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

December 14, 2001

TVA-BFN-TS-414 Supplement 1 10 CFR 50.90

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - TVA RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION (RAI) REGARDING UNITS 2 AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGE NO. 414 - PRESSURE - TEMPERATURE (P-T) CURVE UPDATE

By letter dated August 17, 2001, BFN submitted a license amendment request for NRC approval of updated P-T curves for BFN Units 2 and 3. Subsequent to the submittal of that request, the proposed changes to the P-T curves were discussed in teleconferences between members of the NRC staff and TVA personnel on October 18, 2001, on October 31, 2001, and again on November 26, 2001. This supplement to TS-414 provides TVA's response to the issues raised by the NRC staff in these discussions. The P-T curves themselves have been revised from those originally submitted, and these revised curves are submitted under this cover.

The NRC staff questions arising from the review of the TS-414 submittal along with the corresponding TVA responses are included as Enclosure 1 to this letter. Enclosure 2 provides the description and evaluation of the proposed change. Enclosure 3 contains marked up pages of the appropriate TS for Units 2 and 3 (unchanged from the original TS-414 submittal). Enclosure 4 contains copies of the revised pages as they would appear following approval of this request.

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TVA has determined that the proposed finding of no significant hazards considerations and environmental impact consideration as submitted in the August 17, 2001 letter remain valid. 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.

TVA's request for exemption from the requirements of 10 CFR 50, Appendix G, which was submitted in conjunction with TS-414 to allow the use of ASME Code Case N-640 as a basis for the revised curves, is not affected by this supplemental submittal.

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

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Subscribed and sworn to before me
on this 14 day of December 2001.

Notary Public
My Commission Expires 4/12/2003

Enclosures
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NRC Question	*Page 3 of your 17 August 2001 submittal states that the revised Pressure-Temperature (P-T) limits requested were calculated using a neutron fluence value of 1.12 E18 n/cm^2 at 32 Effective Full Power Years (EFPYs). This fluence was characterized as "conservative" at end-of-life. This fluence appears to have been established from SwRI Project 02-4884-0001 dated August 1978. Given Regulatory Guide 1.190, March 2001 which provides methods acceptable for determining the neutron fluence; justify the fluence used in your calculations and quantify the uncertainty using an acceptable methodology. Clarify whether the methodology used to support the fluence value submitted for this amendment request is the same used to support current operation. Review the effects on operability due to uncertainty in your current fluence calculation.*
TVA Response	The original calculated neutron fluence value for 32 EFPY is 1.07 E18 n/cm^2 for Units 2 and 3. This value was obtained by multiplying the value of neutron fluence per EFPY (3.34 E16 n/cm^2) documented in SWRI Project 02-4884-002 by 32 EFPY. This calculated value has been the basis for previous P-T curve submittals, until the TS393 submittal in January 1999 increased the value to 1.12 E18 n/cm^2 to allow for the effects of 5% power uprate. The 5% power uprate was conservatively applied for the entire 32 EFPY operating period for both Units 2 and 3, even though the 5% power uprate was not initiated until the U2C11 Fuel Cycle in May 1999 and U3C9 Fuel Cycle in October 1998. In Fall 1994, at 8 EFPY, BFN removed the first surveillance capsule for Unit 2 during the U2C7 Refueling Outage. Data from this capsule showed that the 32 EFPY fluence is about 43% less (6.05 E17 n/cm^2) than the calculated fluence value, showing that the fluence value of 1.12 E18 n/cm^2 used in the present submittal is conservative. The current fluence value was not calculated in accordance with the guidelines of Regulatory Guide 1.190. However, BFN plans to submit new fluence calculations as part of its Extended Power Uprate (EPU) and License Renewal efforts prior to expiration of the new P-T curves provided with this submittal.

NRC Question	*Page 2 of your October 18, 1995 surveillance specimen test results report stated that TVA intended to withdraw test capsules every six EFPY following the removal of the eight EFPY capsule. If this has been performed, this data should be included to qualify the calculational methodology. Discuss how this commitment meshes with the Integrated Surveillance Program.*
TVA Response	**Per NRC SERs dated April 2, 2001 (TAC No. MB0741), and September 20, 1999 (TAC No. MA5403), the BFN RPV Material Surveillance Program was revised to change the surveillance capsule withdrawal schedules for both Units 2 and 3 respectively. For Unit 2, while the first capsule was removed at 8 EFPY, the second capsule is not required to be removed until 16 EFPY. BFN Unit 2 should reach this value in 2003. For Unit 3, the initial capsule is not required to be removed until 18 EFPY. BFN Unit 3 should reach this value in 2009.**

NRC Question	*(A) Confirm that, according to your most recent evaluation, the BF 2 and BF 3 axial RPV electroslag welds continue to be the limiting material for the P-T limit evaluations.*
TVA Response	**10CFR50 Appendix G states the beltline region of the reactor vessel (shell material including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core is predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. A comparison of ART values for the beltline, upper vessel, and bottom head regions of the RPV showed that the value of 110.6° for the axial RPV electroslag welds was the highest ART value throughout the RPV and demonstrated that the electroslag welds were the most limiting material for the P-T curve evaluations.**

NRC Question	*(B) Confirm that you assigned a conservative, peak neutron fluence value to all RPV materials and explain how the 1/4T fluence value was determined (direct calculation or ID surface calculation plus attenuation through the RPV wall using RG 1.99 Rev. 2 formula).*
TVA Response	**The peak fluence value of 1.12 E18 n/cm^2 at 32 EFPY was used in the calculations for all RPV materials. The 1/4T fluence value was determined by ID surface calculation plus attenuation through the RPV wall using the formula in RG 1.99, Rev. 2.**

NRC Question	*Given (A) and (B), explain how the limiting ART value of 102.9 ^{O}F was calculated (the NRC staff does not get the same result based on our understanding of your assumptions). Explain why this value is being compared to the 200 ^{O}F criteria found in RG 1.99 Rev. 2, Section C.3. for "new plants" at "end of life" when BF 2 and 3 are neither "new plants" nor, we assume, is this value related to an "end of life" calculation for the BF 2 and 3 RPVs.*
TVA Response	**The ART value cited in the 8/17/01 TS-414 submittal was in error. A σ_i value of 0°F was used in computing this number rather than the appropriate σ_i value of 13°F. The correct ART value for BFN Unit 2 and Unit 3 is 110.6°F.** **The calculated ART value is compared to the end-of-life temperature criteria of 200°F in RG 1.99 Rev. 2, Section C.3, only as a means of demonstrating that there is adequate margin remaining for the new P-T curves at 19.5 EFPY.**

NRC Question	*A discussion is provided in Enclosure 1[(a)] about the use of a "conservative" approach to addressing thermal loads (assuming tensile thermal loads at the 1/4T location for both cooldown and heatup operation). However, the NRC staff cannot find anywhere in the submittal, in Enclosure 1, on the P-T limit curves, etc., where the limiting heatup or cooldown rate is specified. This limiting rate information is important since it will define the magnitude of the thermal loads and the temperature lag between the fluid and the 1/4T location. Is this rate 100 F/hr, or something less?*

(a)- this refers to Enclosure 1 of the August 17, 2001 TS-414 submittal

TVA Response	**A rate of 100°F/hr is used in the calculations for Curves 2 and 3. This corresponds to the heatup/cooldown rate value found in SR 3.4.9.1 of the BFN Improved Technical Specifications. Thermal stresses are assumed to be insignificant in the calculation for Curve 1. TS SR 3.4.9.1 specifies a maximum heatup/cooldown rate of 15°F/hr during vessel pressure tests; this limitation supports the Curve 1 calculation.**

NRC Question	*Further, is the maximum thermal stress intensity (K_{IT}) calculated based on an equilibrium temperature distribution at some point in the heatup/cooldown transient and is this maximum K_{IT} value then conservatively applied to determine all pressure-temperature pairs comprising "limiting curves" 2 and 3 in Table 1 and Table 2[(a)]? Or is a time-dependent K_{IT} value determined at different points during the heatup/cooldown transient?*
TVA Response	**The maximum thermal stress intensity (K_{IT}) is based on an equilibrium temperature distribution at some point in the heatup/cooldown transient and is then conservatively applied to determine all P-T pairs comprising limiting curves. The methodology used to determine the stress intensity factor due to the thermal load is taken from ASME Code Case N-588. Paragraph 2214.3 of ASME Code Case N-588 discusses the determination of the thermal stress intensity factor. It is stated that the thermal stress intensity factor is the maximum stress intensity factor with heatup/cooldown rates of 100°F/hour. Thus, the K_{IT} corresponds to the maximum value that occurs during the thermal loading.**

(a)- this refers to Tables 1 and 2 of Enclosure 1 of the August 17, 2001 TS-414 submittal

NRC Question	*Does the term "minimum reactor vessel metal temperature" refer to the RPV metal temperature at the inner wall location, 1/4T location, outer wall, or somewhere else?*
TVA Response	**The term means the lowest inner wall temperature at the appropriate vessel locations.**

NRC Question	*How is "temperature" measured for this application (thermocouples attached to the loops and bottom head, RTDs, etc.)? Is the method of establishing "temperature" different depending on whether the reactor is in a hydrotest condition vs. normal operations?*
TVA Response	**The temperature data are obtained by one or more of three different methods:** **1. from thermocouples which are in direct physical contact with the reactor vessel external surface** **2. from recirculation pump suction water temperatures** **3. from the saturation temperature corresponding to the current reactor dome pressure during steaming conditions** **During hydrotest conditions and during approach to criticality from cold conditions the vessel metal temperature thermocouple data are used. During normal operations or hot shutdown conditions (where steaming conditions exist) the steam dome pressure-saturation temperature correlation and the reactor water temperature values are used.**

NRC Question	*Based on the means being used to measure temperature, how is the measured temperature then related to the temperature of the fluid, the temperature of the RPV at that 1/4T location (which is the critical depth for these calculations based on the size of the flaw one must assume per ASME Code requirements), and the minimum reactor vessel metal temperature as it is defined?*
TVA Response	**For all conditions, the temperature at the 1/4T and 3/4T locations was assumed to be equivalent to the fluid temperature. This is justified for the following reasons:** **For cooldown conditions the limiting location is at 1/4T. The use of the fluid temperature is conservative to represent the 1/4T location, as it will result in a lower K_{IC}. This K_{IC} calculation uses the appropriate fluence at the 1/4T location.** **For heatup conditions, the limiting location is at the 3/4T location as it will be in tension and 1/4T will be in compression. At this location, the fluence is**

	significantly lower than at the 1/4T location and the inside surface. This serves to reduce the fluence factor (FF) term in the ART calculation. However, the temperature at the 3/4T location will be lower than that at the 1/4T location or inside surface. Thus, fluence and temperature at the 3/4T location have an offsetting effect on K_{IC}. Finite element analyses for a 100 ^{o}F/hour event has demonstrated a through-wall temperature distribution of approximately 25 ^{o}F. Using the temperatures and fluence at 1/4T and 3/4T from the finite element model, the appropriate ART's at these locations were determined (note that at 3/4T, the limit on σ_{Δ} is included). Results of the K_{IC} calculations using the correct 1/4T and 3/4T fluence and temperatures showed that use of the fluid temperature to represent the 1/4T and 3/4T locations to develop the P-T curves was justified. In addition, when determining the K_{IT} term (maximum thermal induced stress intensity anywhere in the vessel wall) for heatup, the coefficients from Code Case N-588 was used. Cool down results in an approximately 20% higher K_{IT} than that for the same through-wall ΔT for heat-up.

ENCLOSURE 2

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

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414 SUPPLEMENT 1
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 10.6 EFPY in its operating history (through November 2001), 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.

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 2 is itself a composite of the most limiting conditions of the heatup and cooldown curves, therefore it specifies satisfactory limitations whether a heatup or cooldown is occurring.

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.

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 2-8).

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 110.6 °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 A and B below 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. In addition, the analysis conforms to the requirements of 10 CFR 50, Appendix G, which ensures that the most limiting material is considered in the development of the P-T curves. 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 2
PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414 SUPPLEMENT 1
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

Table A
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	652	142.0	182.0
581	112.0	695	147.0	187.0
615	112.0	743	152.0	192.0
653	112.0	796	157.0	197.0
753	112.0	855	162.0	202.0
784	117.0	920	167.0	207.0
820	122.0	991	172.0	212.0
858	127.0	1070	177.0	217.0
901	132.0	1158	182.0	222.0
949	137.0	1255	187.0	227.0
1001	142.0	1361	192.0	232.0
1059	147.0	1480	197.0	237.0
1123	152.0			
1194	157.0			
1272	162.0			
1358	167.0			
1454	172.0			

Note: the values in the above table have been revised to reflect a limiting ART value of 110.6 °F.

Table B
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	743	152.0	192.0
661	100.0	796	157.0	197.0
703	105.0	855	162.0	202.0
741	110.0	920	167.0	207.0
753	112.0	991	172.0	212.0
784	117.0	1070	177.0	217.0
820	122.0	1158	182.0	222.0
858	127.0	1255	187.0	227.0
901	132.0	1361	192.0	232.0
949	137.0	1480	197.0	237.0
1001	142.0			
1059	147.0			
1123	152.0			
1194	157.0			
1272	162.0			
1358	167.0			
1454	172.0			

Note: the values in the above table have been revised to reflect a limiting ART value of 110.6 OF.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-414 SUPPLEMENT 1
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

REFERENCES

1. Letter from TVA to NRC, dated August 17, 2001, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specifications (TS) Change No. 414 - Pressure - Temperature Curve Update (TAC numbers MB2751/2/3/4)

2. 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

3. 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

4. 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

5. 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

6. 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"

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

8. 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 3

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

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-414 SUPPLEMENT 1
MARKED PAGES

I. AFFECTED PAGE LIST

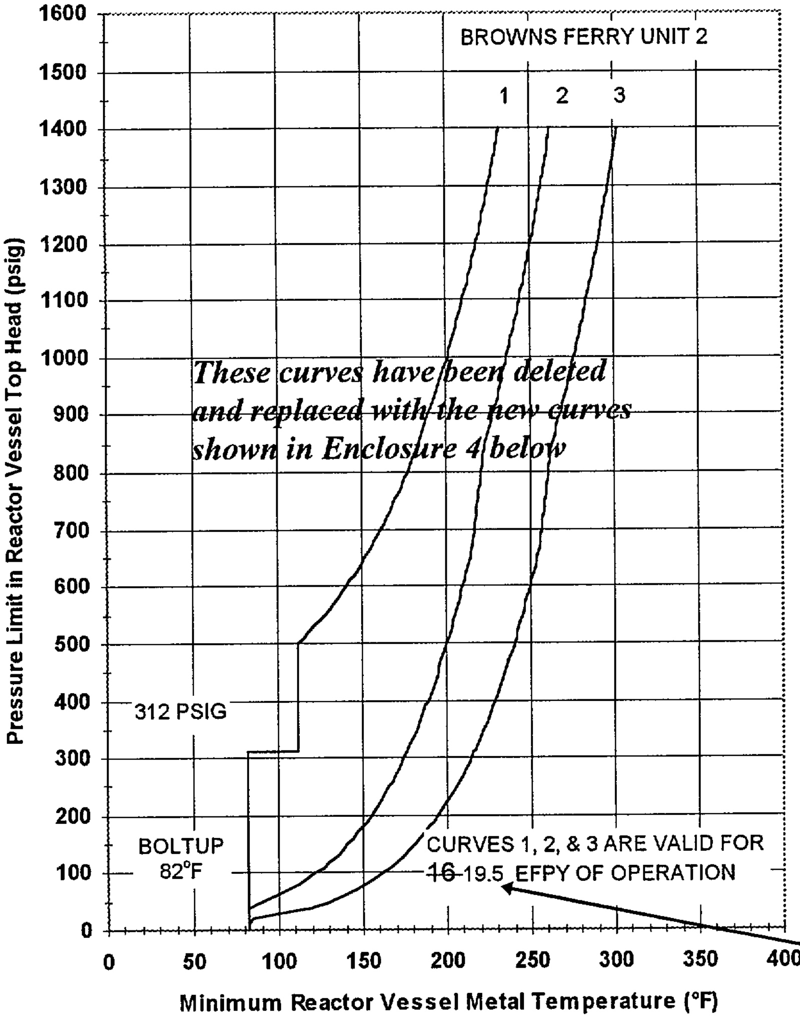
Unit 2 - page 3.4-29
Unit 3 - page 3.4-29

II. MARKED PAGES

See attached.

Note: these pages are unchanged from the previous TS-414 submittal

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.

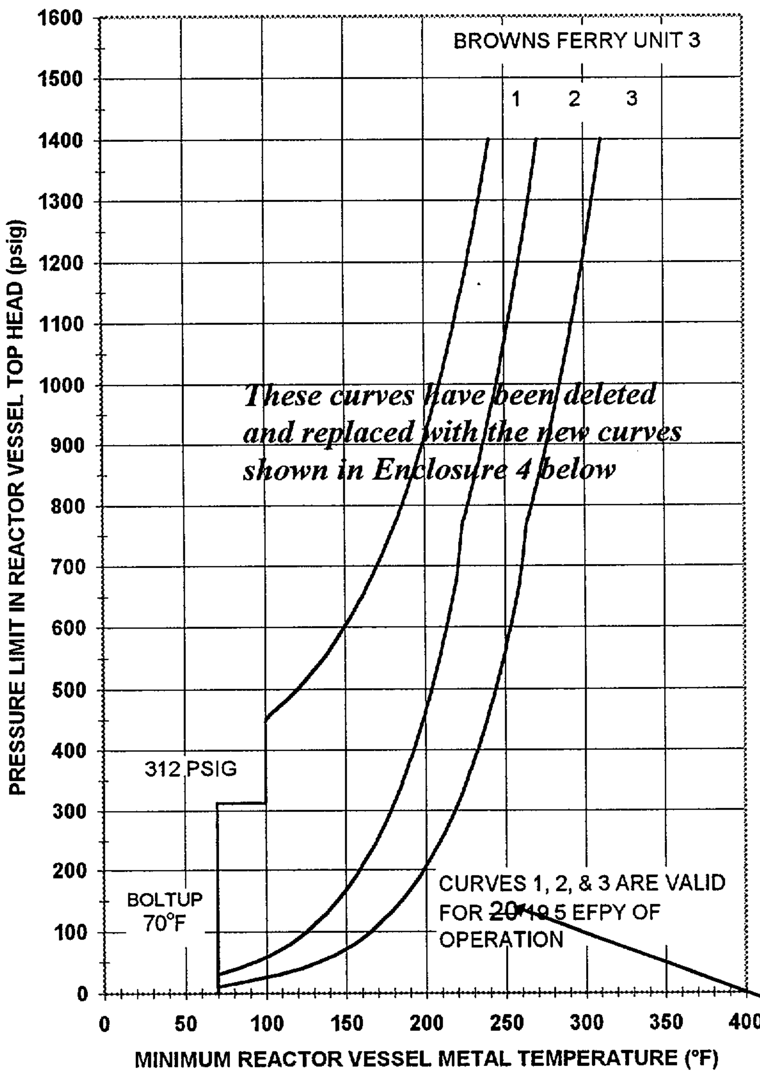
Curve No. 3
Minimum temperature for core operation (criticality).

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.



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Minimum temperature for core operation (criticality).

Notes
These curves include sufficient margin to provide protection againstfeedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vesselbeltline 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 4

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

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-414 SUPPLEMENT 1
REVISED PAGES

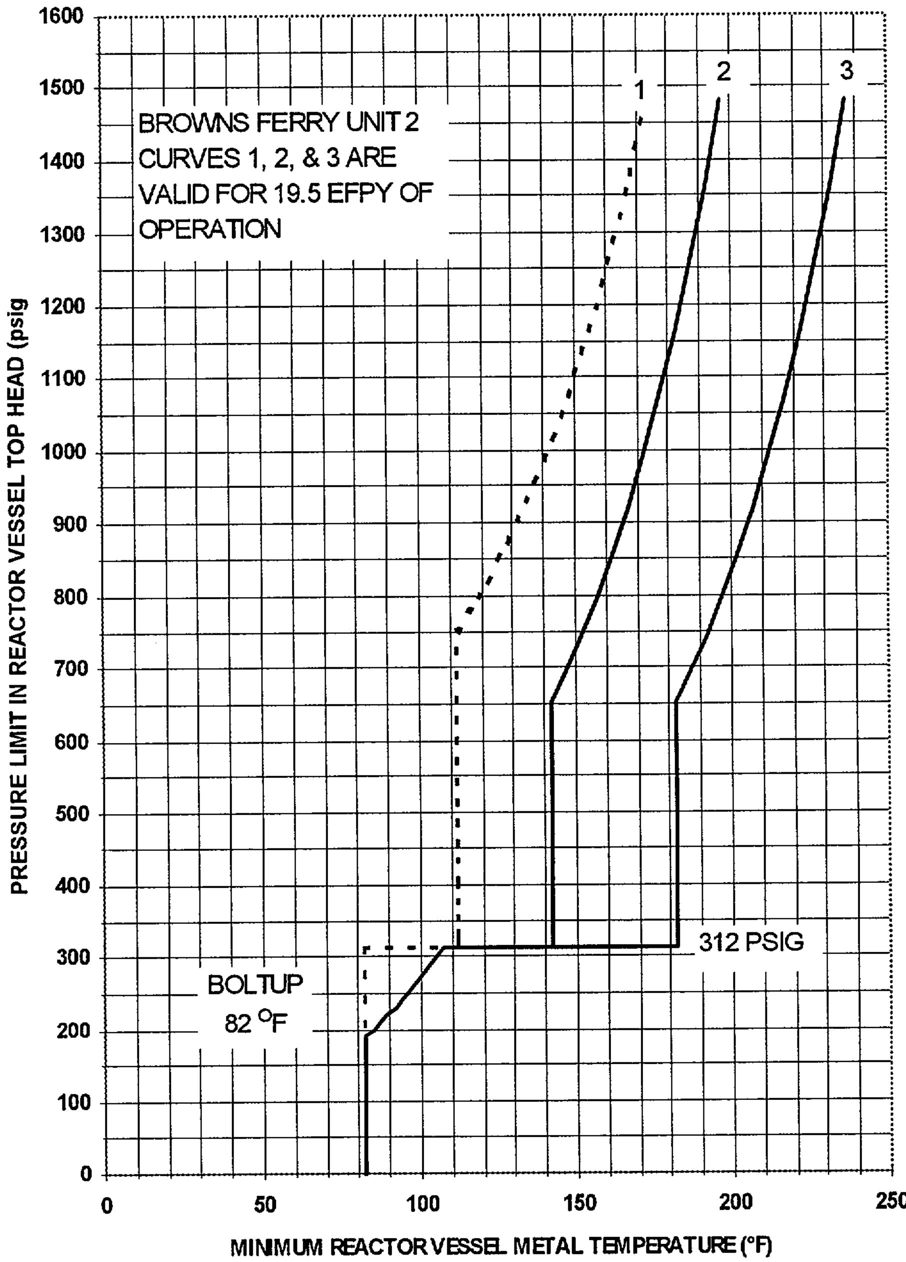
I. AFFECTED PAGE LIST

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

II. REVISED PAGES

See attached.

Note: these figures have been revised from those contained in TS-414 due to the change in the limiting ART from 102.9 °F to 110.6 °F.



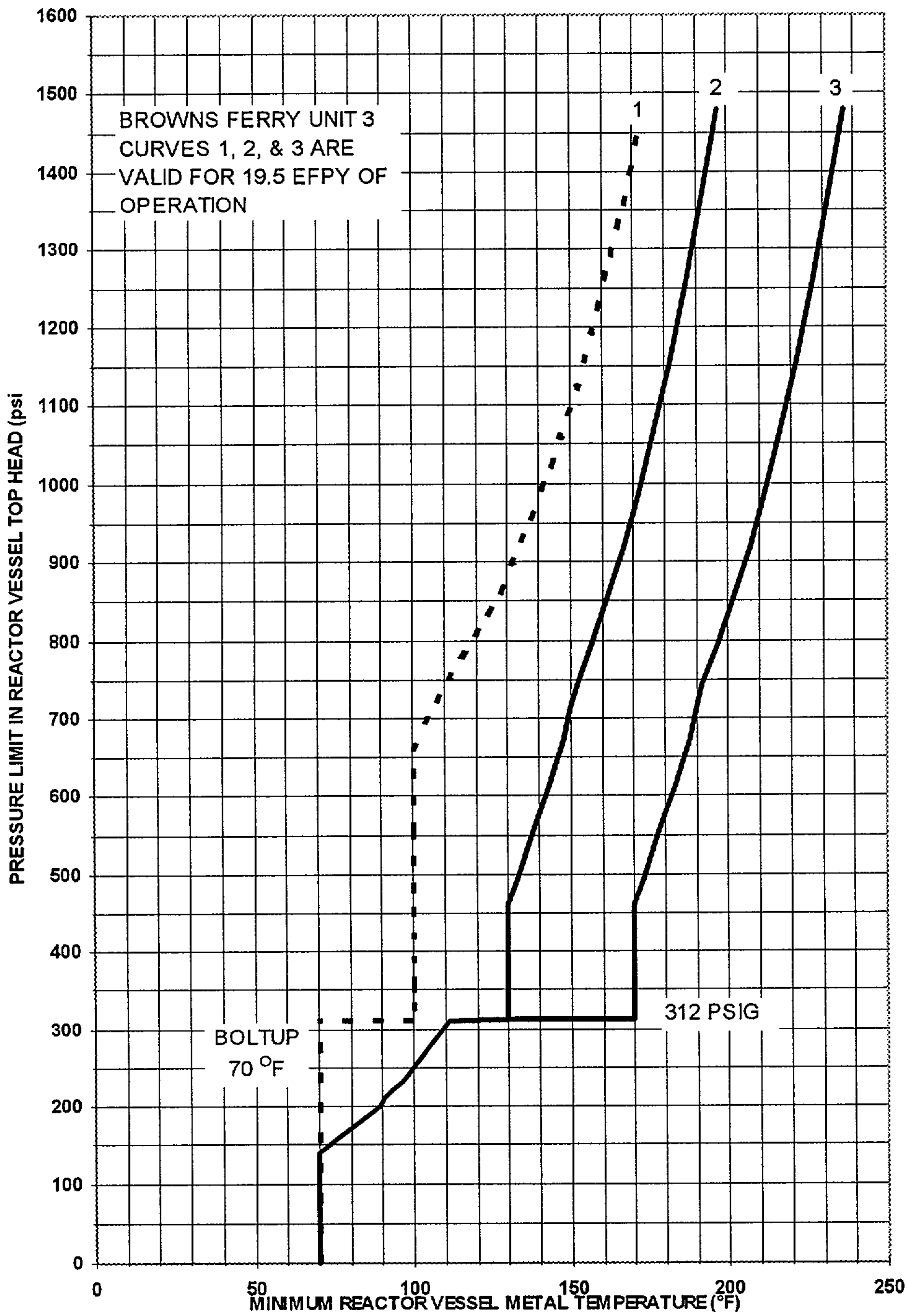
Curve No. 1
Minimum temperature for pressure tests such as required by ASME Section XI.

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



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