec
RPV ID peak surface fluence	7.07E17 n/cm^2
Ratio peak/weld C-1-2 location	0.81
Weld C-1-2 peak surface fluence[2]	5.78E17 n/cm^2
Cu (Wt %)	0.27%
Ni (Wt %)	0.60%
Chemistry Factor	184
Weld C-1-2 ID surface ΔRT_{NDT} at 16 EFPY	58.2°F
Weld C-1-2 initial RT_{NDT}	20°F
Weld C-1-2 mean ID surface RT_{NDT} at 16 EFPY	78.2°F

1. This value is conservative, as less than 14 EFPY of operation is expected by the end of the current operating license.

2. Fluence conservatively increased 1%, from 5.73E17 n/cm^2 to 5.78E17 n/cm^2 to account for planned future operation in an expanded operating domain (MELLLA+ operation).

As shown in the table above, TVA's revised analysis results in a conservatively calculated Reactor Vessel Beltline Circumferential Weld C-1-2 inner surface end-of-license mean RT_{NDT} of 78.2°F. This value is well below the end-of-license mean RT_{NDT} used in the NRC's conditional failure probability assessment for the bounding Babcox & Wilcox reactor vessel of 99.8°F. Accordingly, BFN Unit 1 meets the NRC's criteria for granting the requested relief.

The reactor vessel beltline circumferential weld C-1-2 fluence and RT_{NDT} information provided in this letter supersedes that provided previously in Reference 1 and 3. This information does not alter the information provided in Reference 3 regarding the flux evaluation methodology utilized.

There are no new commitments contained in this letter. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

ORIGINAL SIGNED BY:

T. E. Abney
Manager of Licensing
and Industry Affairs

References:

1. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 – American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections - Request for Relief, 1-ISI-19, Regarding Reactor Pressure Vessel (RPV) Circumferential Shell Welds," dated May 12, 2004.

2. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

3. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - American Society of Mechanical Engineers (ASME) Section XI and Augmented Inspections –

Response to Request for Additional Information - Request
Shell Welds," dated August 13, 2004.

4. NRC Letter, G. C. Lainas (NRC) to C. Terry (BWRVIP), Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), dated July 28, 1998.

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