Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

August 17, 2001

TVA-BFN-TS-414

10 CFR 50.90
10 CFR 50.12
10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2 AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGE NO. 414 - PRESSURE - TEMPERATURE CURVE UPDATE

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to BFN's licenses DPR-52 and DPR-68 to change the Technical Specifications for Units 2 and 3. The proposed change revises the reactor vessel pressure-temperature (P-T) limit curves for both units.

In addition, pursuant to 10 CFR 50.12, TVA is requesting an exemption from the requirements of 10 CFR 50, Appendix G, to allow the use of ASME Code Case N-640 as a basis for these revised curves. Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," permits the use of the plane strain fracture toughness (K_{Ic}) curve instead of the crack arrest fracture toughness (K_{Ia}) curve for RPV materials in determining the P-T limits.

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TVA also used ASME Code Case N-588 as a basis for revising the curves. Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1," permits the use of an alternative procedure for calculating applied stress intensity factors during normal operation and pressure test conditions due to pressure and thermal gradients for axial flaws. This methodology has been incorporated into the ASME Code, 1995 Edition, 1996 Addenda, which is currently endorsed by the U. S. Nuclear Regulatory Commission (NRC). This is the code of record for the Browns Ferry Unit 2 In-Service Inspection (ISI) Program. The Browns Ferry Unit 3 ISI Program code of record is currently the 1989 Edition of the ASME Code. TVA will adopt the methodology from the latest approved edition in order to keep the P-T calculation methodology consistent between the operating BFN units. Consequently, in accordance with 10 CFR 55a(g)(4)(iv), TVA will implement the 1995 Edition, 1996 Addenda of Section XI, Appendix G requirements for Unit 3 in conjunction with the approval of the subject TS change. Therefore, an exemption to allow the use of the Code Case N-588 methodology of calculating stress intensity factors for axial flaws is not required for either Unit 2 or Unit 3.

The proposed P-T curves were developed in accordance with the 1995 Edition with 1996 Addenda, ASME Section XI, Appendix G, 10 CFR 50, Appendix G, and the previously mentioned ASME Code Case N-640. The use of Code Case N-640 as a basis for the proposed P-T curves constitutes an alternative to the requirements of 10 CFR 50, Appendix G. 10 CFR 50.60(b) provides that the NRC may grant alternatives to the requirements of Appendix G by using the procedures for exemption specified in 10 CFR 50.12. Exemptions to use Code Case N-640 have been granted for several other boiling water reactors, including Quad Cities Units 1 and 2 (February 4, 2000); Dresden Units 2 and 3 (August 25, 2000); Hatch Units 1 and 2 (August 29, 2000); Limerick Unit 1 (September 7, 2000), and Pilgrim (April 13, 2001).

The proposed change revises the reactor vessel P-T limits depicted on current TS Figure 3.4.9-1 for Units 2 and 3. The revised P-T limits being requested were calculated using a neutron fluence value of 1.12×10^{18} n/cm^2 at 32 Effective Full Power Years (EFPY). This conservative end-of-life fluence value has been previously approved by NRC in the safety evaluation for amendment 257 to facility operating license no. DPR-52 and amendment 217 to facility operating license no. DPR-68. TVA recognizes that NRC currently has under review a Licensing Topical Report from General Electric which, upon its approval, will result in changes in the methodology of calculating neutron fluence values. TVA plans to utilize this updated methodology, if approved by NRC, for the next P-T curve update.

The current Unit 2 P-T curves are valid to 16 EFPY. The Unit 2 curves are currently projected to expire about January 2003. The current Unit 3 P-T curves are valid to 20 EFPY.

TVA recognizes that it has recently been the staff's practice to approve P-T curves for only one cycle of operation, due to the pending staff review and approval of the revised fluence methodology. TVA is requesting approval of the P-T curves through 19.5 EFPY to avoid unnecessary costs. TVA is currently planning to submit within the next year a license amendment to uprate BFN Units 2 and 3 and is also considering a license renewal application in late 2003. By extending approval of the P-T curves through 19.5 EFPY, analysis for new P-T curves resulting from power uprate or license renewal activities can be planned appropriately to support those projects. Additionally, it is anticipated that the new P-T curves could then be developed using the revised fluence methodology.

TVA requests NRC approval of the requested changes by March 1, 2002, to allow use of the new curves during reactor pressure testing following the Unit 3 Cycle 10 refueling outage.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The BFN Plant Operations Review Committee and the BFN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 2 and 3 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Alabama State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration and is exempt from environmental review. Enclosure 2 contains marked up pages of the appropriate TS for Units 2 and 3. Enclosure 3 contains copies of the revised pages as they would appear following approval of this request. Enclosure 4 contains the information supporting the exemption request for the use of ASME Code Case N-640.

TVA requests that the revised TS be made effective within 30 days of NRC approval. If you have any questions about this change, please telephone me at (256) 729-2636.

Enclosure 5 details the commitment made in this submittal.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Subscribed and sworn to before me
on this 17th day of August 2001.

Barbara A. Blanton
Notary Public
My Commission Expires 09/22/2002

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

The proposed change revises the Units 2 and 3 reactor vessel pressure-temperature (P-T) curves to reflect the results of an analysis which validates the curves for both units to 19.5 Effective Full Power Years (EFPY). The current BFN P-T curves are valid up to 16 EFPY and 20 EFPY for Units 2 and 3, respectively.

The specific changes are described below.

1. TS Figure 3.4.9-1 on page 3.4-29 for Units 2 and 3 is deleted and replaced in its entirety.

2. The last sentence of the notes on current TS Figure 3.4.9-1 on page 3.4-29 for Units 2 and 3 is revised to read as follows:

   The curves allow for shifts in RT_{NDT} of the reactor vessel beltline materials in accordance with Reg. Guide 1.99, Rev. 2, to compensate for radiation embrittlement for 19.5 EFPY.

3. Added "ASME" to the note for "Curve 1" on the TS Figure for additional clarity.

II. REASON FOR THE PROPOSED CHANGE

The present BFN P-T curves are valid up to 16 EFPY for Unit 2 and up to 20 EFPY for Unit 3. Expiration of the Unit 2 curves is expected to occur about January 2003. Since Unit 3 has accrued approximately 10.2 EFPY in its operating history, the Unit 3 curves will not expire in the near future. However, the Unit 3 curves are being updated for consistency with Unit 2 and to allow flexibility for pressure testing of the reactor.

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PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

III. SAFETY ANALYSIS

Background

All components of the reactor coolant system are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. Therefore, P-T limits are established to ensure the reactor coolant system is operated under conditions that preclude brittle failure of the reactor coolant pressure boundary.

10 CFR 50, Appendix G, requires the establishment of these P-T limits for reactor coolant pressure boundary materials. Appendix G also requires an adequate margin to brittle failure be maintained during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P-T limits are acceptance limits themselves, since they preclude operation in an unanalyzed condition. The P-T limits are not derived from Design Basis Accident (DBA) analyses.

The proposed P-T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P-T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

For BFN Units 2 and 3, the P-T limits are specified in Technical Specification Figure 3.4.9-1. The figure contains three separate P-T curves, which define the pressure-temperature limitations for the following reactor operating conditions:

- Curve 1 specifies the P-T limits during primary system hydrostatic and leakage testing,
- Curve 2 specifies the P-T limits during heatup and cooldown when the core is not critical, and
- Curve 3 specifies the P-T limits during operations when the core is critical.

Curve 1 includes P-T restrictions on reactor vessel head boltup. Hydrostatic/leak testing of the reactor vessel is performed in accordance with Curve 1 limitations prior to startup after a refueling outage to verify that the vessel is leak tight. The minimum temperature is established by the P-T curves.

Curve 2, the heatup and cooldown curve, is used for startup and shutdown operations. The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall for heatup and cooldown are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

Curve 3, the operational P-T limit curve, provides operational boundaries during maneuvers at power. The primary system pressure and temperature are monitored and compared to the applicable curve to determine that operation is within the allowable region.

Methodology

The P-T limits are primarily dependent upon the fracture toughness of the vessel ferritic materials. The key parameters which characterize a material's fracture toughness are the reference temperature of nil-ductility transition (RT_{NDT}) and the Upper Shelf Energy (USE). These parameters are defined in 10 CFR 50, Appendix G, and in Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI. These documents also contain the requirements used to establish the P-T operating limits that must be met to avoid brittle fracture.

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PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

Regulatory Guide (RG) 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, provides an acceptable method for calculating P-T limits that satisfy the requirements of 10 CFR 50, Appendix G. TVA has recalculated the P-T curves for BFN Units 2 and 3 based on methodologies that are in accordance with Regulatory Guide 1.99, Revision 2 and ASME Section XI Code Case N-640 using plant-specific material and fluence information. The BFN Units 1, 2 and 3 specific RT_{NDT}, weld material composition, and fluence information have been previously provided by TVA to NRC (see References 1-7).

Principal assumptions for this analysis include:

- 1140 psig inservice system hydrostatic pressure (110% of the normal operating pressure)
- 80% capacity factor for thermal generation which results in 32 EFPY over 40 years of plant operation
- 8.6×10^8 n/cm^2-sec peak neutron flux

Results

A conservative estimate of the neutron flux was utilized to calculate the end of life core neutron fluence. This 32 EFPY fluence value is 1.12×10^{18} n/cm^2. The 19.5 EFPY fluence at the vessel inside surface was determined to be 6.83×10^{17} n/cm^2 for both Unit 2 and Unit 3. The 19.5 EFPY peak 1/4T fluence was calculated to be 4.73×10^{17} n/cm^2 for both units.

The limiting adjusted reference temperature (ART) values of 102.9°F for both Unit 2 and Unit 3 remain well below the 200°F criterion of RG 1.99, Revision 2. All USE values calculated for end of life remain greater than 50 ft-lb. A single set of P-T curves for the heatup and cooldown operating condition at a given EFPY that apply for both the 1/4T and 3/4T locations was developed. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (assumed inside surface flaw) and the 3/4T location (assumed outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4T location is assumed to be tensile for both heatup and cooldown. This results in the approach

of applying the maximum tensile stress at the 1/4T location. This approach is conservative because irradiation effects cause the allowable toughness, K_{IR}, at 1/4T to be less than that at 3/4T for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, well above the heatup/cooldown curve limits.

Tables 1 and 2 contain the data for the composite P-T curves valid to 19.5 EFPY for Units 2 and 3.

Conclusion

The proposed P-T curves have been developed utilizing the methodology of RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" and ASME Section XI Code Case N-640. The regulatory guide provides an allowance for margin to be included in the bounding values of the ART. Use of this methodology ensures that adequate safety margins are maintained. The vessel is in compliance with the regulatory requirements, adequate safety margins are maintained, and, therefore, operation to 19.5 EFPY will not have an adverse effect on reactor vessel fracture toughness.

ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

Table 1
BFN Unit 2 Composite P-T Curve Data

PRESSURE (PSIG)	LIMITING CURVE 1 (°F)	PRESSURE (PSIG)	LIMITING CURVE 2 (°F)	LIMITING CURVE 3 (°F)
0	82.0	0	82.0	82.0
312	82.0	180	82.0	82.0
312	112.0	190	82.0	82.0
317	112.0	200	85.0	85.0
323	112.0	210	87.0	87.0
330	112.0	220	89.0	89.0
338	112.0	230	92.0	92.0
346	112.0	240	94.0	94.0
355	112.0	250	96.0	96.0
366	112.0	311	107.0	107.0
377	112.0	312	142.0	142.0
389	112.0	312	142.0	142.0
403	112.0	313	142.0	182.0
418	112.0	373	142.0	182.0
435	112.0	433	142.0	182.0
454	112.0	493	142.0	182.0
475	112.0	553	142.0	182.0
497	112.0	613	142.0	182.0
523	112.0	620	142.0	182.0
550	112.0	650	142.0	182.0
581	112.0	710	146.0	186.0
615	112.0	714	146.0	186.0
653	112.0	774	150.0	190.0
803	112.0	827	152.0	192.0
840	117.0	889	157.0	197.0
881	122.0	957	162.0	202.0
926	127.0	1033	167.0	207.0
976	132.0	1117	172.0	212.0
1032	137.0	1209	177.0	217.0
1093	142.0	1311	182.0	222.0
1160	147.0	1424	187.0	227.0
1235	152.0			
1318	157.0			
1409	162.0			

ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

Table 2
BFN Unit 3 Composite P-T Curve Data

PRESSURE (PSIG)	LIMITING CURVE 1 (°F)	PRESSURE (PSIG)	LIMITING CURVE 2 (°F)	LIMITING CURVE 3 (°F)
0	70.0	0	70.0	70.0
312	70.0	140	70.0	70.0
312	100.0	190	86.0	86.0
312	100.0	200	89.0	89.0
318	100.0	210	91.0	91.0
325	100.0	220	93.0	93.0
332	100.0	230	96.0	96.0
339	100.0	240	98.0	98.0
348	100.0	250	100.0	100.0
357	100.0	311	111.0	111.0
368	100.0	312	130.0	130.0
379	100.0	312	130.0	130.0
392	100.0	313	130.0	170.0
406	100.0	373	130.0	170.0
422	100.0	433	130.0	170.0
439	100.0	440	130.0	170.0
458	100.0	460	130.0	170.0
479	100.0	493	133.0	173.0
502	100.0	553	138.0	178.0
528	100.0	613	143.0	183.0
556	100.0	673	148.0	188.0
588	100.0	714	150.0	190.0
622	100.0	774	154.0	194.0
661	100.0	834	158.0	198.0
703	105.0	894	161.0	201.0
750	110.0	954	164.0	204.0
770	112.0	1014	167.0	207.0
824	117.0	1074	170.0	210.0
1032	137.0	1117	172.0	212.0
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ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Browns Ferry Nuclear Plant (BFN) Units 2 and 3 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed Units 2 and 3 change deals exclusively with the reactor vessel pressure-temperature (P-T) curves which define the permissible regions for operation and testing. Failure of the reactor vessel is not considered as a design basis accident. Through the design conservatisms used to calculate the P-T curves, reactor vessel failure has a low probability of occurrence and is not considered in the safety analyses. The proposed changes adjust the reference temperature for the limiting material to account for irradiation effects and provide the same level of protection as previously evaluated and approved. The adjusted reference temperature calculations were performed using the guidance contained in Regulatory Guide 1.99, Revision 2, and ASME Section XI Code Case N-640 to reflect use of the operating limits to 19.5 Effective Full Power Years (EFPY). These changes do not alter or prevent the operation of equipment required to mitigate any accident analyzed in the BFN Final Safety Analysis Report. Therefore, this change does not increase the probability or consequences of any previously evaluated accident.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change to the Units 2 and 3 reactor vessel P-T curves does not involve a modification to plant equipment. No new failure modes are introduced. There is no effect on the function of any plant system, and no new system interactions are introduced by this change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

The proposed curves conform to the guidance contained in Regulatory Guide 1.99, Revision 2, and maintain the safety margins specified in 10 CFR 50, Appendix G. Therefore, the proposed amendment does not involve a reduction in a margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed amendment does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed amendment is not required.

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PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

REFERENCES

1. Letter from TVA to NRC, dated December 15, 1998, Browns Ferry Nuclear Plant - Units 2 and 3 - TS Change No. 393, Supplement 1, P-T Curve Update

2. Letter from TVA to NRC, dated March 3, 1998, Browns Ferry Nuclear Plant - Units 2 and 3 - TS Change No. 393, P-T Curve Update

3. Letter from TVA to NRC, dated March 27, 1995, Generic Letter 92-01, Reactor Vessel Structural Integrity - Update To The Initial Reference Nil-Ductility Temperature (RTNDT), Chemical Composition And Fluence Values

4. Letter from TVA to NRC, dated July 28, 1994, Supplemental Response To TVA Letter dated May 23, 1994, Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

5. Letter from TVA to NRC, dated May 23, 1994, TVA's response to NRC's letter dated April 19, 1994, "Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity"

6. Letter from TVA to NRC, dated August 2, 1993, Response To Request For Additional Information, Generic Letter 92-01, Revision 1

7. Letter from TVA to NRC, dated July 7, 1992, Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), Watts Bar Nuclear plant (WBN), Response To Generic Letter 92-01 (Reactor Vessel Structural Integrity)

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-414
MARKED PAGES

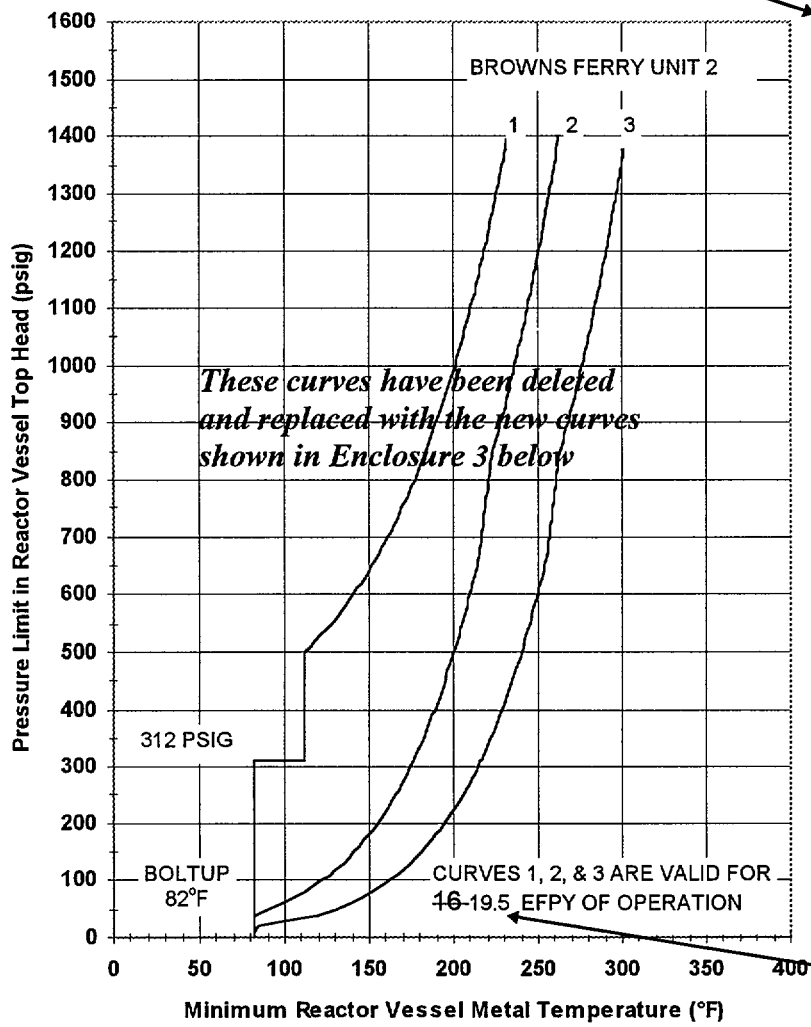
I. AFFECTED PAGE LIST

Unit 2 - page 3.4-29
Unit 3 - page 3.4-29

II. MARKED PAGES

See attached.

Added "ASME" for additional clarity in this note.



Curve No. 1
Minimum temperature for pressure tests such as required by *ASME* Section XI.

Curve No. 2
Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

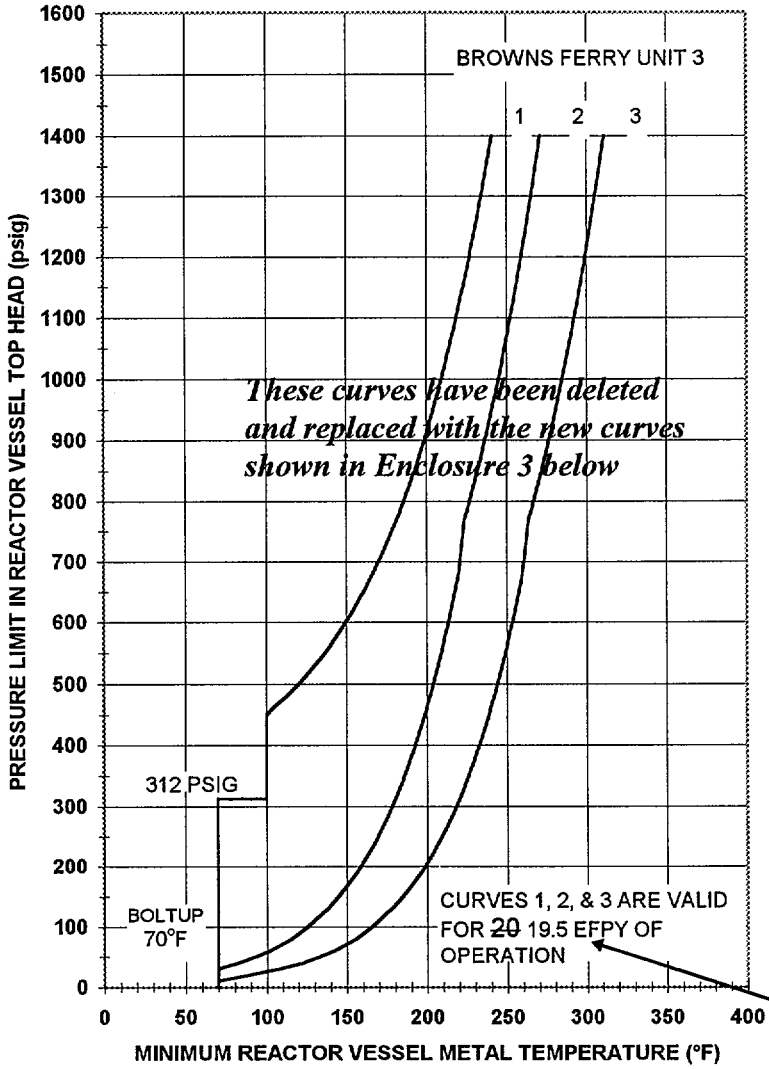
Curve No. 3
Minimum temperature for core operation (criticality).

Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel beltline materials, in accordance with Reg. Guide 1.99, Rev. 2, to compensate for radiation embrittlement for ~~16~~ 19.5 EFPY.

The period of valid use of these curves has been changed from 16 to 19.5 EFPY.

Figure 3.4.9-1
Pressure/Temperature Limits

Added "ASME" for additional clarity in this note.



Curve No. 1
Minimum temperature for pressure tests such as required by *ASME* Section XI.

Curve No. 2
Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3
Minimum temperature for core operation (criticality).

Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel beltline materials, in accordance with Reg. Guide 1.99, Rev 2, to compensate for radiation embrittlement for ~~20~~ 19.5 EFPY.

The period of valid use of these curves has been changed from 20 to 19.5 EFPY.

Figure 3.4.9-1
Pressure/Temperature Limits

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
Units 2 and 3

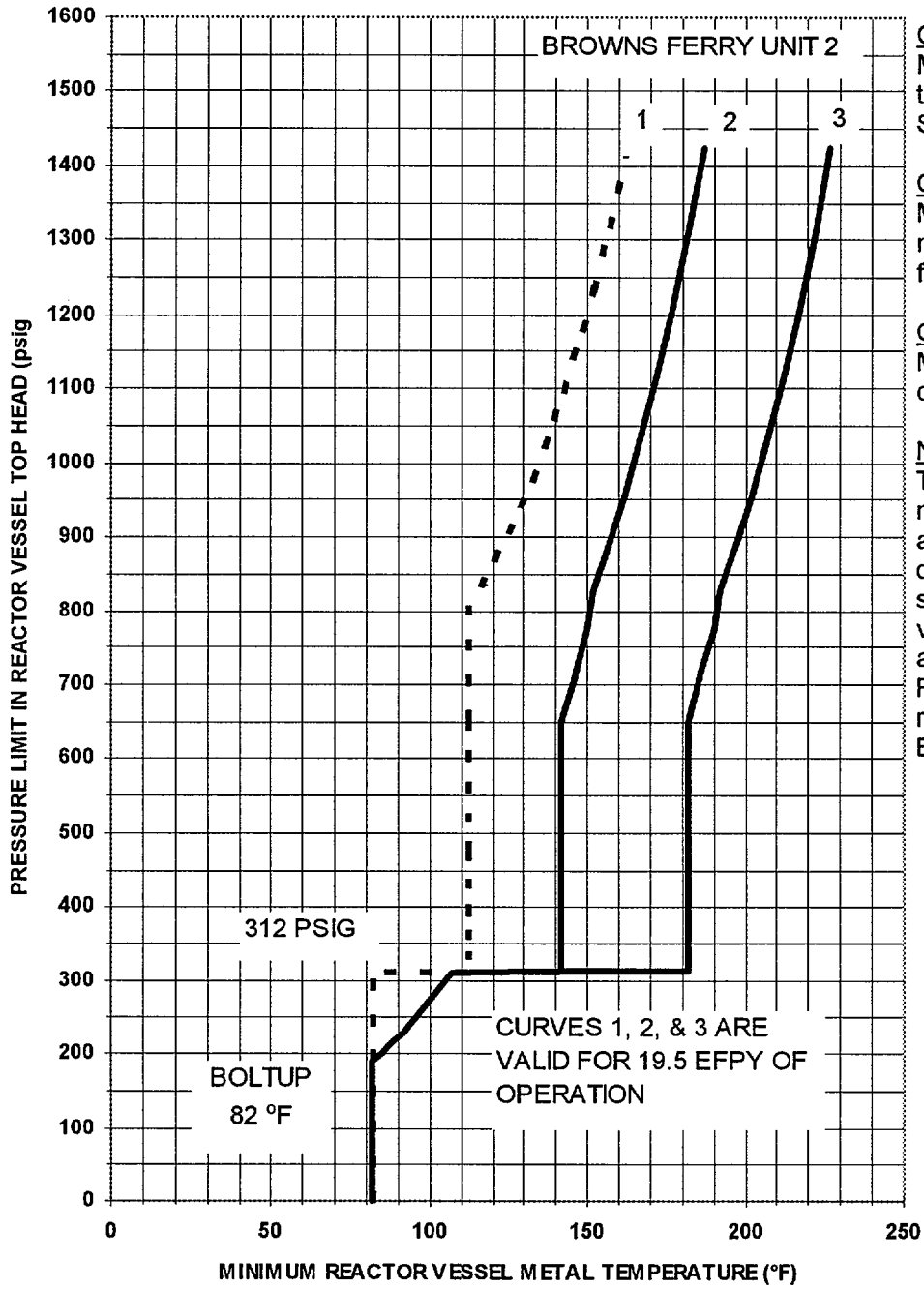
PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-414
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 2 Page 3.4-29
Unit 3 Page 3.4-29

II. REVISED PAGES

See attached.



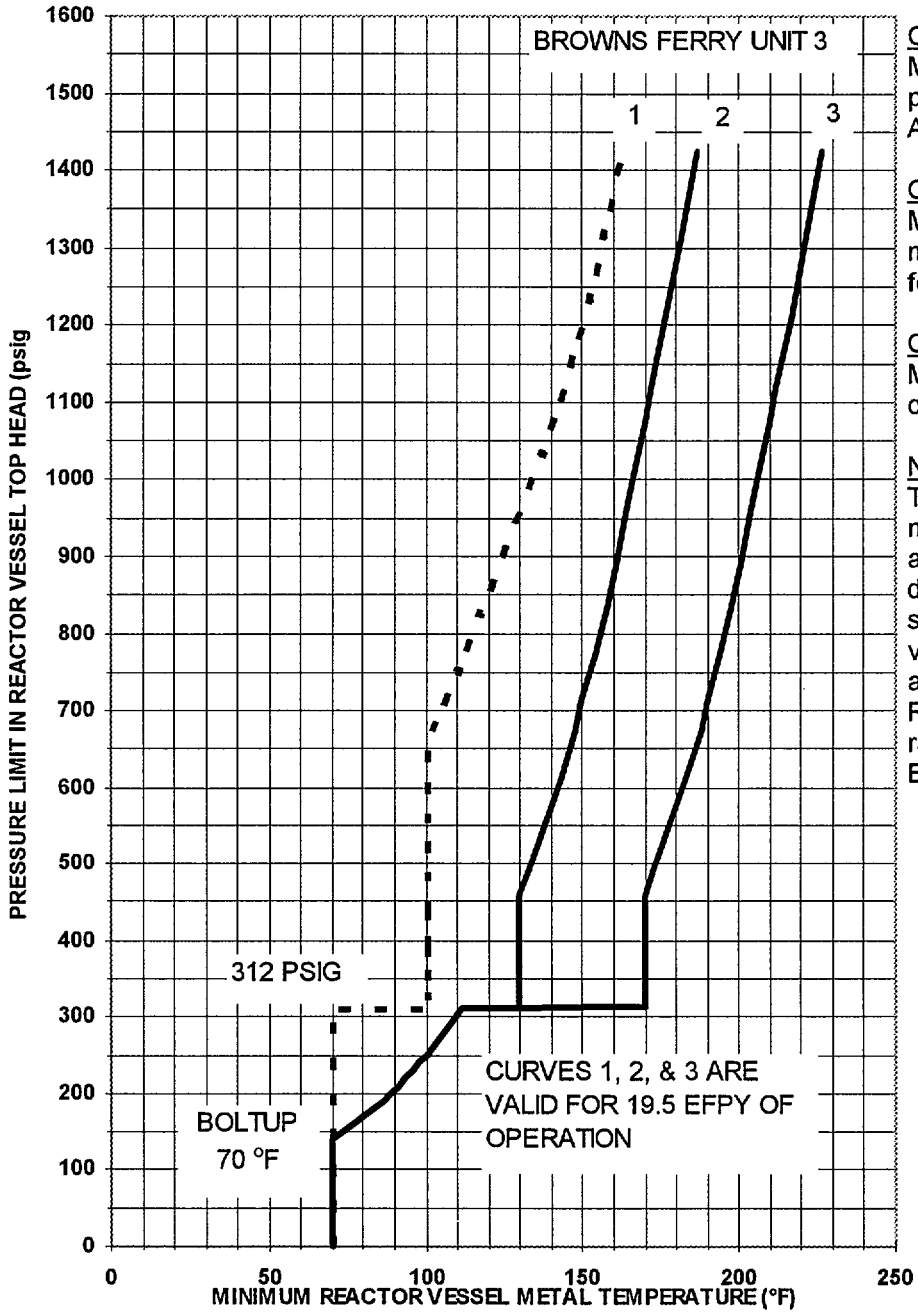
Curve No. 1
Minimum temperature for pressure tests such as required by ASME Section XI.

Curve No. 2
Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3
Minimum temperature for core operation (criticality).

Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel beltline materials, in accordance with Reg. Guide 1.99, Rev. 2, to compensate for radiation embrittlement for 19.5 EFPY.

Figure 3.4.9-1
Pressure/Temperature Limits



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Curve No. 3
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Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel beltline materials, in accordance with Reg. Guide 1.99, Rev. 2, to compensate for radiation embrittlement for 19.5 EFPY.

Figure 3.4.9-1
Pressure/Temperature Limits

ENCLOSURE 4

JUSTIFICATION FOR EXEMPTION ALLOWING USE OF ASME CODE CASE N-640

In accordance with 10 CFR 50.12, "Specific exemptions," TVA is requesting an exemption from the requirements of 10 CFR 50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b.

10 CFR 50.12(a) Requirements

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME B&PV Section XI, Appendix G to determine the pressure-temperature (P-T) limits for the reactor pressure vessel meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that the following is met:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H, when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   The revised P-T curve limits being proposed for Browns Ferry Units 2 and 3 rely in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining P-T limit curves remain unchanged.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve reduces the excess conservatism in the current Appendix G approach, that could, in fact, reduce overall plant safety. The K_{Ic} curve models the slow heat-up and cooldown process of a reactor pressure vessel. Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel material fracture toughness. Since 1974 additional knowledge has been gained about the fracture toughness of reactor pressure vessel materials and their fracture response to applied loads. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential reactor pressure vessel failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential reactor pressure vessel failure and does not present an undue risk to public health and safety.

P-T curves based on the K_{Ic} fracture toughness limits will enhance overall plant safety by opening the pressure-temperature operating window. Two primary safety benefits would be realized with no decrease to the margin of safety.

- Challenges to the operators would be reduced since the requirements for maintaining a high vessel temperature during pressure testing would be lessened.
- Enhanced personnel safety would result because of the lower temperatures which would exist during the conduct of inspections in primary containment.

3. <u>The requested exemption is consistent with the common defense and security</u>.

The subject of this exemption does not affect national defense or security issues. The common defense and security are not impacted by approval of this exemption request.

4. <u>Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60</u>.

In accordance with 10 CFR 50.12(a)(2), NRC will consider granting an exemption to the regulations if special circumstances are present. This requested exemption meets the special circumstances of the following paragraphs of 10 CFR 50.12:

- (a)(2)(ii) - demonstrates the underlying purpose of the regulation will continue to be achieved;
- (a)(2)(iii) - would result in undue hardship or other costs that are significant if the regulation is enforced and;
- (a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

10 CFR 50.12(a)(2)(ii): ASME B&PV Code, Section XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure vessel and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfy the underlying requirement that:

- The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the reactor pressure vessel boundary behaves in a non-brittle manner, and the probability of a rapidly propagating fracture is minimized, and

- P-T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The ASME B&PV Code, Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME B&PV Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable margin of safety.

10 CFR 50.12(a)(2)(iii): The reactor pressure vessel pressure-temperature operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of Browns Ferry Units 2 and 3 with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the pressure-temperature operating window. This restriction challenges the operations staff during pressure tests to maintain a high temperature within a limited operating window, and the higher

temperatures result in greater physical stress on the inspection personnel working in the vicinity of the piping.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety below that established by the original requirement.

10 CFR 50.12(a)(2)(v): The requested exemption provides only temporary relief from the applicable regulation and TVA has made a good faith effort to comply with the regulation. TVA requests the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry.

Code Case N-640, Conclusion for Exemption Acceptability:

Compliance with the specified requirement of 10 CFR 50.60(a) would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the lower bound fracture toughness used in ASME B&PV Code, Section XI, Appendix G, in the determination of reactor coolant system pressure-temperature limits. This proposed alternative is acceptable because the ASME Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for Browns Ferry Units 2 and 3 will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414
PRESSURE - TEMPERATURE CURVE UPDATE
COMMITMENT LISTING

1. In accordance with 10 CFR 55a(g)(4)(iv), TVA will implement the 1995 Edition, 1996 Addenda of Section XI, Appendix G requirements for Unit 3 in conjunction with the approval of the TS-414.