~~Proprietary Information Withhold Under 10 CFR 2.390(d)(1)~~
~~This letter is decontrolled when separated from Enclosure 2~~

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-13-148

December 18, 2013

10 CFR 50.90

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

Browns Ferry Nuclear Plant, Unit 1
Renewed Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Browns Ferry Nuclear Plant (BFN), Unit 1 - Application to Modify Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits" (BFN TS-484)**

In accordance with the provisions of Title 10 of the Code of Federal Regulations (CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," the Tennessee Valley Authority (TVA) is submitting a license amendment request to revise the Browns Ferry Nuclear Plant, Unit 1, Technical Specifications (TS) for Limiting Condition for Operation (LCO) 3.4.9, "RCS Pressure and Temperature (P/T) Limits." This submittal satisfies the commitment to prepare and submit revised BFN, Unit 1, P/T limits prior to the start of the period of extended operation as discussed in Section 4.2.5 provided in "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Application for Renewed Operating Licenses," dated December 31, 2003 (ADAMS Accession No. ML040060359).

This submittal satisfies the requirements of NUREG-1843, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated April 2006 (ADAMs Accession No. ML061030032), commitment 39 which required the development and submittal of revised P/T limit curves for NRC approval prior to the period of extended operation. In the "Browns Ferry Nuclear Plant - NRC Post-Approval Site Inspection for License Renewal, Inspection Report," dated October 3, 2013 (Inspection Report 005000259/2013009, 005000260/2013009, 005000296/2013009), with respect to commitment 39, it was stated that new P/T limits will be calculated and approved before the period of extended operation. The current NRC approved P/T limit curves were approved prior to the period of extended operation and are applicable for operation into, but not to the end of the period of extended operation. Therefore, the proposed P/T limit curves were developed based on analyses projected to the end of the period of extended operation as required by 10 CFR 54.21(c)(1)(ii).

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NRR

Enclosure 1 to this letter provides a description, technical evaluation, regulatory evaluation and environmental consideration of the proposed changes. Attachments 1 and 2 to Enclosure 1 provide a markup of the proposed changes to the BFN, Unit 1, TS and TS Bases (information only), respectively. Attachments 3 and 4 to Enclosure 1 provide retyped versions of the BFN, Unit 1, TS and TS Bases (information only), respectively, to show the incorporation of the proposed changes. Attachment 5 to Enclosure 1 provides a conforming markup of the BFN Updated Final Safety Analysis Report (UFSAR) that result from this proposed change.

Enclosure 2 to this letter contains information which General Electric - Hitachi (GEH) and Electric Power Research Institute (EPRI) consider to be proprietary in nature and subsequently, pursuant to 10 CFR 2.930, "Public inspections, exceptions, request for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure. Enclosure 4 provides the affidavits supporting the request. Enclosure 3 contains the redacted version of the proprietary attachment with the proprietary material removed, which is suitable for public disclosure.

TVA requests that the NRC approve this amendment by December 22, 2014, with implementation within 60 days of issuance.

TVA has determined that there are no significant hazards considerations associated with the proposed changes and that the changes qualify for a categorical exclusion 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 the proposed changes and determined that operation of BFN in accordance with the proposed changes 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 the non-proprietary enclosures to the Alabama State Department of Public Health.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Edward D. Schrull at 423-751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 18th day of December 2013.

Respectfully,

J.W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Evaluation of Proposed Changes
2. NEDC-33445P - Pressure and Temperature Limits Report (PTLR) Up to 25 and 38 Effective Full-Power Years (Proprietary)
3. NEDO-33445 - Pressure and Temperature Limits Report (PTLR) Up to 25 and 38 Effective Full-Power Years (Non-Proprietary)
4. Affidavits

cc: See page 3

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector – Browns Ferry Nuclear Plant
State Health Officer, Alabama State Department of Public Health

ENCLOSURE 1
EVALUATION OF PROPOSED CHANGES

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNIT 1

BFN 13-484, Browns Ferry Nuclear Plant (BFN), Unit 1, Application to Modify Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

1.0 SUMMARY DESCRIPTION

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4.0 REGULATORY EVALUATION

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- 4.2 Precedent
- 4.3 No Significant Hazards Consideration
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6.0 REFERENCES

ATTACHMENTS

1. Proposed Technical Specifications Pages Markups
2. Proposed Technical Specifications Bases Pages Markups (for information only)
3. Proposed Retyped Technical Specifications Pages
4. Proposed Retyped Technical Specifications Bases Pages (for Information only)
5. Proposed Updated Final Safety Analysis Report Page Markups

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend the Browns Ferry Nuclear Plant (BFN), Unit 1, Renewed Facility Operating License No. DPR-33 (Reference 1). The proposed changes modify the BFN, Unit 1, Technical Specification (TS) requirements related to the Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits in TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

Specifically, the proposed change replaces the current sets of TS Figures 3.4.9-1 and 3.4.9-2. The figures proposed to be replaced consist of two sets of P/T limit curves, one set valid up to 12 effective full power years (EFPY) of operation and another set valid from 12 EFPY to 16 EFPY of operation. The proposed change replaces the current curves with a set of figures valid for operation up to 25 EFPY and another set valid for operation from >25 EFPY and ≤38 EFPY. This proposed TS change is necessary to satisfy the commitment to develop and submit to the NRC P/T limits that have been projected to the end of the period of extended operation, prior to the start of the period of extended operation.

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

TVA proposes to delete and replace TS Figures 3.4.9-1 and 3.4.9-2 that are applicable for operations up to 12 EFPY and for operations > 12 EFPY and ≤ 16 EFPY, respectively, with new TS Figures 3.4.9-1 and 3.4.9-2 that are applicable for operations up to 25 EFPY and for operations > 25 EFPY and ≤ 38 EFPY, respectively.

In addition, an associated note for each figure is changed to reflect the new operational applicability limit with respect to EFPY.

2.2 Need for Proposed Changes

The License Renewal Application for BFN Units 1, 2, and 3 (Reference 2) states in Appendix A - UFSAR Supplement, Section A.3.1.5, "Because of the relationship between the operating pressure-temperature limits and the fracture toughness transition of the reactor vessel, all three units will require new operating pressure-temperature limit curves to be calculated and approved for the extended period of operation."

License Condition 2.E in the BFN, Unit 1, renewed operating license (Reference 1) states, "The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection."

The current Unit 1 P/T limit curves were approved by the NRC through the license amendment process (Reference 3) and are valid to 16 EFPY, well into the period of extended operation.

The proposed P/T limit curves which reflect the analyses projected to the end of the period of extended operation required by 10 CFR 54.21(c)(1)(ii) use conservative fluence values calculated for a period of 60 years operation. The current approved Unit 1 P/T limit curves are valid up to 16 EFPY which covers entry into the period of extended operation. The proposed Unit 1 P/T limit curves are necessary to replace the current curves and provide operational coverage beyond the current approved limit of 16 EFPY to 38 EFPY.

3.0 TECHNICAL EVALUATION

10 CFR 50 Appendix G, "Fracture Toughness Requirements," requires the establishment of 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 in themselves, because operation in accordance with these limitations precludes operation in an unanalyzed condition. The P/T limits are not derived from Design Basis Accident 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.

These proposed P/T limit curves 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 to avoid brittle fracture. The method used to account for irradiation embrittlement is described in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Rev. 2.

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," provides an acceptable method for calculating P/T limits that satisfies the requirements of 10 CFR 50 Appendix G.

The P/T limit curves for BFN, Unit 1, have been revised based on methodologies consistent with this regulatory guide using plant-specific material and fluence information. The proposed P/T limit curves reflect changes from those currently licensed. The new P/T limit curves incorporate a revised fluence calculated in accordance with NRC approved GE Licensing Topical Report NEDC-32983P-A, (Reference 4) representing BFN, Unit 1, operating conditions of up to 3952 MWt and incorporate the NRC approved methodologies described in GEH Topical Report NEDC-33178P-A (Reference 5). The operating condition of up to 3952 MWt represents the Extended Power Uprate (EPU) power level; however, the current license power level is 3458 MWt. The latest information from the BWRVIP Integrated Surveillance Program (ISP) applicable to BFN, Unit 1, has been incorporated.

Five separate curves are depicted in the revised TS Figure 3.4.9-1 and Figure 3.4.9-2 to clearly show the P/T limitations for the reactor bottom head area and the vessel beltline and upper vessel areas for all operating conditions.

- Curve 1 of Figure 3.4.9-1 specifies minimum temperature for bottom head during mechanical heatup or cooldown following nuclear shutdown (i.e., reactor not critical).
- Curve 2 of Figure 3.4.9-1 specifies minimum temperature for upper RPV and beltline during mechanical heatup or cooldown following nuclear shutdown (i.e., reactor not critical).
- Curve 3 of Figure 3.4.9-1 specifies minimum temperature for core operation (i.e., reactor critical).

- Curve 1 of Figure 3.4.9-2 specifies minimum temperature for bottom head during in-service leak or hydrostatic testing.
- Curve 2 of Figure 3.4.9-2 specifies minimum temperature for upper RPV and beltline during in-service leak or hydrostatic testing.

The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used for the evaluation to generate the P/T curves.

Values of initial RT_{NDT} for the vessel materials were calculated in accordance with the methods described in GEH Topical Report NEDC-33178P-A. The values used to determine the initial RT_{NDT} were obtained from the Certified Material Test Reports (CMTRs) for BFN, Unit 1. Initial RT_{NDT} values for the BFN, Unit 1, weld materials were not calculated; these values were obtained from previous reports (References 6 and 7) and verified for input for this evaluation.

The adjusted reference temperature (ART) for the beltline region was determined using the methods described in Regulatory Guide 1.99, Rev. 2. These values are summarized in Tables B-4 and B-5 of Enclosures 2 and 3.

The limiting adjusted reference temperature value of 193°F for the 38 EFPY calculation remains below the 200°F criterion of Regulatory Guide 1.99, Rev. 2. The Upper Shelf Energy (USE) equivalent margin analyses values calculated for end of life (i.e., 38 EFPY) remain within the limits of Regulatory Guide 1.99, Rev. 2 and 10 CFR 50 Appendix G. 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) 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, because BFN, Unit 1, is at steam saturation conditions during normal operation, well above the heatup/cooldown curve limits.

The GEH Pressure-Temperature Limits Report for BFN, Unit 1, provided as Enclosures 2 (proprietary) and 3 (non-proprietary), demonstrates the technical methods and contains the data for producing the composite P/T curves which are proposed to be placed in the TS. Table 1 in the GEH Report for BFN, Unit 1, contains the data for producing the composite 25 EFPY curves. In the same manner, Table 2 of the GEH Report contains the data for producing the composite 38 EFPY curves.

The proposed P/T curves have been developed utilizing the methodology of Regulatory Guide 1.190 and ASME Section XI. The regulatory guidance provides an allowance for margin to be included in the bounding values of the ART. Use of this methodology ensures adequate safety margins are maintained. In addition, the analysis conforms to the requirements of 10 CFR 50, Appendix G, which ensures 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, BFN, Unit 1 operation to 38 EFPY, representing a 60 year license, will not have an adverse effect on reactor vessel fracture toughness.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Criteria

Pursuant to 10 CFR 50.90, TVA is submitting a request for a Technical Specifications (TS) change to renewed license DPR-33 for BFN, Unit 1. The proposed change revises the reactor vessel pressure-temperature (P/T) limits depicted in current TS Figure 3.4.9-1 and Figure 3.4.9-2.

10 CFR 50, Appendix G, contains the requirements for the P/T limit curves, and requires that P/T curves for the reactor pressure vessel be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code.

The regulatory requirements for fluence calculations are contained in General Design Criteria (GDC) 30 and 31. NRC issued Regulatory Guide 1.190 in March 2001, which provided state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. NRC has approved vessel fluence calculation methodologies which satisfy the requirements of GDCs 30 and 31 performed with approved methodologies or with methods which are shown to adhere to the guidance in Regulatory Guide 1.190. The analyses supporting this submittal were performed in accordance with Regulatory Guide 1.190 guidance.

4.2 Precedent

The NRC has previously approved a License Amendment Request (LAR) that included two sets of P/T limit curves for BFN Units 2 and 3 (Amendments 288 and 247), one set for operation up to 23 EFPY (Unit 2) and up to 20 EFPY (Unit 3), and a second set for operations up to 30 EFPY (Unit 2) and up to 28 EFPY (Unit 3) (Reference 8). TVA proposed to the NRC in the February 24, 2004 LAR letter (Reference 9) that both sets of curves be placed in the TS upon approval of the amendments. In the NRC Safety Evaluation for BFN Units 2 and 3, Amendments 288 and 247, dated March 10, 2004 (Reference 8), the NRC stated that because the applicability of each set of curves was clearly defined, the approach proposed by TVA was acceptable.

The NRC has previously approved an LAR that revised facility P/T limit curves based on the application of methodology in GE Hitachi Nuclear Energy (GEH) Licensing Topical Reports NEDC-33178P-A (Reference 5) and NEDC-32983P-A (Reference 4) for the Peach Bottom Atomic Power Station, Units 2 and 3, Amendment Nos. 286 and 289, issued by NRC letter dated April 1, 2013 (ADAMS Accession No. ML13079A219). The Peach Bottom Atomic Power Station LAR used the 1998 edition of the ASME Boiler and Pressure Vessel Code including 2000 Addenda in its evaluation; and, applied calculated fluence values based on uprated conditions that conservatively bounded future operations.

4.3 No Significant Hazards Consideration

The proposed change modifies the Browns Ferry Nuclear Plant (BFN), Unit 1, Technical Specification (TS) requirements related to TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

Tennessee Valley Authority (TVA) has concluded that the changes to BFN, Unit 1, TS 3.4.9 do 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, "Issuance of Amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of any accident previously evaluated?**

   Response: No

   The proposed changes are to accepted operating parameters that have been approved in previous license amendments. The changes to P/T curves were developed based on NRC approved methodologies. The proposed changes deal exclusively with the reactor vessel 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 in accordance with the requirements of 10 CFR 50 Appendix G using the guidance contained in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," to reflect use of the operating limits to no more than 38 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, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   The proposed changes are to accepted operating parameters that have been approved in previous license amendments. The changes to P/T curves were developed based on NRC approved methodologies. The proposed changes to the reactor vessel P/T curves do 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 change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No

   The proposed changes are to accepted operating parameters that have been approved in previous license amendments. The changes to P/T curves were developed based on NRC approved methodologies. The proposed curves conform to the guidance contained in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," and maintain the safety margins specified in 10 CFR 50 Appendix G. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

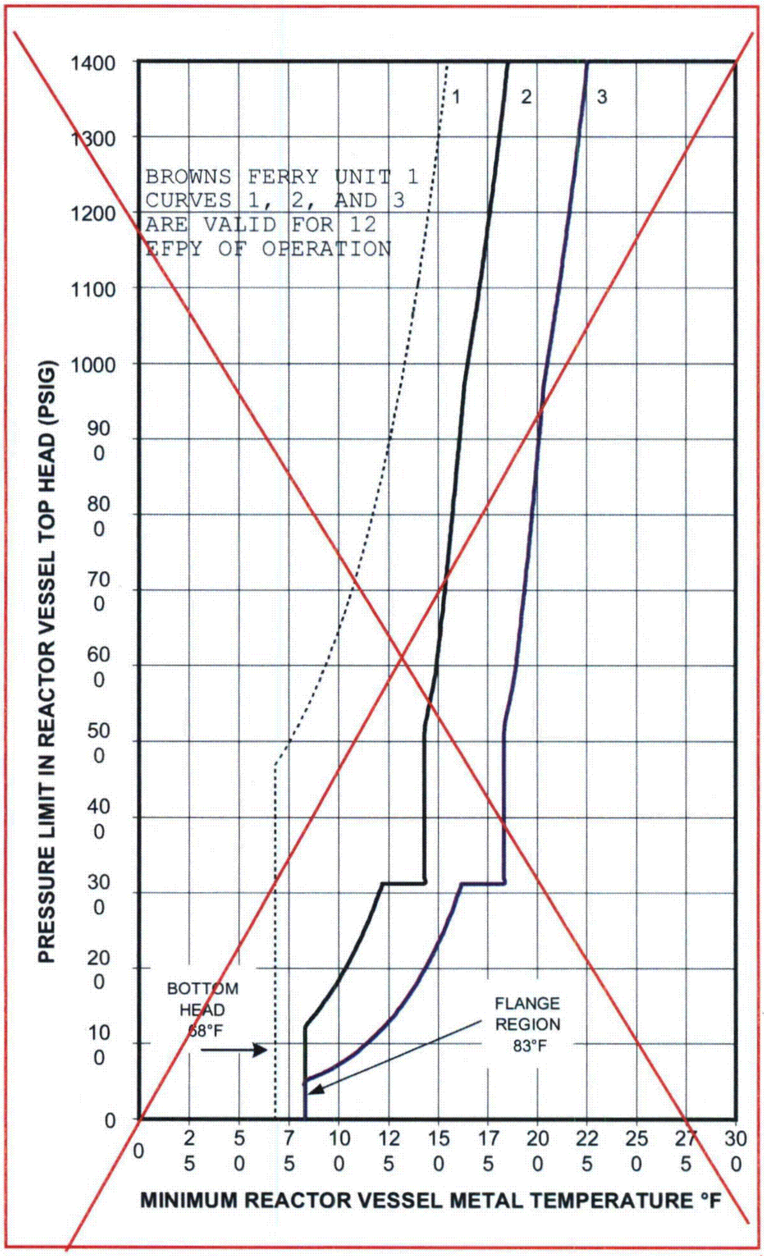
1. Tennessee Valley Authority Browns Ferry Nuclear Plant, Unit 1, Docket No. 50-259, Renewed Facility Operating License, Renewed License No. DPR-33 (ADAMS Accession No. ML061110094).
2. Letter from TVA to NRC, "Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3 - Application for Renewed Operating Licenses," dated December 31, 2003 (ADAMS Accession No. ML040060359).
3. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding Update of Pressure-Temperature Limit Curves (TAC No. MC5373) (TS 428)," dated July 26, 2006 (ADAMS Accession No. ML061090658).
4. GE Nuclear Energy, NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," (TAC No. MC9891).
5. GEH Nuclear Energy, NEDC-33178P-A, Revision 1, "GE Hitachi Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," June 2009 (GEH Proprietary).
6. Evaluation of RTNDT, USE, and Chemical Composition of Core Region Electroslag Welds for Quad Cities Units 1 and 2, Framatome Technologies, Lynchburg, Virginia, January 1996 (BAW-2259).
7. Letter, TE Abney (TVA) to US NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Generic Letter (GL) 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity - Response to NRC Request for Additional Information (TAC Nos. MA1179, MA1180, and MA1181)", September 8, 1998.

8. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Pressure-Temperature Limit Curves (TAC Nos. MC0807 and MC0808)," dated March 10, 2004 (ADAMS Accession Nos. ML040480013, ML040750188, and ML040750194).

9. Letter from TVA to NRC, "Browns Ferry Nuclear Plant (BFN) - TVA Revision to Implementation Plant Described in Units 2 and 3 - Technical Specifications (TS) Change No. 441 Revision 1 - Pressure-Temperature (P-T) Curve Update (MC0807 and MC0808)," dated February 24, 2004 (ADAMS Accession No. ML040550496).

ATTACHMENT 1

Proposed Technical Specifications Pages Markups



Curve No. 1
Minimum temperature for bottom head during mechanical heatup or cooldown following nuclear shutdown.

Curve No. 2
Minimum temperature for upper RPV and beltline during mechanical heatup or cooldown following nuclear shutdown.

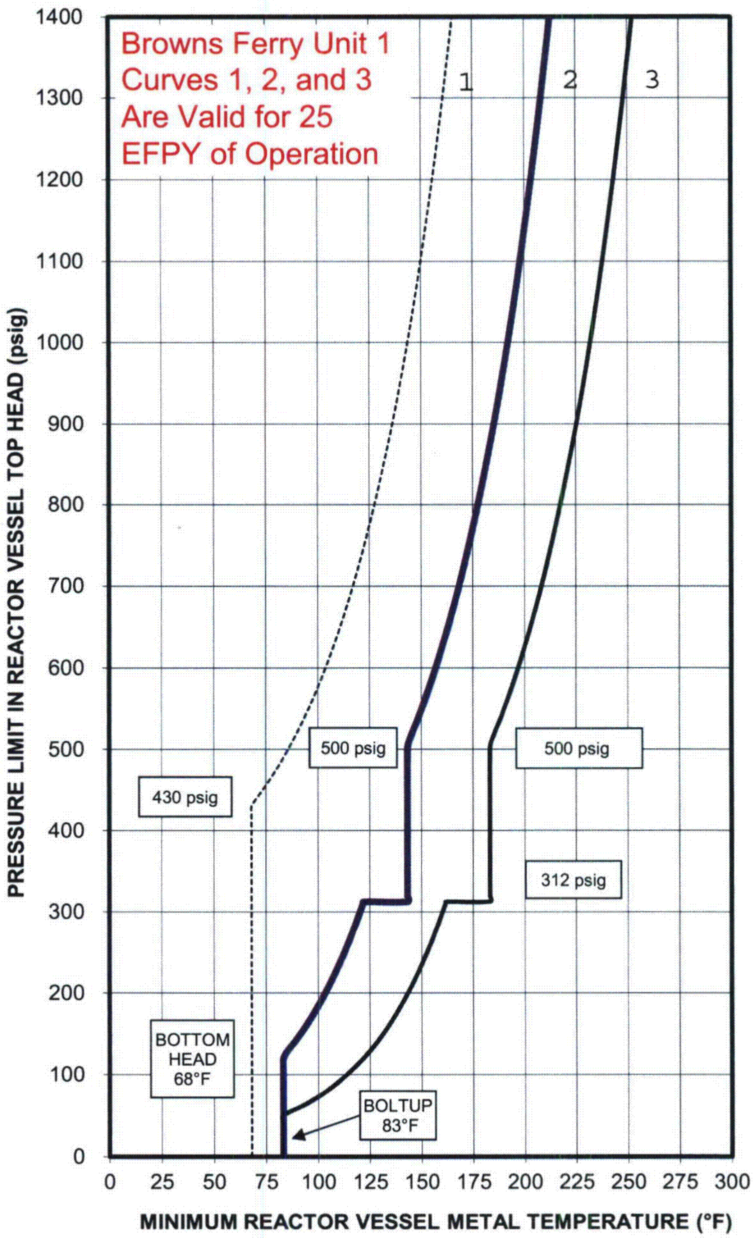
Curve No. 3
Minimum temperature for core operation (criticality).

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 ~~12~~ 25 EFPY.

The acceptable area for operation is to the right of the applicable curves.

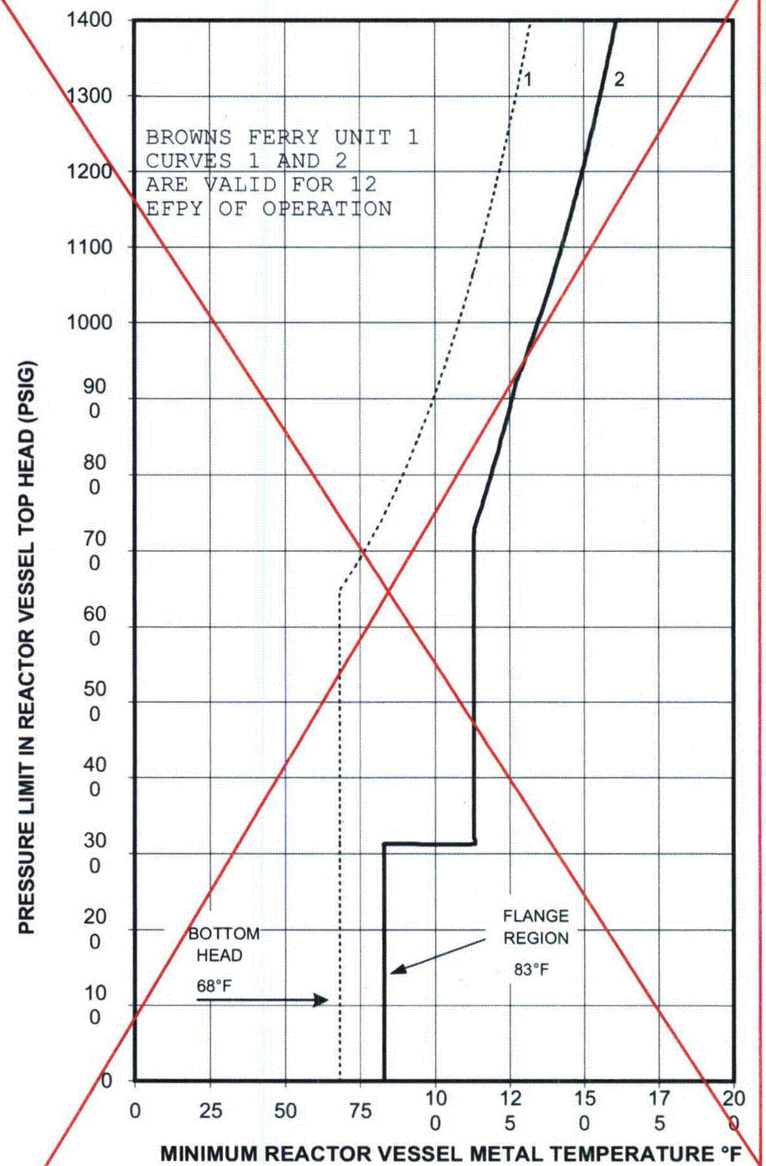
Replace with Insert 1

Figure 3.4.9-1
Pressure/Temperature Limits for Mechanical Heatup, Cooldown following Shutdown, and Reactor Critical Operations



Browns Ferry Unit 1
Curves 1, 2, and 3
Are Valid for 25
EFPY of Operation
1
2
3
500 psig
500 psig
430 psig
312 psig
BOTTOM HEAD 68°F
BOLTUP 83°F
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
0
25
50
75
100
125
150
175
200
225
250
275
300

Insert 1



Curve No. 1
Minimum temperature for bottom head during in-service leak or hydrostatic testing.

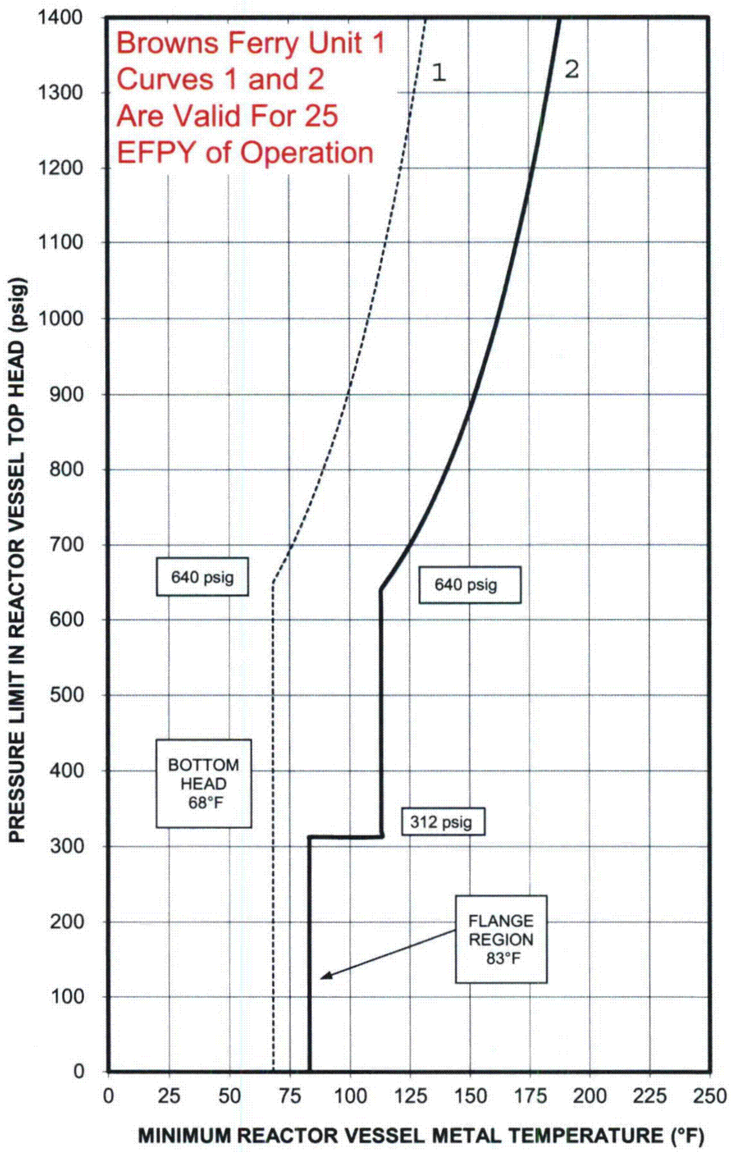
Curve No. 2
Minimum temperature for upper RPV and beltline during in-service leak or hydrostatic testing.

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 ~~12~~ 25 EFPY.

The acceptable area for operation is to the right of the applicable curves.

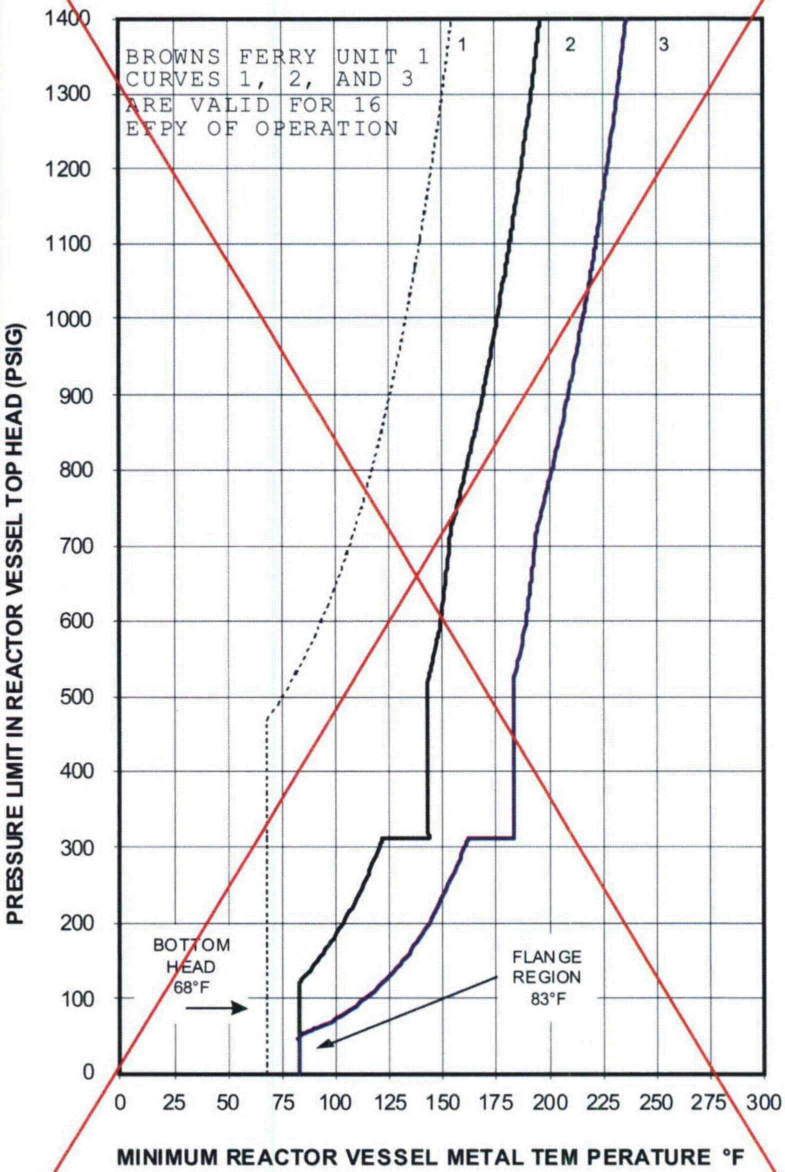
Replace with Insert 2

Figure 3.4.9-2
Pressure/Temperature Limits for
Reactor In-Service Leak and Hydrostatic Testing



Browns Ferry Unit 1
Curves 1 and 2
Are Valid For 25
EFPY of Operation
1
2
640 psig
640 psig
BOTTOM
HEAD
68°F
312 psig
FLANGE
REGION
83°F
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
0
25
50
75
100
125
150
175
200
225
250

Insert 2



Curve No. 1
Minimum temperature for bottom head during mechanical heatup or cooldown following nuclear shutdown.

Curve No. 2
Minimum temperature for upper RPV and beltline during mechanical heatup or cooldown following nuclear shutdown.

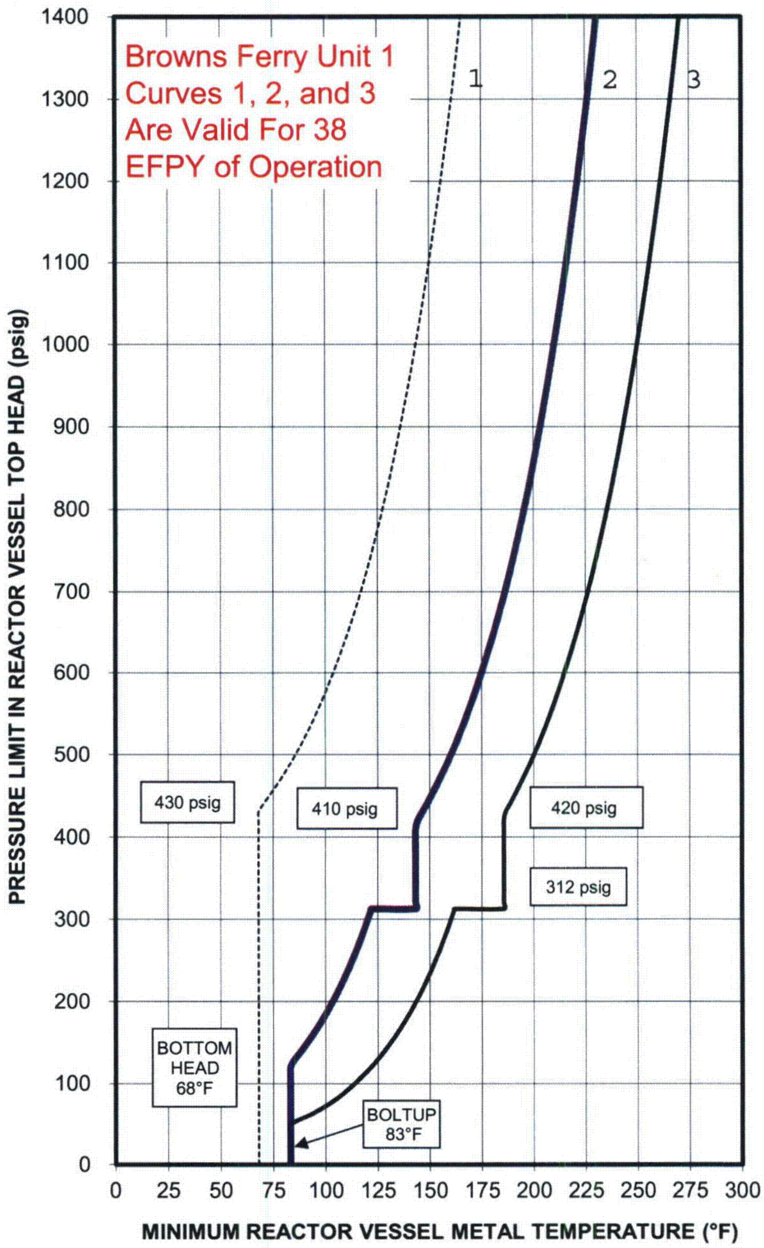
Curve No. 3
Minimum temperature for core operation (criticality).

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~~ 38 EFPY.

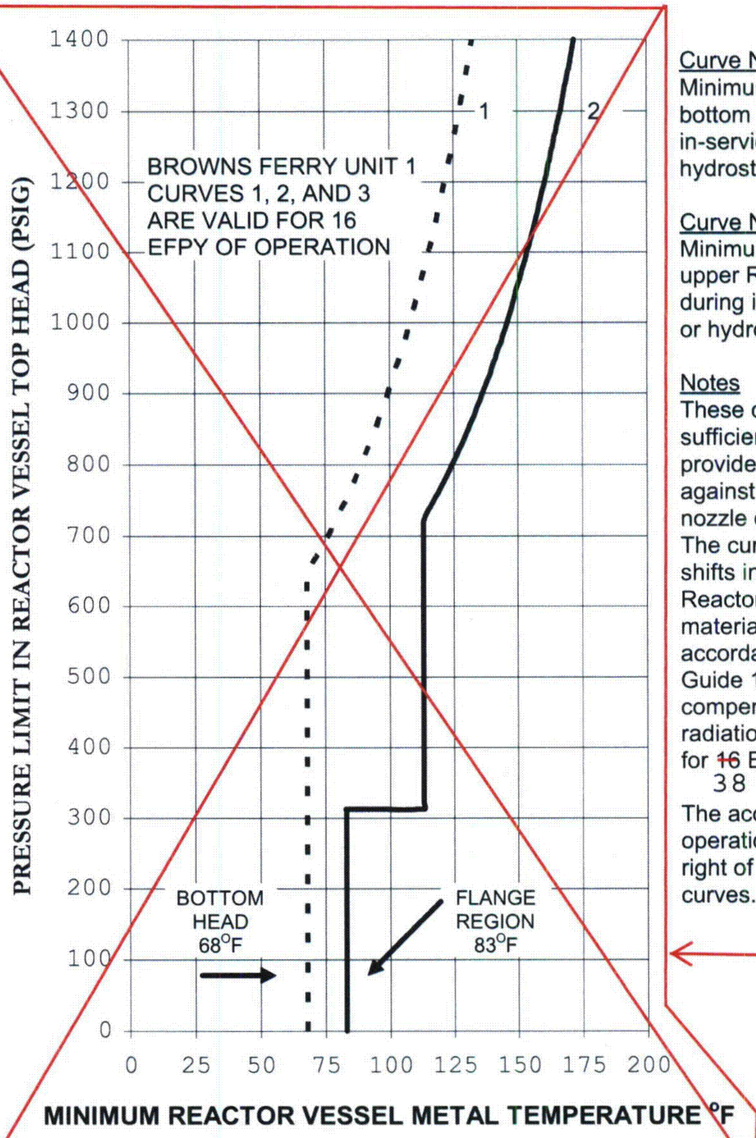
The acceptable area for operation is to the right of the applicable curves.

Replace with Insert 3

Figure 3.4.9-1
Pressure/Temperature Limits for Mechanical Heatup, Cooldown following Shutdown, and Reactor Critical Operations


Browns Ferry Unit 1
Curves 1, 2, and 3
Are Valid For 38
EFPY of Operation
1
2
3
430 psig
410 psig
420 psig
312 psig
BOTTOM
HEAD
68°F
BOLTUP
83°F
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
0
25
50
75
100
125
150
175
200
225
250
275
300
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

Insert 3



Curve No. 1
Minimum temperature for bottom head during in-service leak or hydrostatic testing.

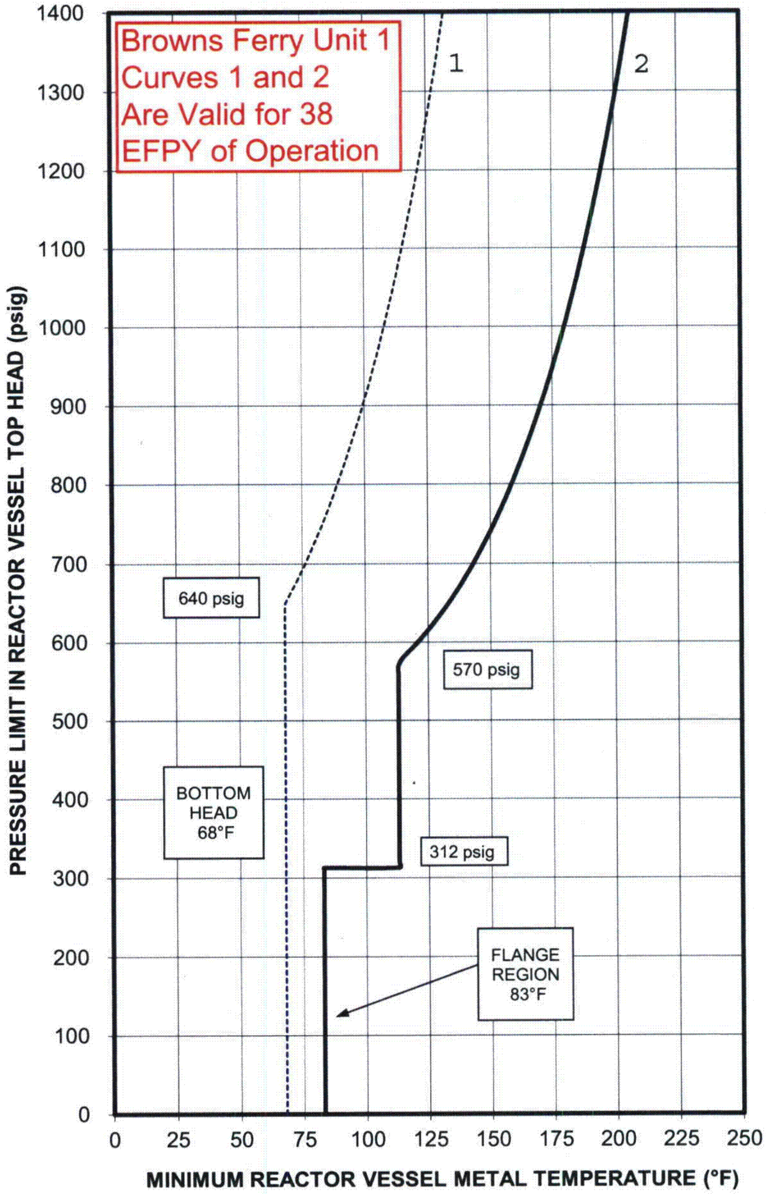
Curve No. 2
Minimum temperature for upper RPV and beltline during in-service leak or hydrostatic testing.

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~~ 38 EFPY.

The acceptable area for operation is to the right of the applicable curves.

Replace with Insert 4

Figure 3.4.9-2
Pressure/Temperature Limits for
Reactor In-Service Leak and Hydrostatic Testing



Browns Ferry Unit 1
Curves 1 and 2
Are Valid for 38
EFPY of Operation
1
2
640 psig
570 psig
BOTTOM
HEAD
68°F
312 psig
FLANGE
REGION
83°F
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

Insert 4

ATTACHMENT 2

Proposed Technical Specifications Bases Pages Markups (for information only)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS 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. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The pressure-temperature (P-T) curves included in the Technical Specifications have been developed to present steam-dome pressure versus minimum vessel metal temperature, incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. There are two sets of curves provided for Unit 1. The first set applies to operation up to **25**~~12~~ effective full power years (EFPY), and the second set applies to operation greater than **25**~~12~~ EFPY and less than **or equal to 38**~~16~~ EFPY, where **38**~~16~~ EFPY represents the end of the **renewed**~~40-year~~ license and **25**~~12~~ EFPY is provided as a midpoint between the current EFPY and **38**~~16~~ EFPY. The P-T curves are provided in Figure 3.4.9-1 and Figure 3.4.9-2, respectively. Figure 3.4.9-1 contains P-T limit curves for mechanical heatup or cooldown following nuclear shutdown (bottom head and upper RPV/beltline) and for core operation (criticality). Figure 3.4.9-2 contains P-T limit curves for inservice leakage and hydrostatic testing (bottom head and upper RPV/beltline). The maximum rate of change of reactor coolant temperature is contained in SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7.

(continued)

BASES

BACKGROUND (continued)

The P-T curves incorporate a fluence calculated in accordance with GE Licensing Topical Report NEDC-32983P (Ref. 10), which has been approved by the NRC and is in compliance with Regulatory Guide 1.190 (Ref. 11). The fluence represents an Extended Power Uprate (EPU) for the rated power of 3952 MW_t, and is conservatively applied for the rated power of 3458 MW_t. The **1998**~~1995~~ Edition of the ASME **Section XI** Boiler and Pressure Vessel Code including **2000**~~1996~~ Addenda was used in accordance with 10 CFR 50.55a.

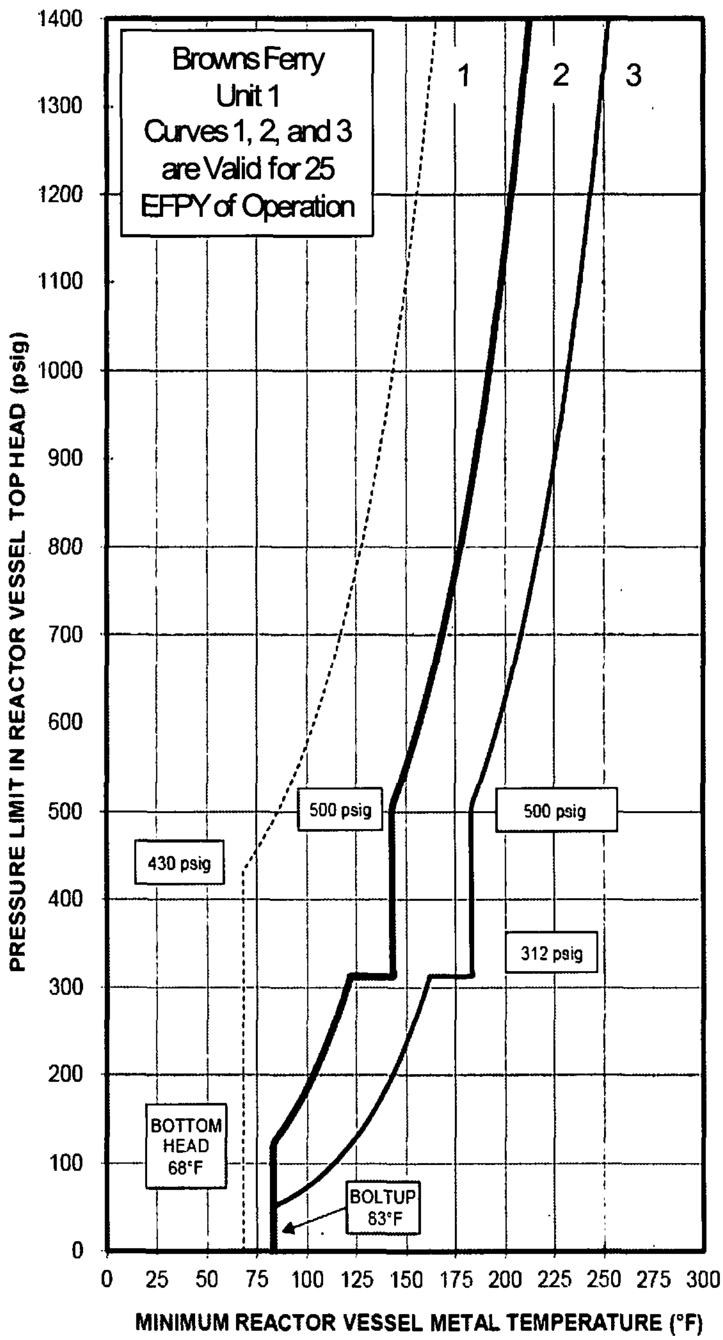
Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

(continued)

ATTACHMENT 3

Proposed Retyped Technical Specifications Pages



Curve No. 1
Minimum temperature for bottom head during mechanical heatup or cooldown following nuclear shutdown.

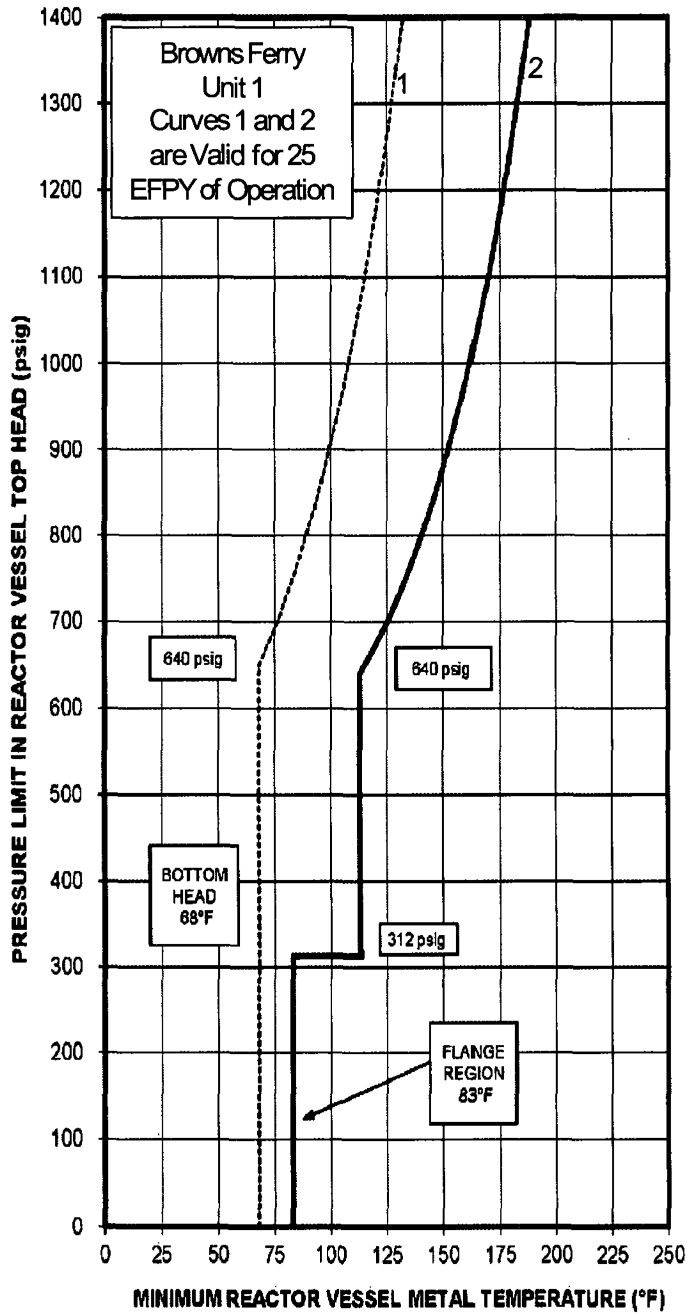
Curve No. 2
Minimum temperature for upper RPV and beltline during mechanical heatup or cooldown following nuclear shutdown.

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 25 EFPY.

The acceptable area for operation is to the right of the applicable curves.

Figure 3.4.9-1
Pressure/Temperature Limits for
Mechanical Heatup, Cooldown following Shutdown, and
Reactor Critical Operations



Curve No. 1

Minimum temperature for bottom head during in-service leak or hydrostatic testing.

Curve No. 2

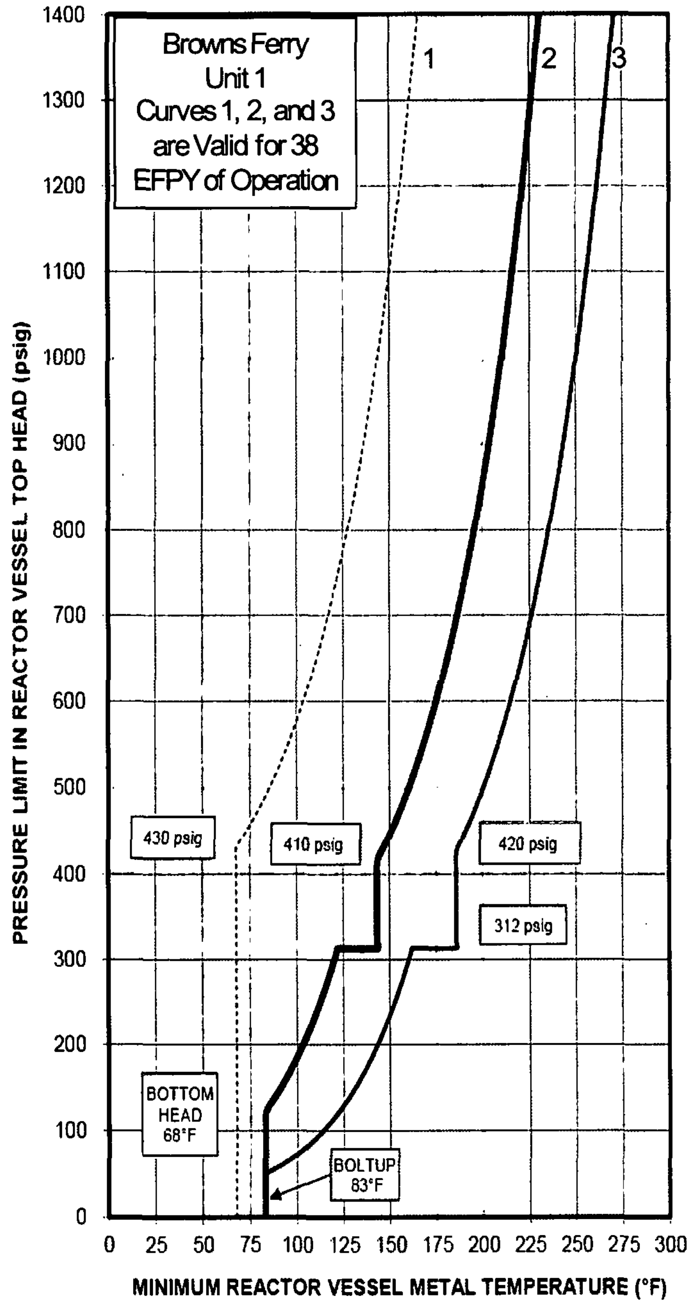
Minimum temperature for upper RPV and beltline during in-service leak or hydrostatic testing.

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 25 EFPY.

The acceptable area for operation is to the right of the applicable curves.

Figure 3.4.9-2
Pressure/Temperature Limits for
Reactor In-Service Leak and Hydrostatic Testing



Curve No. 1

Minimum temperature for bottom head during mechanical heatup or cooldown following nuclear shutdown.

Curve No. 2

Minimum temperature for upper RPV and beltline during mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

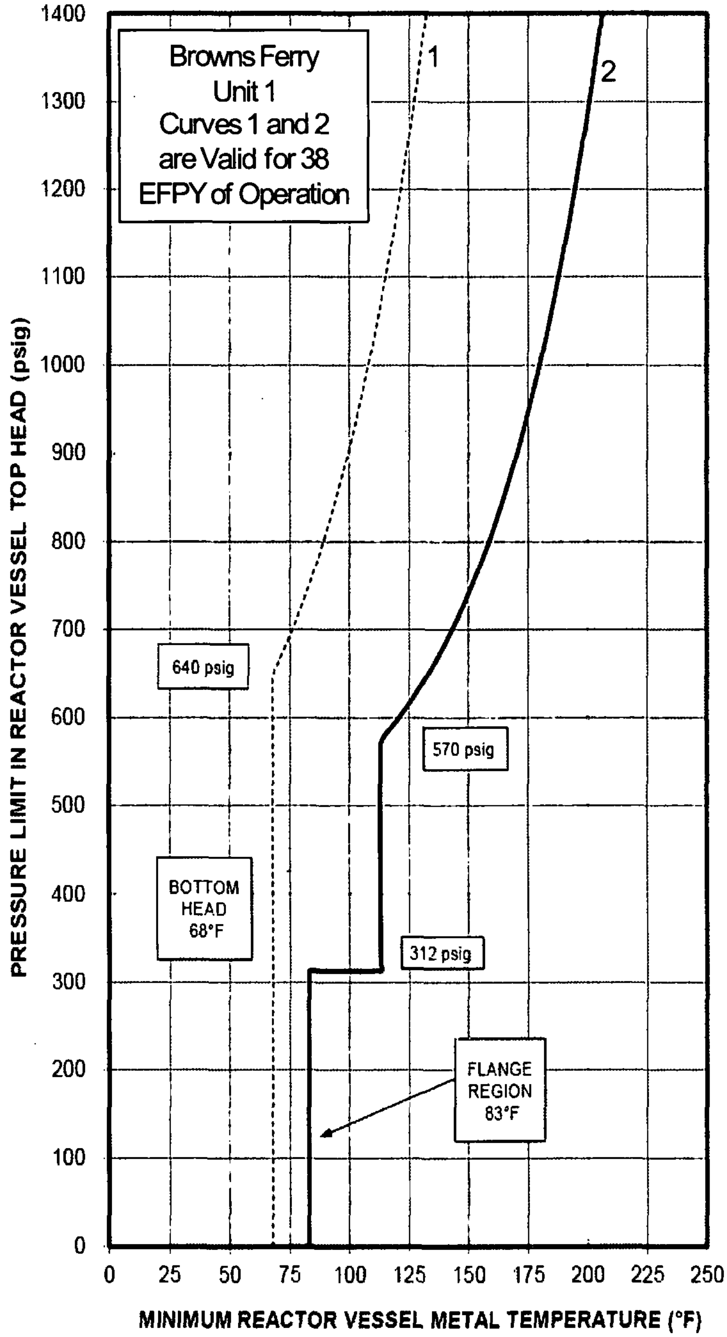
Minimum temperature for core operation (criticality).

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 38 EFPY.

The acceptable area for operation is to the right of the applicable curves.

Figure 3.4.9-1
Pressure/Temperature Limits for
Mechanical Heatup, Cooldown following Shutdown, and
Reactor Critical Operations



Curve No. 1

Minimum temperature for bottom head during in-service leak or hydrostatic testing.

Curve No. 2

Minimum temperature for upper RPV and beltline during in-service leak or hydrostatic testing.

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 38 EFPY.

The acceptable area for operation is to the right of the applicable curves.

Figure 3.4.9-2
Pressure/Temperature Limits for
Reactor In-Service Leak and Hydrostatic Testing

ATTACHMENT 4

Proposed Retyped Technical Specifications Bases Pages (for information only)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS 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. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The pressure-temperature (P-T) curves included in the Technical Specifications have been developed to present steam-dome pressure versus minimum vessel metal temperature, incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. There are two sets of curves provided for Unit 1. The first set applies to operation up to 25 effective full power years (EFPY), and the second set applies to operation greater than 25 EFPY and less than or equal to 38 EFPY, where 38 EFPY represents the end of the renewed license and 25 EFPY is provided as a midpoint between the current EFPY and 38 EFPY. The P-T curves are provided in Figure 3.4.9-1 and Figure 3.4.9-2, respectively. Figure 3.4.9-1 contains P-T limit curves for mechanical heatup or cooldown following nuclear shutdown (bottom head and upper RPV/beltline) and for core operation (criticality). Figure 3.4.9-2 contains P-T limit curves for inservice leakage and hydrostatic testing (bottom head and upper RPV/beltline). The maximum rate of change of reactor coolant temperature is contained in SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7.

(continued)

BASES

BACKGROUND (continued)

The P-T curves incorporate a fluence calculated in accordance with GE Licensing Topical Report NEDC-32983P (Ref. 10), which has been approved by the NRC and is in compliance with Regulatory Guide 1.190 (Ref. 11). The fluence represents an Extended Power Uprate (EPU) for the rated power of 3952 MW_t, and is conservatively applied for the rated power of 3458 MW_t. The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in accordance with 10 CFR 50.55a.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

(continued)

ATTACHMENT 5

Proposed Updated Final Safety Analysis Report Page Markups

Although little corrosion of plain carbon or low-alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at temperatures around 140°F. The 0.125-inch minimum-thickness cladding provides the necessary corrosion resistance during reactor shutdown and also helps maintain water clarity during refueling operations. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not required over its interior surfaces. Exterior, exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/16 inch. The interior surfaces of the top head and all carbon and low-alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16 inch. The vessel shape is designed to limit coolant retention pockets and crevices.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle, rather than ductile, manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1×10^{17} nvt with neutrons of energies in excess of 1 MeV. Since the material NDT temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDT temperature as low as possible. One way that this is accomplished is by selecting fine-grained steels and by using advanced fabrication techniques to minimize radiation effects. The as-fabricated initial NDT temperature for all carbon and low-alloy steel used in the main closure flanges, closure bolting material, and the shell and head materials connecting to these flanges, including the connecting circumferential weld material, is limited to a maximum of 10°F as determined by ASTM E208. For each main closure flange forging, a minimum of 1 tensile, 3 Charpy V-notch, and 2 drop weight test specimens have been tested from each of two locations about 180° apart on the flange. For all other carbon and low-alloy steel pressure-containing materials, including weld materials and the vessel support skirt material, the ~~as-fabricated~~ initial NDT temperature is no higher than ~~40~~ 56°F. A grain size of 5 or finer, as determined by the method in ASTM E112, is maintained.

Another way of minimizing any changes (elevating) to the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1×10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the 40-year design lifetime of the vessel. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and practical to maintain. The maximum calculated exposure for neutrons of 1 MeV or greater is ~~3.8×10^{17}~~ 1.58×10^{18} nvt per GEH Report No. 0000-0166-0632-R0.

for the surveillance capsule pulled in 1994

demonstrate that all materials and weld metal meet brittle fracture requirements at test temperature. Test specimens were prepared and tested with minimum impact energy requirements in accordance with Table N-421 and the general provisions of N-313, N-331, N-332, and N-511 of Section III of the ASME Boiler and Pressure Vessel Code. Prior to the Summer 1972 Addenda of the 1971 ASME Section III Boiler and Pressure Vessel Code, impact testing was not required on materials with a nominal section thickness of 1/2 inch or less. However, this 1/2 inch thickness exclusion was increased to 5/8 inch by the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, Summer 1972 Addenda. Therefore, after issuance of the Summer 1972 Addenda, impact testing is not required on materials with a nominal section thickness of 5/8 inch or less. The welding procedures used were qualified by impact testing of weld metal and heat affected zone to the same requirements as the base metal in accordance with N-541.

For the surveillance capsule pulled in 2011, per BWRVIP-27NP the Charpy Impact tests were conducted in accordance with ASTM Standards E185-82 and E 23-02.

B. Impact tests were not required for the following:

1. Bolting, including nuts, 1-inch nominal diameter or less,

2. Bars with a nominal cross-sectional area not exceeding 1 square inch,

3. Materials with a nominal (section) wall thickness of less than 1/2 inch or 5/8 inch (refer to paragraph 4.2.4.10.A),

4. Components including pumps, valves, piping, and fittings with a nominal inlet pipe size of 6-inch-diameter and less, regardless of thickness, and

5. Consumable insert material, austenitic stainless steel, and nonferrous materials.

C. Impact testing was not required on components or equipment pressure parts having a minimum service temperature of 250°F or more when pressured over 20 percent of the design pressure. Example: Steam line is excluded from brittle fracture test requirement since the steam temperature will be over 250°F when the steam line pressure is at the 20 percent design pressure.

D. Impact testing was not required on components or equipment pressure parts whose rupture could not result in a loss of coolant exceeding the capability of normal makeup systems to maintain adequate core cooling for the duration of a reactor shutdown and orderly cooldown.

E. These criteria apply to components and equipment pressure parts, including flange bolts of the reactor coolant pressure boundary, and do not apply to

Type of Cycle	No. of Cycles
Boltup	123
Design hydrostatic test at 1250 psig	130
Startup (100°F/hr heatup rate)	120
Daily reduction to 75 percent power	10,000
Weekly reduction to 50 percent power	2,000
Control rod worth test	400
Loss of feedwater heaters (80 cycles total)	
Turbine trip at 25 percent power	10
Feedwater heater bypass	70
Scram (200 cycles total)	
Loss of feedwater pumps, isolation valves close	10
Turbine trip, feedwater on, isolation valves stay open	40
Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1
Single safety relief valve blowdown	2
All other scrams	147
Improper start of cold recirculation loop	5
Sudden start of pump in cold recirculation loop	5
Shutdown (100°F/hr cooldown rate)	118
Hydrostatic test at 1563 psig	3
Unbolt	123

Stress analysis and load combinations for the reactor vessel are evaluated for the cycles listed above, with the conclusion that ASME code limits are satisfied. The details of assumed loading combinations are described in Appendix C for Class 1 equipment. It is possible that the specified number of cycles for some of the events listed above may be exceeded over the life of the plant. A plant procedure has been implemented at Browns Ferry to maintain surveillance on the number of cycles which have occurred and the resulting fatigue usage factors. When the fatigue usage factor reaches a value of 0.7, the procedure requires a reevaluation to be completed in a timely manner to assure that the allowable fatigue usage factor of 1.0 is not exceeded. Operating limits on pressure and temperature during inservice hydrostatic testing were established using as a guide Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, 1971, which was first added to the code in the summer 1972 addenda. The intent of Appendix G is to set criteria based on fracture toughness to provide a margin of safety against a nonductile failure. The resulting operating limits ensure that a large postulated surface flaw, having a depth of one-quarter of the material thickness and a length of one and one-half of the material thickness, can be safely accommodated in regions of the reactor vessel shell remote from discontinuities. Operating limits on temperature and pressure when the core is critical were established by using 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.C. An exemption from

The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in the development of the Unit 1 P-T curves. The P-T curve methodology includes the following: 1) the use of K_{IC} from Figure 4200-1 of Appendix A to Section XI and 2) the use of the M_m calculation in the ASME Code paragraph G-2214 of Appendix G to Section XI for a postulated defect normal to the direction of maximum stress.

specific requirements of 10 CFR Part 50, Appendix G is taken by use of ASME Code Case N-640 for Unit 2 and Unit 3. ASME Code Case N-640 permits the use of an alternative reference fracture curve K_{1c} for RPV materials for use in determining the PT limits. The PT limit curves based on the K_{1c} fracture toughness curve enhance overall plant safety by minimizing challenges to operators since requirements for maintaining a high vessel temperature during pressure testing are lessened. ASME Code Case N-588 methodology was also used as a basis for the PT curves. This code case 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 is incorporated into the ASME, Section XI Code, 1995 Edition, 1996 Addenda, which is the current code of record for the Unit 2 inservice inspection program. Since Unit 3 uses an earlier code of record for the inservice inspection program, Unit 3 implements the requirements of only the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix G to allow the use of the ASME Code Case N-588 methodology for PT curves. The operating limits are provided in the technical specifications for Browns Ferry. For the purpose of setting these operating limits, the initial RT_{NDT} (nil-ductility reference temperature) was determined from the impact test data taken in accordance with the requirements of the code to which the reactor vessels were designed and manufactured. The maximum NDT temperature allowed by the vessel specifications was 40°F. Although test data on beltline base material show lower NDT temperatures, an assumed RT_{NDT} of 40°F was used in the vessel beltline area, as well as the areas remote from the beltline because the generally accepted NDT temperature for electroslag welds used in the beltline longitudinal seams is 40°F.

23.1 The current operating limits on the pressure/temperature (P/T) curves in the technical specifications are based on the following (RT_{NDT}) values. Unit 1 has used 20°F for the (RT_{NDT}) value, Unit 2 has used 22°F for the (RT_{NDT}) value, and Unit 3 has used 10°F for the (RT_{NDT}) value.

For power uprated conditions, the estimated fluence was conservatively increased above the UFSAR end-of-life value. This fluence increase was estimated to be greater than proportional to the power increase, considering the changes in power distribution. The higher fluence was used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G in accordance with Regulatory Guide 1.99, Revision 2. The results of these evaluations indicated that:

(a) The upper shelf energy will remain greater than 50 ft-lb for the design life of the vessel and maintain the margin requirements of Appendix G.

(b) The effective full power year (EFPY) shift is slightly increased and, consequently requires a change in the adjusted reference temperature (ART), which is the initial RT_{NDT} plus the shift. The beltline material ART will remain within the 200 degree screening criterion.

The surveillance test plate 610-0127 was 139 in. long and 60 in. wide, and all excess material is under TVA control in the event that additional material is needed. It is estimated that enough extra material is available for several hundred additional Charpy specimens.

No weak direction specimens were included in the reactor vessel material surveillance program. All Charpy V-notch specimens were taken parallel to the direction of rolling. The majority of developmental work on radiation effects has been with longitudinal specimens. This is considered the best specimen to be used for determination of changes in transition temperature. At the low neutron fluence levels of BWR plants, no change in transverse shelf level is expected and transition temperature changes are minimal.

Replace with ①

Replace with ②

The specimens and neutron monitor wires were placed near core midheight adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall (see Subsection 3.3). The specimens were installed at startup or just prior to full-power operation. ~~For Units 1, 2, and 3, Integrated Surveillance Program (ISP) implementation and surveillance specimen schedule withdrawal and testing for the initial BWR 40-year operating period is governed and controlled by BWRVIP-86-A {Updated BWR Integrated Surveillance Program (ISP) Implementation Plan}, and the NRC's Safety Evaluation dated February 1, 2002. Surveillance specimen schedule withdrawal and testing during the license renewal period is governed and controlled by BWRVIP-116 (Integrated Surveillance Program (ISP) Implementation for License Renewal), and the BWRVIP response to NRC RAIs dated January 11, 2005.~~ Surveillance and chemistry data for all representative materials in the BWRVIP ISP have been consolidated into BWRVIP-135 {Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations.) ~~The withdrawal schedule for the second Unit 2 capsule located at azimuth 120° (to be withdrawn in 2011) and the third Unit 2 capsule (to be withdrawn during the license renewal period) will be in accordance with the ISP. Presently, there are no plans to withdraw any capsules from Units 1 or 3, as the BFN Unit 2 capsule provides the best representative weld material for both units, and the best representative plate material for Unit 3.~~ Supplemental Surveillance Program (SSP) Capsules A, B, D, G, E, and I provide the best representative weld material for Unit 1. Test results will provide the necessary data to monitor embrittlement for Units 1, 2, and 3. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end-of-life, the use of the capsules per the ISP meets the requirements of 10 CFR 50, Appendix H, and ASTM E185-82. Revisions to fluence calculations using data obtained from the surveillance capsule specimens will use an NRC approved methodology that meets Regulatory Guide 1.190. {By letter dated August 14, 2008 (EDMS Number L44 080828 014), NRC issued License Amendment 273 for BFN Unit 1, and by letter dated January 28, 2003 (EDMS Number L44 030204 001), NRC issued License Amendment Numbers 279 and 238, for BFN Units 2 and 3

For Units 1, 2, and 3, Integrated Surveillance Program (ISP) implementation and surveillance specimen schedule withdrawal and testing is governed and controlled by BWRVIP-86 Revision 1-A, the BWRVIP responses to NRC RAIs dated May 30, 2001, December 22, 2001, and January 11, 2005, and the NRC's Safety Evaluation dated February 1, 2002 (NOTE: BWRVIP-86, Revision 1-A was approved by the NRC and issued in October 2012, superseding both BWRVIP-86-A and BWRVIP-116).

A test specimen surveillance capsule (the second set of Unit 2 test specimens located at Azimuth 120°) was withdrawn in accordance with the ISP in 2011 during Unit 2 Refueling Outage 16 (U2R16) at approximately 23 EFPY of operation. An additional test specimen surveillance capsule is scheduled for withdrawal during the license renewal period, this being the third set of Unit 2 test specimens located at Azimuth 300°, which are currently scheduled for removal in the refueling outage closest to without exceeding 40 EFPY of operation. At the present time, this would correspond to Unit 2 Refueling Outage 24 (U2R24) in 2027. Presently, there are no plans to withdrawal any capsules from either Unit 1 or Unit 3, as per BWRVIP-135 the BFN Unit 2 capsules provide the best representative plate material for all three units and the best representative weld material for Units 2 and 3.