Commonwealth Edison Company
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ComEd

February 29, 2000

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Application for Amendment to Appendix A, Technical Specifications, Section 3/4.4.6, "Pressure/Temperature Limits, Reactor Coolant System," and Request for Exemption from 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"

Reference:

(1) Letter from J.P. Dimmette, Jr. (ComEd) to USNRC, "Request for an Amendment to Technical Specifications Section 3/4.6.K, "Primary System Boundary," Section /4.12.C, "Special Test Exceptions," and Request for Exemption from 10CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," dated November 12, 1999

(2) Letter from S. N. Bailey (USNRC) to O.D. Kingsley (ComEd), "Quad Cities – Exemption from the Requirements of 10CFR Part 50, Section 50.60(a) and Appendix G," dated February 4, 2000

(3) Letter from S. N. Bailey (USNRC) to O.D. Kingsley(ComEd), "Quad Cities - Issuance of Amendments- Revised Pressure-Temperature Limits," dated February 4, 2000

(4) Letter from R. M. Krich (ComEd) to USNRC, "Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity," dated July 30, 1998

Apo1

In accordance with 10 CFR 50.90,"Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company proposes changes to Appendix A, Technical Specifications (TS), to Facility Operating Licenses Nos. NPF-11 and NPF-18. Specifically we propose changes to TS Section 3/4.4.6, "Pressure/Temperature Limits, Reactor Coolant System," and its associated Bases Section. In support of this proposed change, in accordance with 10 CFR 50.12, "Specific exemptions," we are also requesting an exemption from 10 CFR 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The NRC in References 2 and 3, recently approved similar changes for Quad Cities Nuclear Power Station.

The proposed TS changes revise for Units 1 and 2, the pressure-temperature (P-T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the Reactor Pressure Vessel (RPV). The proposed changes replace the current RPV P-T limit TS Figures 3.4.6.1-1 and 3.4.6.1-1a, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure," with three recalculated RPV P-T limit figures that are applicable to 32 Effective Full Power Years (EFPYs). The use of 32 EFPYs RPV P-T limit figures conservatively bounds each unit. In addition, the following TS changes are proposed.

- The hydrostatic/leak test exception footnote is removed,
- The 30 minute requirement of TS 4.4.6.1.1 is applied to proposed TS Figure 3.4.6.1-1b,
- Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," is relocated to the LaSalle County Station Updated Final Safety Analysis Report (UFSAR), and
- The description contained in Bases Section B3/4.4.6 is revised.

Also, for Unit 1 only, the proposed change decreases the minimum allowable RPV/head flange temperature from 80 to 72 degrees Fahrenheit (F) and also lowers the RPV/head flange surveillance temperatures by 8 degrees F.

The requested exemption will allow the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Cases N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640,"Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," in calculating RPV P-T limits. The procedures and methodology that were used to recalculate the RPV P-T in these proposed changes were revised based on the ASME Code Cases cited above. Therefore, it is additionally requested that the requested exemption be approved prior to or concurrent with the approval of the proposed TS changes.

Attachment G of this proposed change includes two General Electric Company reports containing proprietary information about this submittal. Requests for withholding this information from disclosure, in accordance with 10 CFR 2.790(a)(4), are provided in the preface of each report.

The information supporting the proposed TS changes and exemption requests is subdivided as follows.

1. Attachment A gives a description and safety analysis of the proposed change.

2. Attachment B includes the marked-up TS pages with the requested changes indicated.

3. Attachment C describes our evaluation performed in accordance with 10 CFR 50.92(c), which provides information supporting a finding of no significant hazards consideration.

4. Attachment D provides information supporting an Environmental Assessment for the proposed change.

5. Attachment E provides the information justifying the Exemption Request.

6. Attachment F provides a technical basis for revised P-T Limit Curve Methodology.

7. Attachment G provides GE Nuclear Energy Reports, "Pressure Temperature Curves for ComEd LaSalle Unit 1" (GE-NE-B13-02057-05 Rev. 0) and "Pressure Temperature Curves for ComEd LaSalle Unit 2" (GE-NE-B13-02057-00-06 Rev. 0).

The proposed TS changes and exemption requests have been reviewed by the LaSalle County Station Plant Operations Review Committee (PORC) and approved by the Nuclear Safety Review Board (NSRB) in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If there are any questions or comments concerning this letter, please refer them to Mr. R. R. Brady, Jr., Director, LaSalle Licensing and Compliance, at 630-663-7205.

Sincerely,

R. M. Krich
Vice President – Regulatory Services

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B: Marked-up TS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: Exemption Request
Attachment F: Technical Basis for Revised P-T Limit Curve Methodology
Attachment G: GE Nuclear Energy Reports, "Pressure Temperature Curves for ComEd LaSalle Unit 1" (GE-NE-B13-02057-05 Rev. 0) and "Pressure Temperature Curves for ComEd LaSalle Unit 2" (GE-NE-B13-02057-00-06 Rev. 0).

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS)

IN THE MATTER OF:)

COMMONWEALTH EDISON (COMED) COMPANY) Docket Numbers

LASALLE COUNTY STATION - UNITS 1 and 2) 50-373 and 50-374

SUBJECT: Application for Amendment to Appendix A, Technical Specifications, Section 3/4.4.6, "Pressure/Temperature Limits, Reactor Coolant System"

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

OFFICIAL SEAL
LAURA KAMINSKAS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 6-1-2002

R. M. Krich
Vice President – Regulatory Services

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 29 day of

February, 2000

Notary Public

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED TECHNICAL SPECIFICATION CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company proposes a change to Appendix A, Technical Specifications (TS), to Facility Operating License Nos. NPF-11 and NPF-18. The proposed changes are to TS Section 3/4.4.6, "Pressure/Temperature Limits, Reactor Coolant System," and its associated Bases Section.

The proposed changes revise for Units 1 and 2, the pressure-temperature (P-T) limits for heatup, cooldown, critical operation, and inservice leak and hydrostatic test limitations for the Reactor Pressure Vessel (RPV). The proposed changes replace the current RPV P-T limit TS Figures 3.4.6.1-1 and 3.4.6.1-1a, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure," with three recalculated RPV P-T limit figures for each unit that are applicable to 32 Effective Full Power Years (EFPYs). The use of 32 EFPYs RPV P-T limit figures conservatively bounds each unit. In addition, the following TS changes are proposed.

- The hydrostatic/leak test exception footnote is removed,
- The 30 minute requirement of TS 4.4.6.1.1 is applied to proposed TS Figure 3.4.6.1-1b,
- Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," is relocated to the LaSalle County Station Updated Final Safety Analysis Report (UFSAR), and
- The description contained in Bases Section B3/4.4.6 is revised.

Also, for Unit 1 only, the proposed change decreases the minimum allowable RPV/head flange temperature from 80 to 72 degrees Fahrenheit (F) and also lowers the RPV/head flange surveillance temperatures by 8 degrees F.

The proposed changes are described in detail in Section E of this Attachment. The marked-up TS and Bases Section pages are shown in Attachment B.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS Section 3/4.4.6, "Pressure/Temperature Limits, Reactor Coolant System," requires the following.

- The reactor coolant system pressure is limited in accordance with the RPV metal temperature limits shown on TS Figures 3.4.6.1-1 and 3.4.6.1-1a,
- The reactor coolant maximum heatup or cooldown in any one hour period is 100 degrees F,
- The maximum temperature change in any one hour period is less than or equal to 20 degrees F during inservice leak/hydrostatic testing, and
- The RPV/head flanges are greater than or equal to 80 (Unit 1) or 86 (Unit 2) degrees F when the RPV head studs are under tension.

These requirements are applicable at all times except for the specific conditions described in a footnote.

If the reactor coolant system pressure and temperature is found to exceed the limits, they must be restored to within the limits within 30 minutes, an engineering evaluation must be performed to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system, and a determination must be performed that the reactor coolant system remains acceptable for continued operation. If the above conditions can not be met within 30 minutes, then be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

Surveillance requirements are provided to ensure the following.

- At least once per 30 minutes, during system heatup, cooldown, inservice leak and hydrostatic testing, the reactor coolant system temperature and pressure shall be determined to be within the allowable limits specified on TS Figures 3.4.6.1-1 and 3.4.6.1-1a,
- Within 15 minutes prior to control rod withdrawal for reactor criticality, the reactor coolant system temperature and pressure shall be determined to be within the allowable limits specified on TS Figures 3.4.6.1-1 and 3.4.6.1-1a,

- The reactor vessel material specimens shall be removed and examined in accordance with TS Table 4.4.6.1.3-1 and that the results of the fluence determinations be used to update TS Figures 3.4.6.1-1 and 3.4.6.1-1a, and

- The reactor vessel/head flanges shall be verified to be greater than 80 (Unit 1) or 86 (Unit 2) degrees F during specified reactor coolant system conditions.

C. BASES FOR THE CURRENT REQUIREMENT

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9, "Mechanical Systems and Components," of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The reactor vessel materials have been tested to determine their initial reference temperature nil ductility transition (RT_{NDT}) and the initial RPV P-T limits. Reactor operation and resultant fast neutron radiation will cause an adjustment of the reference temperature nil ductility transition and a shift in the RPV P-T limits. The Adjusted Reference Temperature (ART) has been predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Material." TS Figure 3.4.6.1-1 provides the predicted RPV P-T limit curves, including the predicted ART, at the end of sixteen EFPYs and TS Figure 3.4.6.1-1a provides the predicted RPV P-T limit curves, including the predicted ART, at the end of thirty two EFPYs.

The actual ART of the reactor vessel materials will be established periodically during operation by removing and evaluating irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel. The RPV P-T limit curves of TS Figures 3.4.6.1-1 and 3.4.6.1-1a will be adjusted, as required, on the specimen data and the recommendations of Regulatory Guide 1.99, Revision 2.

D. NEED FOR REVISION OF THE REQUIREMENT

We submitted a request for a license amendment on July 14,1999. The submittal requested to operate both LaSalle County Station units at 3489 MWT, which represented an increase of 5 percent rated core thermal power. We recently contracted with General Electric Company (GE) to recalculate the P-T limit curves for LaSalle County Station Units 1 and 2, including the effect of the proposed power uprate on fast neutron radiation of the RPV.

The methodology used to generate the P-T limit curves was similar to the methodology to generate the current P-T limit curves shown on TS Figures 3.4.6.1-1 and 3.4.6.1-1a. However, several improvements were made to the P-T limit curve methodology. The improvements included, but were not limited to, the incorporation of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Cases N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," and N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1." ASME B&PV Code Case N-640 allows the use of K_{IC} rather than K_{Ia} to determine T-RT_{NDT}. ASME B&PV Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors for axial and circumferential flaws. A detailed description of the methodology used and the results obtained are contained in Attachment G to this letter. We have determined that the use of ASME B&PV Code Cases N-640 and N-588 will require prior NRC review and approval of an exemption to 10 CFR 50.60. The proposed exemption request is contained in Attachment E to this letter.

The resultant benefits of the proposed changes would include the following.

- Reduction in the challenges to operators in conducting pressure testing of the reactor coolant system at less than or equal to 212 degrees F and maintaining the reactor coolant system within a narrow temperature band,

- Personnel safety; conducting inspections at lower coolant temperatures,

- Potential dose savings by increasing the effectiveness of inspectors in the containment at lower ambient temperatures,

- Potential outage critical path schedule savings by the reduction of time to achieve reactor coolant system temperature and RPV pressure requirements for testing, and

- Elimination of the hydrostatic/leak test exception specified in the footnote to TS 3.4.6.1 Applicability.

E. DESCRIPTION OF THE PROPOSED CHANGES

Unit 1 Proposed Changes

The proposed changes add a new page reference to the TS List of Figures Index and removes two page references from the TS List of Tables Index.

The proposed changes to TS 3.4.6.1 Limiting Condition for Operation are to revise the references to the P-T TS figures and to revise the allowable reactor vessel/head flange temperatures from 80 degrees F to 72 degrees F.

The proposed change to TS 3.4.6.1 Applicability is to remove the referenced footnote.

The proposed change to TS Surveillance 4.4.6.1.1 is to remove the reference to Curves A and B, and to include the reference to TS Figure 3.4.6.1-1b.

The proposed change to TS Surveillance 4.4.6.1.2 is to revise the referenced P-T figure number.

The proposed changes to TS Surveillance 4.4.6.1.3 are to remove the reference to the relocated TS Table 4.4.6.1.3-1 and to revise the references to the P-T TS figures.

The proposed change to TS Surveillance 4.4.6.1.4 is to revise the allowable reactor vessel/head flange temperature from 80 degrees F to 72 degrees F and also lowers the surveillance temperatures by 8 degrees.

The proposed changes to TS Figures 3.4.6.1-1 and 3.4.6.1-1a are to replace the current 16 and 32 EFPY curves with three recalculated 32 EFPY curves and to change the associated descriptions contained on each page.

The proposed change to TS Table 4.4.6.1.3-1 is to relocate it to the UFSAR.

The proposed change to TS Bases Section 3/4.4.4 is to reword the references to the P-T figures and to relocate Table B 3/4.4.6-1,"Reactor Vessel Toughness," to the UFSAR.

Unit 2 Proposed Changes

The proposed changes add a new page reference to the TS List of Figures Index and removes two page references from the TS List of Tables Index.

The proposed changes to TS 3.4.6.1 Limiting Condition for Operation is to revise the references to the P-T TS figures.

The proposed change to TS 3.4.6.1 Applicability is to remove the referenced footnote.

The proposed change to TS Surveillance 4.4.6.1.1 is to remove the reference to Curves A and B, and to include the reference to TS Figure 3.4.6.1-1b.

The proposed change to TS Surveillance 4.4.6.1.2 is to revise the referenced P-T figure number.

The proposed changes to TS Surveillance 4.4.6.1.3 are to remove the reference to the relocated TS Table 4.4.6.1.3-1 and to revise the references to the P-T TS figures.

The proposed changes to TS Figures 3.4.6.1-1 and 3.4.6.1-1a are to replace the current 16 and 32 EFPY curves with three recalculated 32 EFPY curves and to change the associated descriptions contained on each page.

The proposed change to TS Table 4.4.6.1.3-1 is to relocate it to the UFSAR.

The proposed change to TS Bases Section 3/4.4.4 is to reword the references to the P-T figures and to relocate Table B 3/4.4.6-1,"Reactor Vessel Toughness," to the UFSAR.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P-T limits. Appendix G of 10 CFR 50 specifies fracture toughness and testing requirements for reactor vessel material in accordance with the ASME B&PV Code and that the beltline material in the surveillance capsules be tested in accordance with Appendix H of 10 CFR 50. Appendix G of 10 CFR 50 also requires the prediction of the effects of neutron irradiation on the vessel embrittlement by calculating the ART and Charpy upper shelf energy. Generic Letter 88-11, "NRC Position on Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations," requests that the methods in Regulatory Guide 1.99, Revision 2, be used to predict the effect of neutron irradiation on the reactor vessel material. Appendix H of 10 CFR 50 requires the establishment of a surveillance program to periodically withdraw surveillance capsules from the reactor vessel.

The current P-T limits for LaSalle County Station were approved by the NRC in Amendment No. 71 for Unit 1 and Amendment No. 55 for Unit 2. The NRC approval of the current P-T limits was based on their conformance to the requirements of Appendices G and H of 10 CFR 50. The NRC also noted that current P-T limits satisfied Generic Letter 88-11 since the method in Regulatory Guide 1.99, Revision 2 was used to calculate ART.

These P-T limits were developed based on the methodology specified in ASME B&PV Code Section XI, Appendix G, as modified by ASME B&PV Nuclear Code Cases N-588 and N-640. Code case N-588 allows the use of alternate procedures for defining the postulated flaw orientation and for calculating the applied stress intensity factors for the postulated axial and circumferential flaws. Code case N-640 allows the use of alternate material fracture toughness when determining minimum vessel temperatures, that is K_{Ic} rather than K_{Ia} values defined in ASME B&PV Code Section XI, Appendix A. For the beltline materials, the RT_{NDT} was adjusted based on the analytical methods specified in Regulatory Guide 1.99, Revision 2. The ART was determined using the bounding fluence values contained in our July 14, 1999 submittal for power uprate. The beltline material unirradiated RT_{NDT} values and best estimate chemistries used were consistent with our Generic Letter 92-01, "Reactor Vessel Structural Integrity," Request for Additional Information responses submitted July 30, 1998. The Unit 1 limiting beltline material did not change as a result of this correction. Also, as a result of additional certified material test reports, the RT_{NDT} of the limiting material in the Unit 1 closure flange assembly was reduced to 12 °F from 20 °F. Details of these changes are provided in Tables 4-1 and 4-2 of the Unit 1 report in Attachment G. Details of the analytical methods and evaluations performed to calculate the P-T limits are provided in Attachment G. As noted previously, we have determined that the use of ASME B&PV Code Cases N-640 and N-588 will require prior NRC review and approval of an exemption to 10 CFR 50.60. The proposed exemption request is contained in Attachment E to this letter.

The proposed change to eliminate the hydrostatic/leak test exception specified in the footnote to TS Section 3.4.6.1 Applicability, is a result of the increase in test temperature margin between the P-T Limit curve and 212 degrees F. This change will limit hydrostatic/leak test reactor coolant temperature to less than or equal to 200 degrees F per TS Table 1.2, "Operational Conditions," for Cold Shutdown.

The proposed change to TS Table 4.4.6.1.3-1 to relocate it to the UFSAR is consistent with the guidance contained in Generic Letter 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," dated January 4, 1991.

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding impact on any previous submittals and have determined that there is one impact on an outstanding previous submittal. Our July 14, 1999 power uprate submittal proposed changes to TS Table B 3/4.4.6-1,"Reactor Vessel Toughness." This proposed change supercedes our July 14, 1999 submittal by relocating this table to the UFSAR.

H. SCHEDULE REQUIREMENTS

We request approval of this amendment to support the Unit 2 outage in the fall 2000 (L2R08).

ATTACHMENT B
Proposed Technical Specification Changes

MARKED-UP TECHNICAL SPECIFICATION PAGES

NPF-11	NPF-18
XIX	XIX
XXII	XXII
XXIII	XXIII
3/4 4-16	3/4 4-17
3/4 4-17	3/4 4-18
3/4 4-18	3/4 4-19
Insert Page 3/4 4-18	Insert Page 3/4 4-19
3/4 4-18a	3/4 4-19a
Insert Page 3/4 4-18a	Insert Page 3/4 4-19a
3/4 4-19	3/4 4-20
Insert Page 3/4 4-19	Insert Page 3/4 4-20
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6

LIST OF FIGURES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1-1 and 3.4.6.1-1a (1) curves A for hydrostatic or leak testing; (2) curves B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C for operations with a critical core other than low power PHYSICS TESTS, with: Figure 3.4.6.1-1b Figure 3.4.6.1-1a

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to ~~80~~ 72°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1 and 3.4.6.1-1a curves A or B, as applicable, at least once per 30 minutes. and 3.4.6.1-1b

*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means the average coolant temperature limit of Table 1.2 for Cold Shutdown and Hot Shutdown may be increased to 212°F.

SURVEILLANCE REQUIREMENTS (Continued) b.

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figures 3.4.6.1-1 and 3.4.6.1-1a curves C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a. , and 3.4.6.1-1b

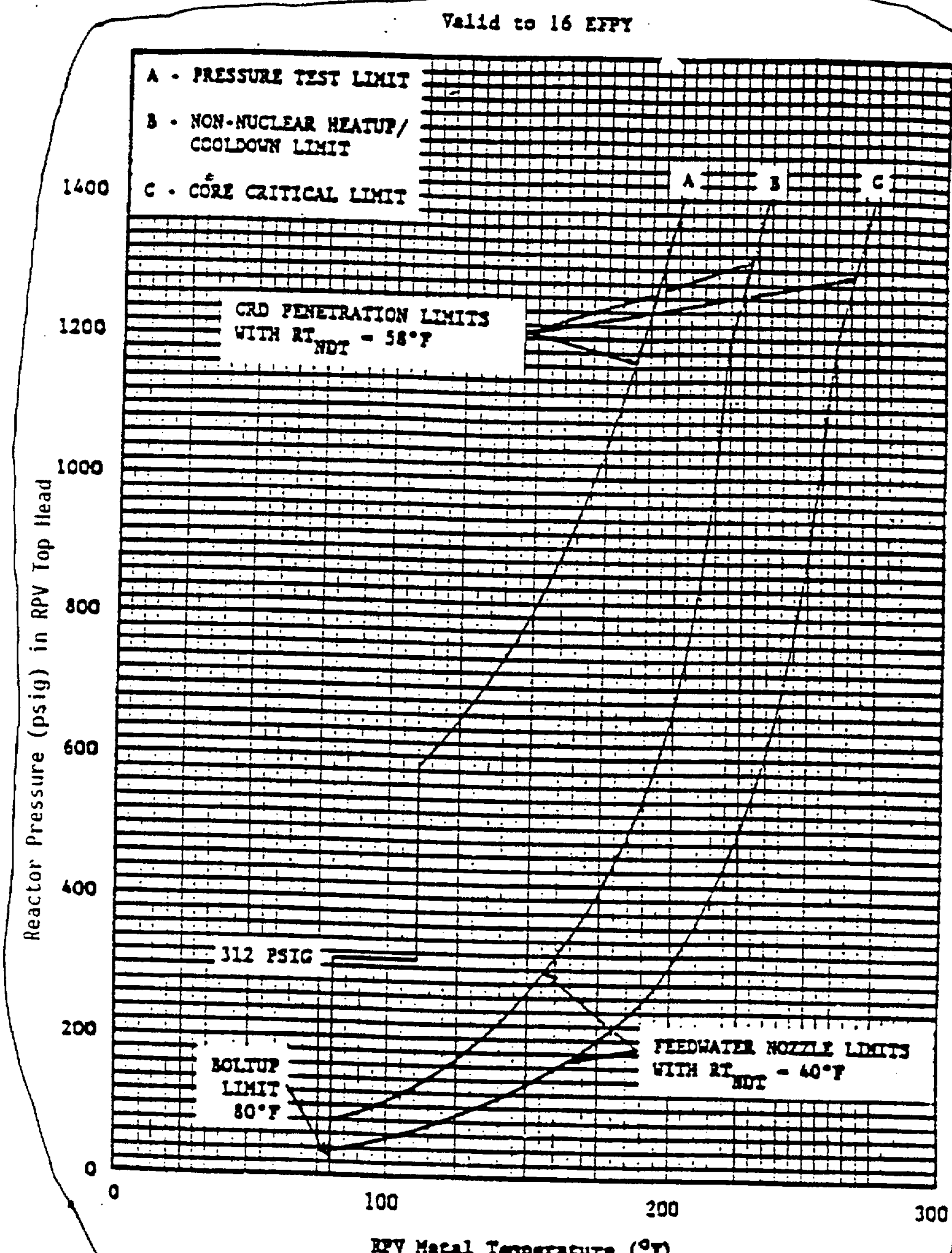
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F: 72

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

92 1. ≤ 100°F, at least once per 12 hours.

77 2. ≤ 85°F, at least once per 30 minutes.

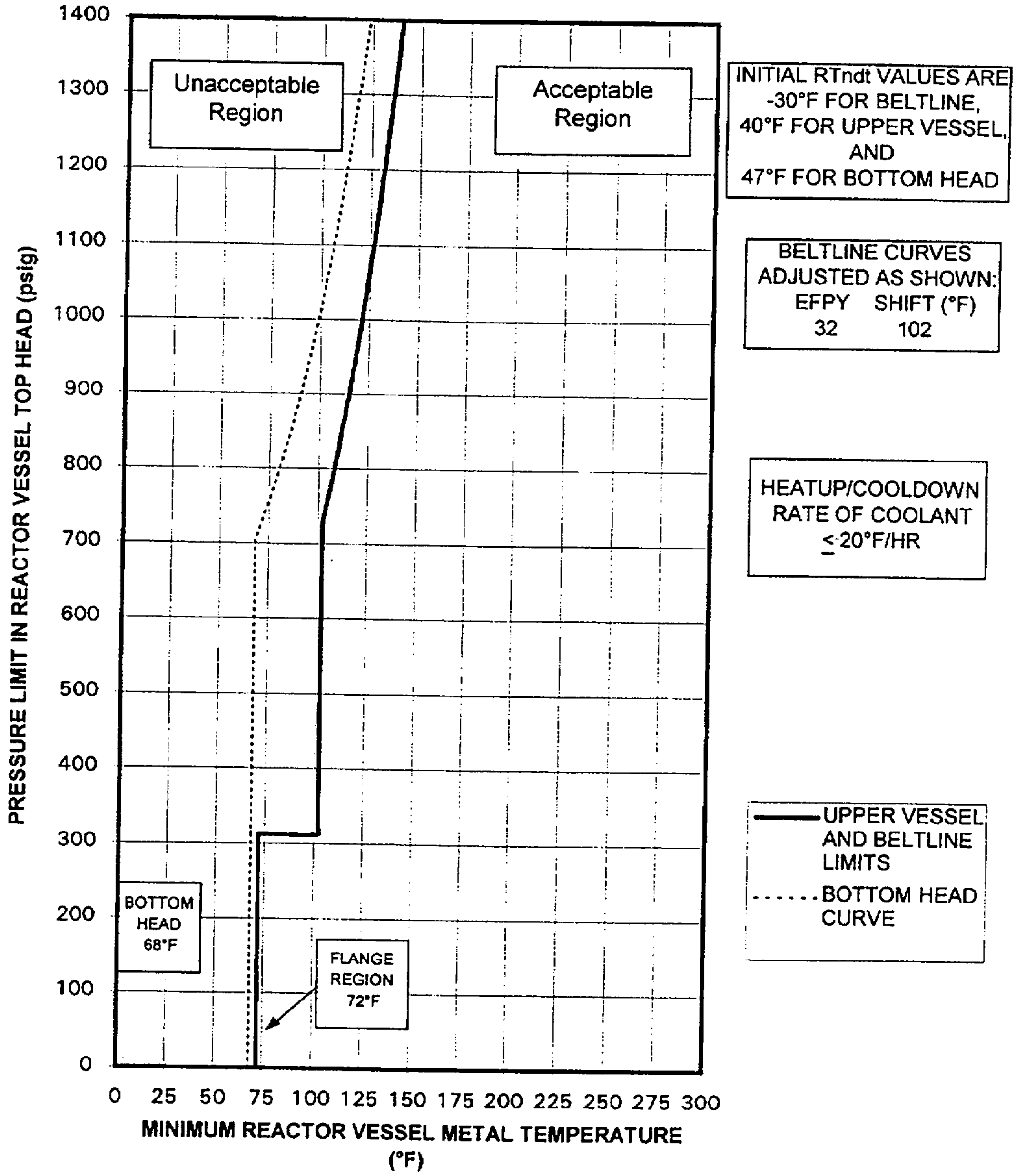
b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



Minimum Reactor Vessel Metal Temperature vs
Reactor Vessel Pressure

Figure 3.4.6.1-1

Replace with new Figure 3.4.6.1-1

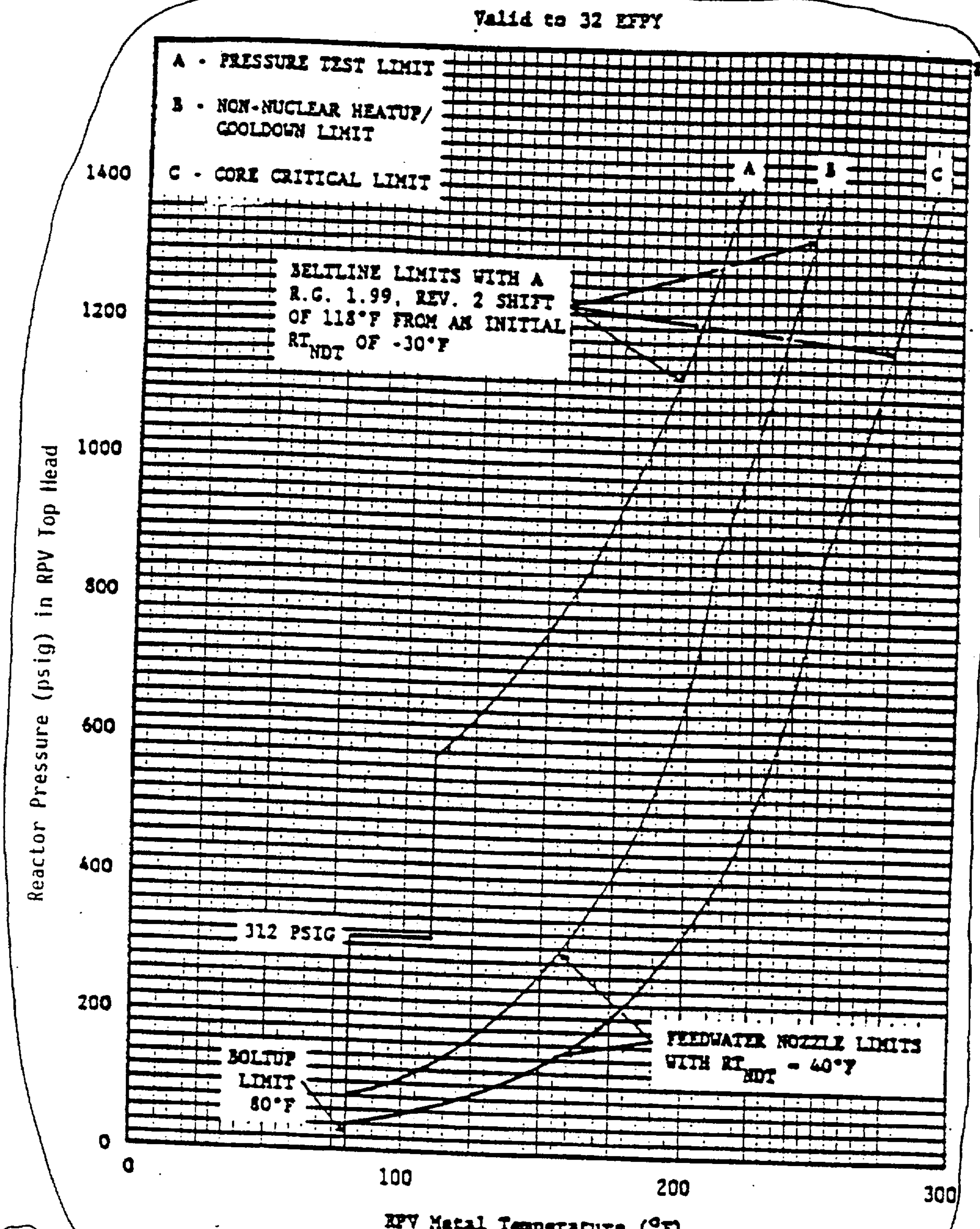


P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

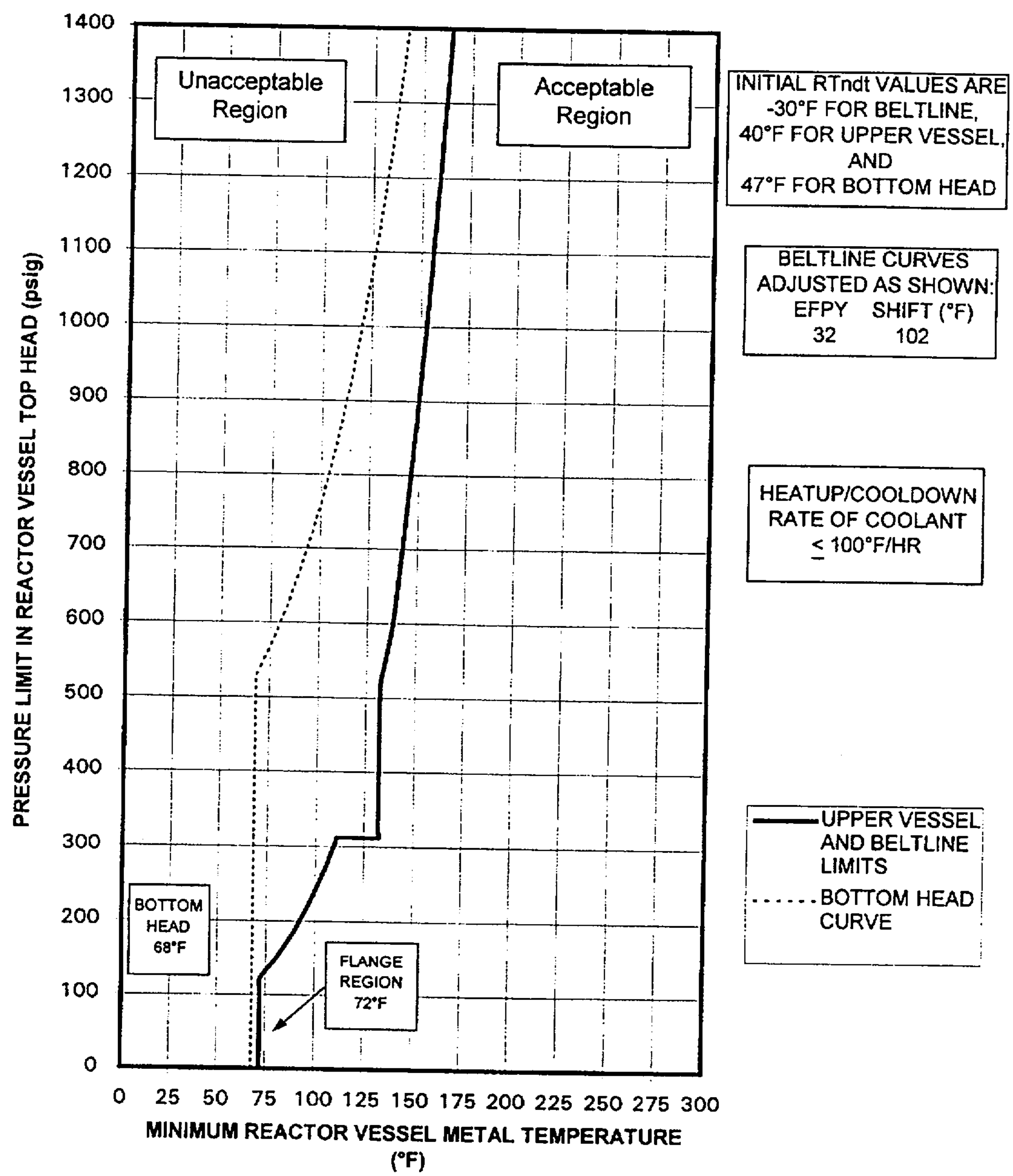
Figure 3.4.6.1-1



Minimum Reactor Vessel metal Temperature vs.
Reactor Vessel Pressure

Figure 3.4.6.1-1a

Replace with new Figure 3.4.6.1-1a



P-T Curve for Heatup by Non-Nuclear Means,
Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

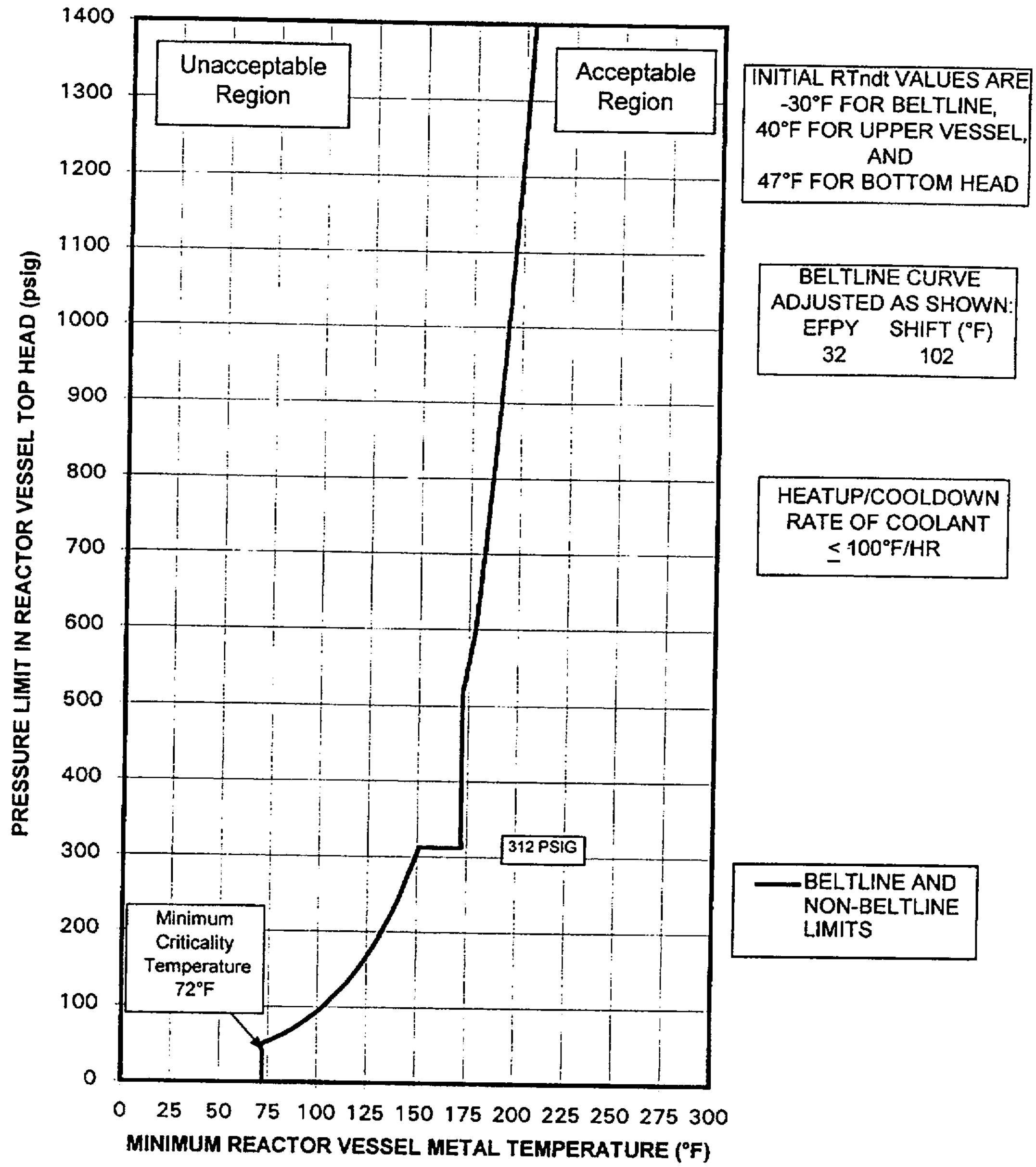
Figure 3.4.6.1-1a

Table 4.4.6.1.3-1

Reactor Vessel Material Surveillance Program Withdrawal Schedule

Specimen holder	Vessel location	Lead factor	Withdrawal time (Effective Full Power Years)
117C4936G010	300°	0.6	6
117C4936G011	120°	0.6	15
117C4936G012	30°	0.6	Spare
Neutron Dosimeter	30°		1st Refuel Outage

Replace with new Figure 3.4.6.1-1b



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

3.4.6.1-1a, and 3.4.6.1-1b

The specimen withdrawal schedule is provided in UFSAR section 4.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

provided in section 4 of the UFSAR

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are ~~shown in Table B 3/4.4.6-1~~. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/temperature limit curve, Figure s 3.4.6.1-1, includes predicted adjustments for this shift in RT_{NDT} at the end of ~~sixteen~~ thirty-two effective full power years (EFPY)~~, while Figure 3.4.6.1-1a includes predicted adjustments in RT_{NDT} at the end of life fluence~~.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with

PRESSURE/TEMPERATURE LIMITS (Continued)

, , and 3.4.6.1-1b

confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1 and 3.4.6.1-1a shall be adjusted, as required, on the basis of the specimen data and the recommendations of Regulatory Guide 1.99, Rev. 2.

, and 3.4.6.1-1b , ed

The ~~pressure-temperature~~ limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-1a for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50. ~~For reactor criticality and for inservice leak and hydrostatic testing, General Electric "Transient Pressure Rise Affecting Fracture Toughness Requirement for Boiling Water Reactors," NEDO-21778-A, December 1978, and "Protection Against Non-Ductile Failure" of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda.~~ e

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE

COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (°F)	MAXIMUM ΔRT_{NDT} (°F)	MIN. UPPER SHELF (ft-lb)	MAX. EOL RT_{NDT}
Plate	SA-533,Gr.B,Cl.1	C5978-2	0.11	0.010	+23	30**	118	
Plate	SA-533,Gr.B,Cl.1	C6345-2	0.15	0.012	-35**	49	153	+72
Weld	3-308-A,B,C	IP3571/3978	0.37	0.017	-30	124	***	+94

NON-BELTLINE

COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA-533,Gr.B,Cl.1	C6003-2	+12
Bottom Head Dollar Plate	SA-533,Gr.B,Cl.1	C6003-3	+58
Bottom Head Radial Plates	SA-533,Gr.B,Cl.1	C5328-1	+10
Top Head Dollar Plate	SA-533,Gr.B,Cl.1	C7343-1	-10
Top Head Side Plates	SA-533,Gr.B,Cl.1	C7376-2	-10
Top Head Flange	SA-508,Cl.2	ACT-USS-4P	+20
Vessel Flange	SA-508,Cl.2	2V-659ATF-112	+20
Feedwater Nozzle	SA-508,Cl.2	#174W-3,Q2Q14VW	+40
Weld	15-308	NA/KOIB	0
Closure Stud	POH-16C,Gr.B and ATSM-A-540	14716	+70

* Combination of the higest starting RT_{NDT} plate and the highest ΔRT_{NDT} plate.

** These values are given only for the benefit of calculating the end-of-life (EOL) RTNDT.

***Not available.

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LIST OF FIGURES

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LIST OF TABLES (Continued)

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1-1 and 3.4.6.1-1a; (1) curves A for hydrostatic or leak testing; (2) curves B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C for operations with a critical core other than low power PHYSICS TESTS, with: Figure 3.4.6.1-1b Figure 3.4.6.1-1a

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 86°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1 and 3.4.6.1-1a curves A or B, as applicable, at least once per 30 minutes.
, and 3.4.6.1-1b

*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means, the average coolant temperature limit of Table 1.2 for Cold Shutdown and Hot Shutdown may be increased to 212°F.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figures 3.4.6.1-1 and 3.4.6.1-1a curves C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality. b e

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1, and 3.4.6.1-1a. , and 3.4.6.1-1b

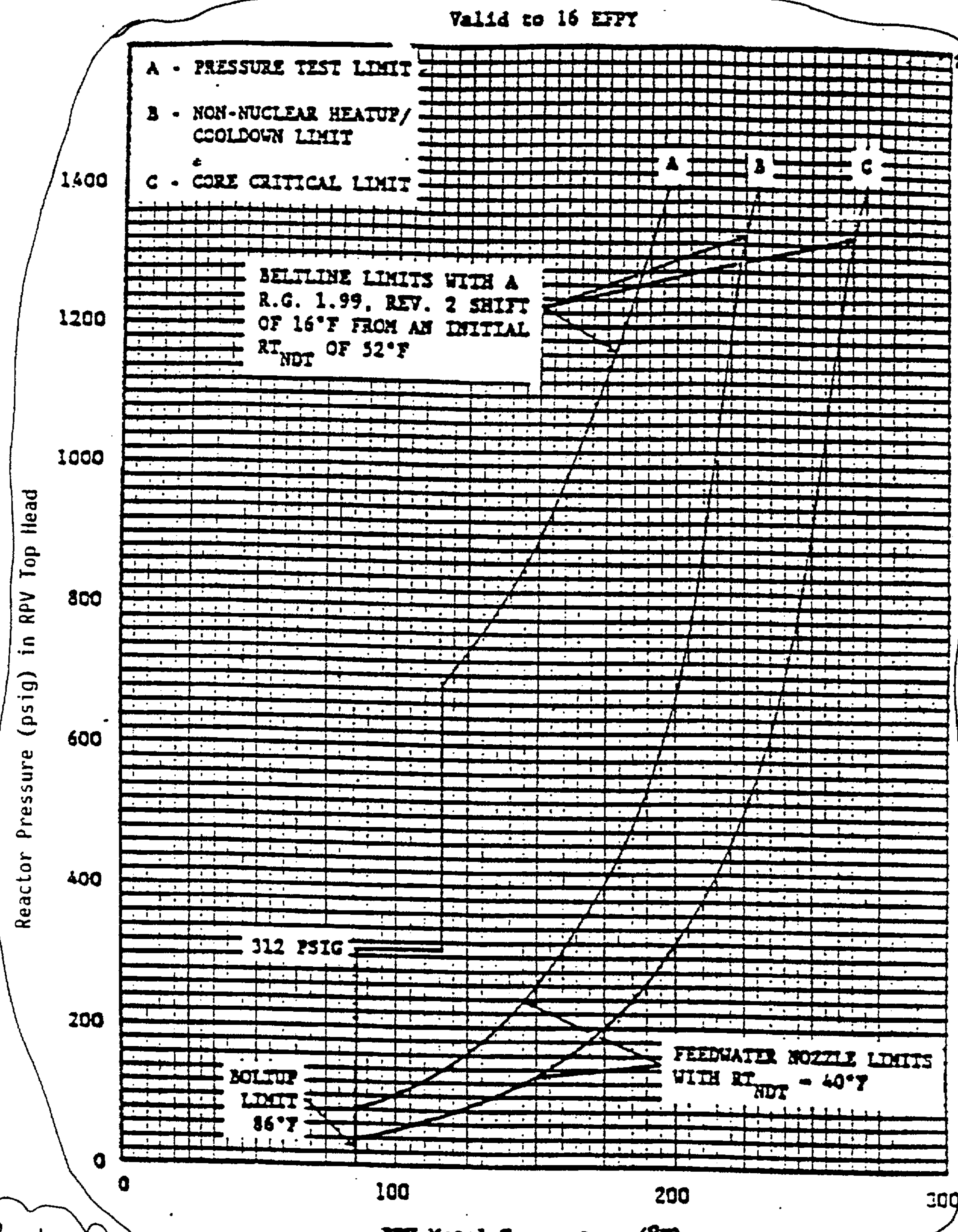
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 86°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

   1. ≤ 106°F, at least once per 12 hours.

   2. ≤ 91°F, at least once per 30 minutes.

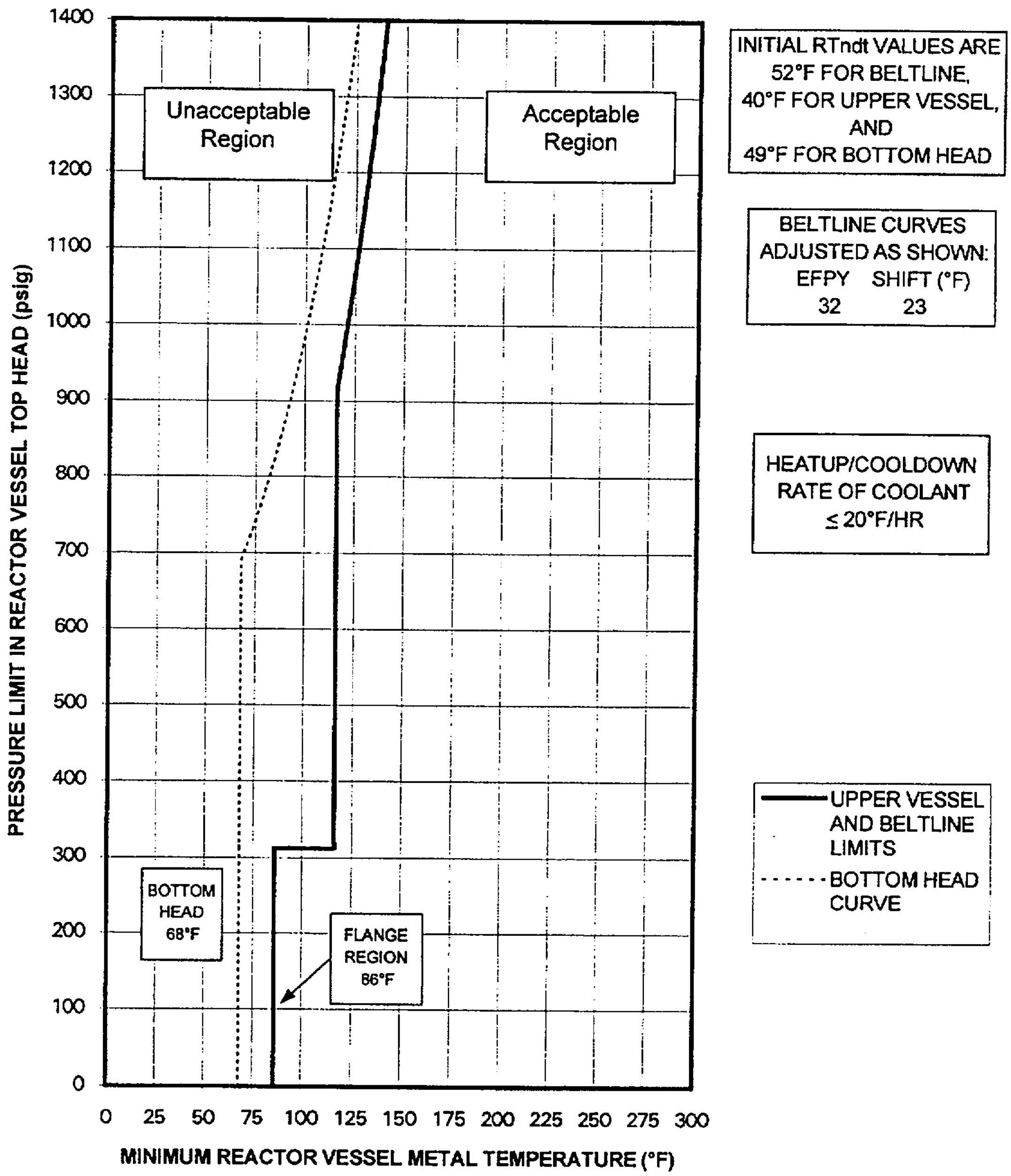
b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



Minimum Reactor Vessel Metal Temperature vs
Reactor Vessel Pressure

Figure 3.4.6.1-1

Replace with new Figure 3.4.6.1-1

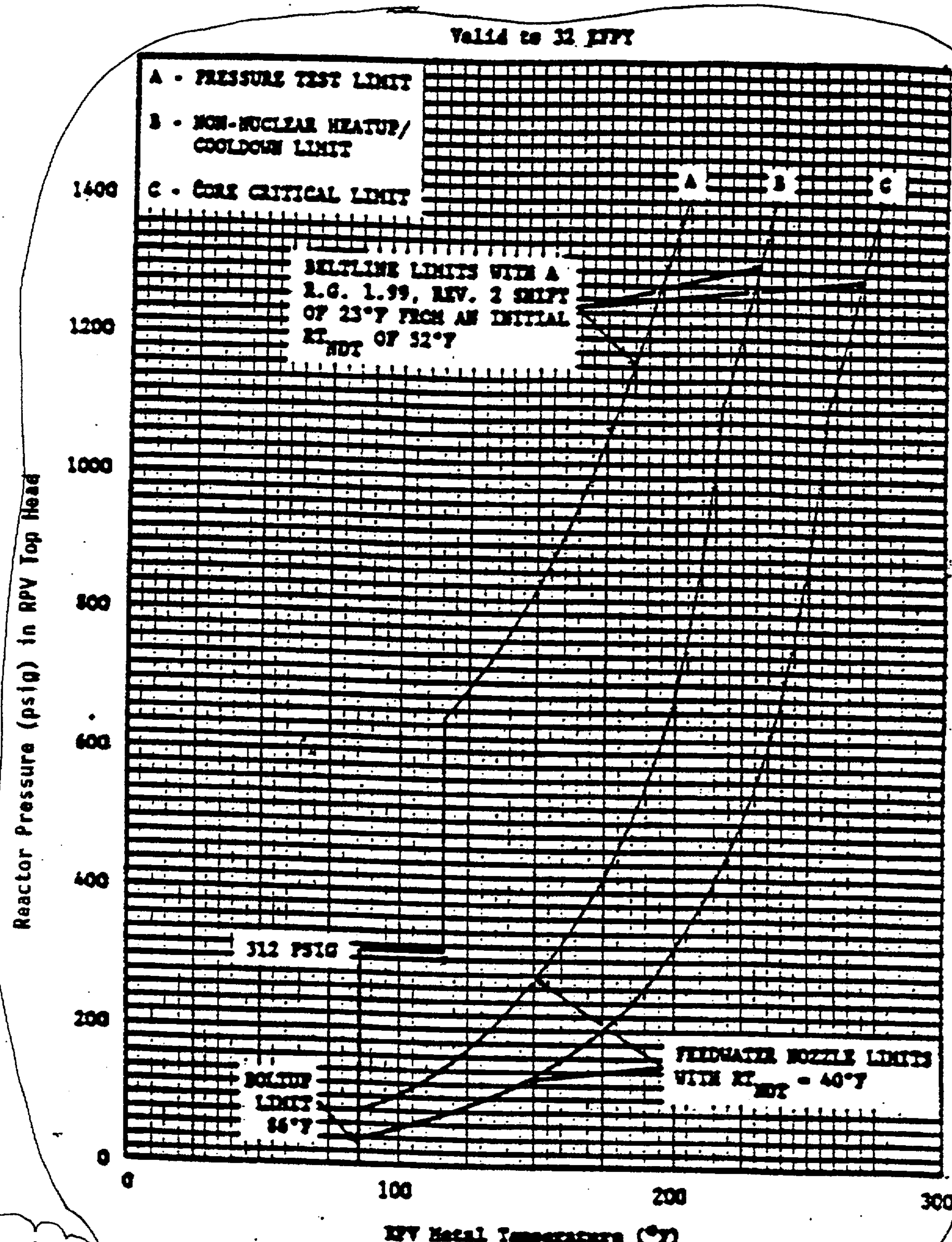


P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

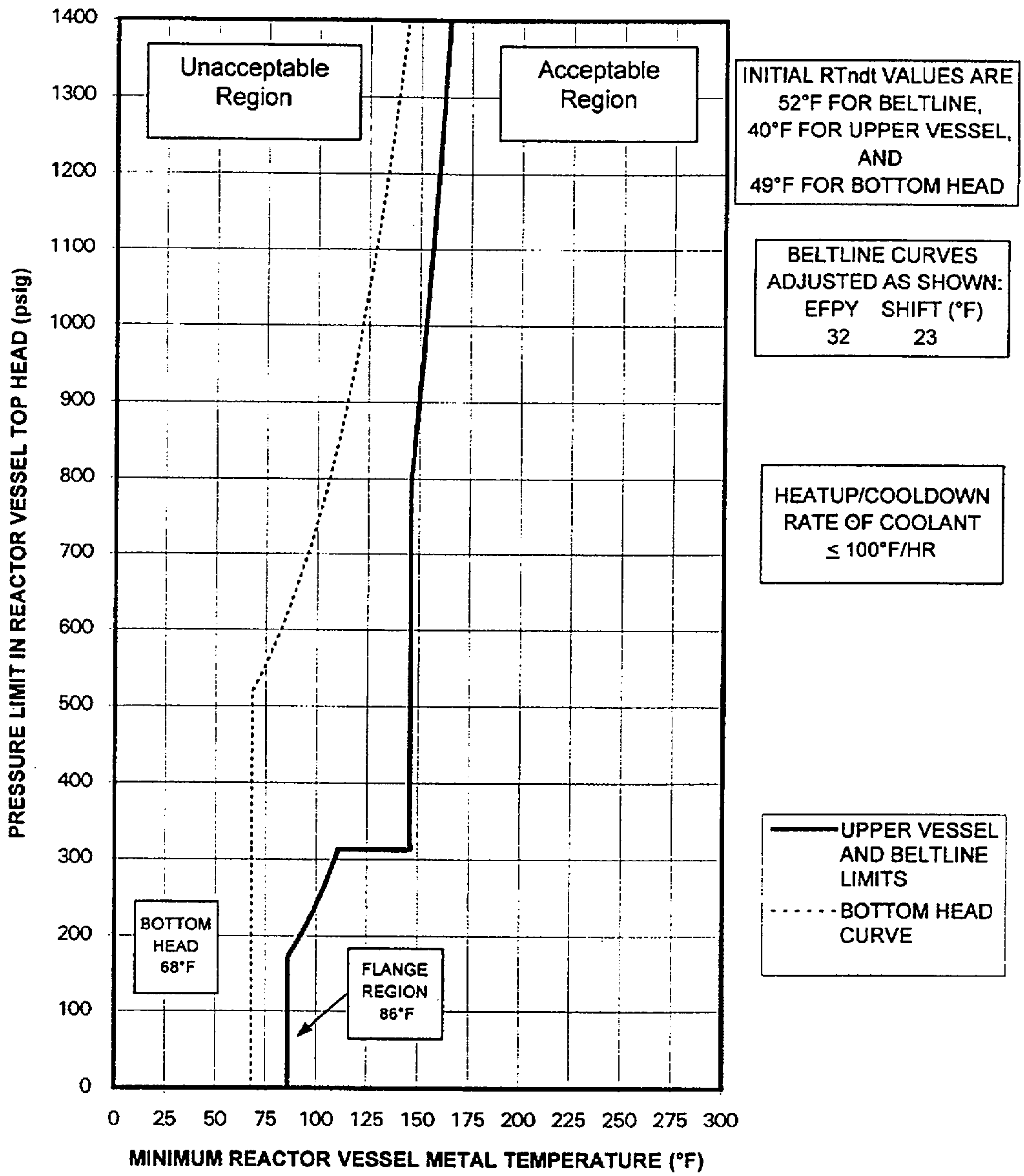
Figure 3.4.6.1-1



Replace with new Figure 3.4.6.1-1a

Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

Figure 3.4.6.1-1a



P-T Curve for Heatup by Non-Nuclear Means,
Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1a

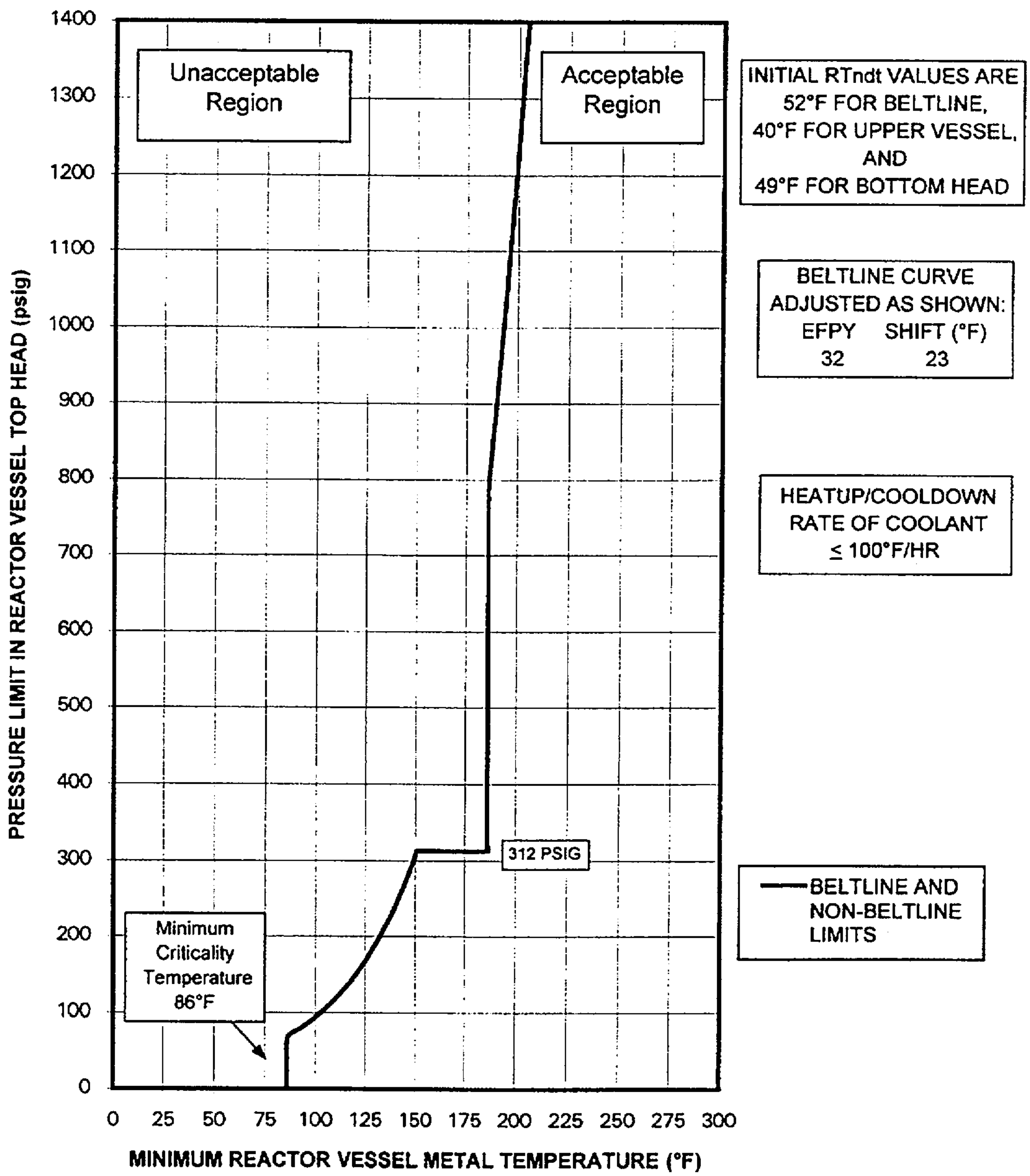
Replace with new Figure 3.4.6.1-1b

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

SPECIMEN HOLDER*	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFFECTIVE FULL POWER YEARS)
Capsule 1	300°	0.6	6
Capsule 2	120°	0.6	15
Capsule 3	30°	0.6	Spare
Neutron Dosimeter	30°	-	1st Refueling Outage

*Each capsule includes an Fe, Ni, and Cu flux wire. The neutron dosimeter contains three Cu and three Fe flux wires.



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

3.4.6.1-1a, and 3.4.6.1-1b

The specimen withdrawal schedule is provided in UFSAR section 4.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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S

thirty-two

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PRESSURE/TEMPERATURE LIMITS (Continued) and 3.4.6.1-1b

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The pressure-temperature limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-1a for reactor criticality and for inservice leak and hydrostatic testing have been established using the requirements of Appendix G to 10 CFR Part 50. for reactor criticality and for inservice leak and hydrostatic testing, General Electric "Transient Pressure Rise Affecting Fracture Toughness Requirement for Boiling Water Reactors," NEDO-21778-A, December 1978, and "Protection Against Non-Ductile Failure" of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda.

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Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

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A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

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COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (°F)	MAXIMUM Δ RT_{NDT} (°F)	MIN. UPPER SHELF (ft-lb)	MAX. EOL RT_{NDT}
Plate	SA-533,Gr.B,Cl.1	C9404-2	0.07	0.008	+52*	13*	29	+65
Plate	SA-533,Gr.B,Cl.1	C9425-1	0.12	0.009	+30*	28*	39	+58
Weld	1NMM/E8018-G	3P4966/1214	0.03	0.010	- 6*	17*	28	+11

NON-BELTLINE

COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA-533,Gr.B,Cl.1	C9401-1	**
Top Head Flange	SA-508,Cl.2	BWB-446	+20
Vessel Flange	SA-508,Cl.2	BRC-424	+26
Feedwater Nozzle	SA-508,Cl.2	Q2Q25W	-6
Weld	1NMW	3P4966/1214	-6
Closure Stud	SA-540, Grade B-24	82552	**

* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

** Not Available.

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INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

ComEd has evaluated the proposed changes to the Technical Specifications (TS) for LaSalle County Station, Units 1 and 2, and has determined that the proposed changes do not involve a significant hazards consideration and provides the following information in support of the NRC determination of no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed; or
>
> Involve a significant reduction in a margin of safety.

We are proposing TS changes for Units 1 and 2, to the pressure-temperature (P-T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the Reactor Pressure Vessel (RPV). The changes replace the current RPV P-T limit TS Figures 3.4.6.1-1 and 3.4.6.1-1a, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure," with three recalculated RPV P-T limit figures that are applicable to 32 Effective Full Power Years (EFPYs). Also the hydrostatic/leak test exception footnote is removed, the 30 minute requirement of TS 4.4.6.1.1 is applied to proposed TS Figure 3.4.6.1-1b, Table 4.4.6.1.3-1, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," is relocated to the LaSalle County Station Updated Final Safety Analysis Report and the description contained in Bases Section B3/4.4.6 is revised. Additionally, for Unit 1 only, the proposed changes decrease the minimum allowable RPV/head flange temperature from 80 to 72 degrees Fahrenheit (F) and also lowers the RPV/head flange surveillance temperatures by 8 degrees.

The information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for these proposed changes is provided below.

Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed changes to the LaSalle County Station reactor pressure vessel (RPV) pressure-temperature (P-T) limits do not modify the boundary, operating pressure, materials or seismic loading of the reactor coolant system. The proposed changes do adjust the P-T limits for radiation effects to ensure that the RPV fracture toughness is consistent with analysis assumptions and NRC regulations. Thus the proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated.

The proposed changes do not adversely affect the integrity of the reactor coolant system such that its function in the control of radiological consequences is affected. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to the reactor pressure vessel pressure-temperature limits do not affect the assumed accident performance of any structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P-T limits. Appendix G of 10 CFR 50 specifies fracture toughness and testing requirements for reactor vessel material in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and that the beltline material in the surveillance capsules be tested in accordance with Appendix H of 10 CFR 50. Appendix G also requires the prediction of the effects of neutron irradiation on the vessel embrittlement. Generic Letter 88-11, "NRC Position on Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations," requests that the methods in Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Material," be used to predict the effect of neutron irradiation on the reactor vessel material.

The current P-T limits for LaSalle County Station were approved by the NRC in Amendment No. 71 for Unit 1 and Amendment No. 55 for Unit 2. The NRC approval of the current pressure-temperature limits was based on their conformance to the requirements of Appendices G and H of 10 CFR 50. The NRC also noted that current P-T limits satisfied Generic Letter 88-11 because the method in Regulatory Guide 1.99, Revision 2 was used to calculate the Adjusted Reference Temperature (ART).

The methodology used to generate the revised P-T limits in the proposed changes is similar to the methodology used to generate the currently approved P-T limits, in conformance with the requirements of Appendices G and H of 10 CFR 50, consistent with the methods of Regulatory Guide 1.99, Revision 2, and consistent with the calculations contained in our July 14, 1999 proposed TS change for power uprate operation. These proposed changes are acceptable because the ASME B&PV Code guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME B&PV Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G was approved in 1974. Thus, the proposed change does not involve a significant reduction in a margin of safety.

Therefore, based upon the above evaluation, we have concluded that these proposed TS changes involve no significant hazards considerations.

ATTACHMENT D
Proposed Technical Specification Changes

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

ComEd has evaluated these proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. We have determined that these proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, have determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that these changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50, that the proposed changes are to a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes are proposed to an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, these proposed changes do not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in Attachment C, there will be no significant increase in the amounts and no significant change in the types of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from these proposed changes.

In accordance with 10 CFR 50.12, "Specific exemptions", Commonwealth Edison (ComEd) Company is requesting an exemption from the requirement 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 B&PV Code Section XI, Division 1" and ASME B&PV Code Section XI Code Case N-588 "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1", in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b.

Justification for Use of Code Case N-640

10 CFR 50.12(a) Requirements

The requested exemption to allow use of ASME B&PV Code Case N-640 in conjunction with ASME B&PV Code XI, Appendix G to determine the pressure-temperature 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 pressure-temperature (P-T) limits being proposed for LaSalle County Station Units 1 and 2 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 B&PV Code, Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME B&PV Code, Section 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 is more technically correct than the K_{Ia} curve. 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. As described in Attachment F, 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 especially in the region of low temperature operations. The two primary safety benefits that would be realized during the pressure test are a reduction in the challenges to operators in maintaining a high temperature in a limited operating window and personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

3. <u>The requested exemption will not endanger the common defense and security</u>:

   The common defense and security are not endangered 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), the 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 cost 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 limit 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 B&PV 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 LaSalle County Station Units 1 and 2, with these P-T curves without the relief provided by ASME B&PV 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. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME B&PV Code Case N-640 in the development of the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME B&PV 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 LaSalle County Station Units 1 and 2 has made a good faith effort to comply with the regulation. We request the exemption be granted until such time that the NRC generically approves ASME B&PV 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 B&PV 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 B&PV 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 B&PV Code Case N-640 for LaSalle County Station Units 1 and 2 will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

Justification for the Use of Code Case N-588

10 CFR 50.12(a) Requirements:

The requested exemption to allow use of ASME B&PV Code Case N-588 to determine stress intensity factors for postulated flaws and postulated flaw orientation for circumferential welds 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 satisfied:

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:

   10 CFR 50, Appendix G, requires that Article G-2120 of ASME B&PV Code, Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels (RPV) for the vessel pressure-temperature limits. These limits are determined for normal operation and pressure/leak test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the RPV material and normal (i.e., perpendicular in the plane of the material) to the direction of maximum stress. ASME B&PV Code, Section XI, Appendix G, also provides methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure-temperature limits.

Code Case N-588 provides benefits, in terms of calculating P-T limits, by revising the Article G-2120 reference flaw orientation for circumferential welds in reactor pressure vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for defects that could physically exist within the geometry of the weldment. The current ASME B&PV Code Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P-T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the reactor pressure vessel wall thickness, which is much longer than the width of circumferential welds. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that defects be postulated in plates/forgings and axial welds. The fabrication of reactor pressure vessels for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties.

These controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during pre-service inspection, in-service non-destructive examinations and data taken from destructive examination of actual reactor pressure vessel welds, confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent non-destructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

ASME B&PV Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME B&PV Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing reactor pressure vessel P-T limits per ASME B&PV Code, Section XI, Appendix G procedures. The procedures allowed by ASME B&PV Code Case N-588 are conservative and provide a margin of safety in the development of reactor pressure vessel pressure-temperature operating and pressure test limits, which will prevent non-ductile fracture of the reactor pressure vessel.

The proposed P-T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in TS Section 3.4.6.K, "Primary System Boundary." Therefore, this requested exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security:

   The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60:

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

   (a)(2)(ii) - demonstrates that the underlying purpose of the regulation will continue to be achieved;

   (a)(2)(iii) - would result in undue hardship or other cost 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):

   The underlying purpose of 10 CFR 50, Appendix G and ASME B&PV Code, Section XI, Appendix G, is to satisfy the underlying requirement that:

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

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

   Application of ASME B&PV Code Case N-588 when determining P-T operating and test limit curves per ASME B&PV Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P-T limits. This application of the code case maintains the margin of safety originally contemplated when ASME B&PV Code, Section XI, Appendix G was developed.

Therefore, use of ASME B&PV Code Case N-588, as described above, satisfies the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable level of safety.

<u>10 CFR 50.12(a)(2)(iii)</u>:

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 with these P-T limit curves without the relief provided by ASME B&PV Code Case N-588 would unnecessarily restrict the pressure-temperature operating window for LaSalle County Station Units 1 and 2. This restriction challenges the operations staff during pressure tests to maintain a high temperature within a limited operating window. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME B&PV Code Case N-588 in the development the proposed P-T curves. Implementation of the proposed P-T limit curves as allowed by ASME B&PV Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

<u>10CFR50.12(a)(2)(v)</u>:

The exemption provides only temporary relief from the applicable regulation and LaSalle County Station has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME B&PV Code Case N-588 for use by the nuclear industry.

ASME B&PV Code Case N-588, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME B&PV Code, Section XI, Appendix G was developed and imposes restrictions on P-T operating limits beyond those originally contemplated.

This proposed alternative is acceptable because the 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 B&PV Code Case N-588 for LaSalle County Station will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor pressure vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in TS Section 3/4.4.6. Therefore, this exemption does not present an undue risk to the public health and safety.

Abstract

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature limits, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate safety margins for nine different parameters; one of which is a lower bound fracture toughness curve.

There are two lower bound fracture toughness curves available in Section XI, K_{Ia}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{Ic}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{Ia} to K_{Ic}. The other margins involved with the process remain unchanged.

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By opening up the operating window relative to the pump seal requirements, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

Technology developed over the last 25 years has provided a strong basis for revising the ASME B&PV Code Section XI pressure-temperature limit curve methodology. The safety margin which exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease.

Introduction

The startup and shutdown process, as well as pressure testing, for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate four specific safety margins:

1. Large flaw, ¼ thickness
2. Safety factor = 2 on pressure stress for startup and shutdown
3. Lower bound fracture toughness
4. Upper bound adjusted reference temperature (RT_{NDT})

Although the above four safety margins were originally included in the methodology used to develop P-T Limit Curves and hydrotest temperatures, it is important to mention that several sources of stress were not considered in the original methodology. The two key factors here are the weld residual stresses, and stresses which result from the clad-base metal differential thermal expansion. Furthermore, the method as originally proposed assumed that the maximum value of the stress intensity factor occurred at the deepest point of the flaw. These elements were all considered in the sample problems which were carried out, so their effects on the margins could be assessed.

There are two lower bound fracture toughness curves available in Section XI, K_{Ia}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{Ic}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{Ia} to K_{Ic}. The other margins involved with the process remain unchanged. There are a number of reasons why the limiting toughness in the Appendix G pressure-temperature limits should be changed from K_{Ia} to K_{Ic}.

Use of K_{Ic} is More Technically Correct

The heatup and cooldown process is a very slow one, with the fastest rate allowed being 100 degrees F per hour. The rate of change of pressure and temperature is often constant, so the rate of change in stress is essentially constant. Both the slow heatup and cooldown and the pressure testing are essentially static processes. In fact, all operating transients (levels A, B, C and D) correspond to static loadings, with regard to fracture toughness.

The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{Ia} should be used for the reactor pressure vessel is when a crack is running. This might happen during a PTS transient event, but not during heatup or cooldown. Therefore, use of the static toughness K_{Ic} lower bound toughness would be more technically correct for development of P-T limit curves.

Use of Historically Large Margin No Longer Necessary

In 1974, when the Appendix G methodology was first codified, the use of K_{Ia} (K_{Ir} in the terminology of the time) to provide additional margin was thought to be necessary to cover uncertainties and a number of postulated but unquantified effects. Almost 25 years later, significantly more is known about these uncertainties and effects.

Flaw Size

With regard to flaw indications in reactor vessels, there have been no indications found at the inside surface of any operating reactor in the core region which exceed the acceptance standards of Section XI, in the entire 28 year history of Section XI. This is a particularly impressive conclusion when considering that core region inspections have been required to concentrate on the inner surface and near inner surface region since the implementation of Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations." Flaws have been found, but all have been qualified as buried, or embedded.

There are a number of reasons why no surface flaws exist, and these are related to the fabrication and inspection practices for vessels. For the base metal and full penetration welds, a full volumetric examination and surface exam is required before cladding is applied, and these exams are repeated after cladding.

Further confirmation of the lack of any surface indications has recently been obtained by the destructive examination of portions of several commercial reactor vessels, for example the Midland vessel and the PVRUF vessel.

Fracture Toughness

Since the original formulation of the K_{Ia} and K_{Ic} curves, in 1972, the fracture toughness database has increased by more than an order of magnitude, and both K_{Ia} and K_{Ic} remain lower bound curves, as shown for example in Figure 1 for K_{Ic} [1] compared to Figure 2, which is the original database [2]. In addition, the temperature range over which the data have been obtained has been extended, to both higher and lower temperatures than the original data base.

In can be seen from Figure 1 that there are a few data points which fall just below the curve. Consideration of these points, as well as the (over 1500) points above the curve, leads to the conclusion that the K_{Ic} curve is a lower bound for a large percentage of the data. An example set of carefully screened data in the extreme range of lower temperatures is shown in Figure 3, from Reference [3].

Local Brittle Zones

A third argument for the use of K_{Ia} in the original version of Appendix G was based upon the concern that there could be a small, local brittle zone in the weld or heat-affected-zone of the base material that could pop-in and produce a dynamically moving cleavage crack. Therefore, the toughness property used to assess the moving crack should be related to dynamic or crack arrest conditions, especially for a ferritic pressure vessel steel showing distinct temperature and loading-rate (strain-rate) dependence. The dynamic crack should arrest at a ¼-T size, and any re-initiation should consider the effects of a minimum toughness associated with dynamic loading. This argument provided a rationale for assuming a ¼-T postulated flaw size and a lower bound fracture toughness curve considering dynamic and crack arrest loading. The K_{Ir} curve in Appendix G of Section III, and the equivalent K_{Ia} curve in Appendix A and Appendix G of Section XI provide this lower bound curve for high-rate loading (above any realistic rates in reactor pressure vessels during any accident condition) and crack arrest conditions. This argument, of course, relies upon the existence of a local brittle zone.

After over 30 years of research on reactor pressure vessel steels fabricated under tight controls, micro-cleavage pop-in has not been found to be significant. This means that researchers have not produced catastrophic failure of a vessel, component, or even a fracture toughness test specimen in the transition temperature regime. The quality of quenched, tempered, and stress-relieved nuclear reactor pressure vessel steels, that typically have a lower bainitic microstructure, is such that there may not be any local brittle zones that can be identified. Testing of some test specimens at ORNL [4] has shown some evidence of early pop-ins for some simulated production weld metals, but the level of fracture toughness for these possible early initiations is within the data scatter for other ASTM-defined fracture toughness values (K_{Ia} and/or K_{Ic}). Therefore, it is time to remove the conservatism associated with this postulated condition and use the ASME B&PV Code lower bound K_{IC} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

<u>Overall Plant Safety is Improved</u>

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument can be made that the proposed change will increase plant safety and reduce personnel exposure for both PWRs and BWRs.

<u>Impact on PWRs:</u>

By opening up the operating window relative to the pump seal requirements, as shown schematically in Figure 4, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

The proposed change also reduces the need for lock-out of the HPSI systems, which improves personnel and plant safety and reduces the potential for a radioactive release. Finally, challenges to the plant low temperature overpressure protection system (LTOP) and potential problems with reseating the valves would also be reduced.

Impact on BWRs:

The primary impact on the BWR will be a reduction in the pressure test temperature. BWRs use pump heat to reach the required pressure test temperatures. Several BWR plants are required to perform the pressure test at temperatures over 212°F under the current Appendix G criteria. The high test temperature poses several concerns: (i) pump cavitation and seal degradation, (ii) primary containment isolation is required and ECCS/safety systems have to be operational at temperatures in excess of 212°F, (iii) leak detection is difficult and more dangerous since the resulting leakage is steam and poses safety hazards of burns and exposure to personnel. The reduced test temperature eliminates these safety issues without reducing overall fracture margin.

Reactor Vessel Fracture Margins

It has long been known that the P-T limit curve methodology is very conservative[5,6]. Changing the reference toughness to K_{Ic} will maintain a very high margin, as illustrated in Figure 5, for a pressurized water reactor. Similar results are shown for a BWR hydrotest in Figure 6. These figures show a series of P-T curves developed for the same plant (either a BWR or a PWR), but with different assumptions concerning flaw size, safety margin and fracture toughness.

Results were obtained for a sample problem which was solved by several members of the Section XI working group on Operating Plant Criteria, for both PWR and BWR plants. The problem statement details are provided in Appendix A (separate problems for the PWR and BWR). The sample problem requires development of an operating P-T cooldown curve or the pressure test for an irradiated vessel. Two P-T curves were required, one using K_{Ia} and the second using K_{Ic}. In both cases the quarter thickness flaw was used, along with the appropriate safety factor on pressure.

To determine the margins (pressure ratios) that are included in these curves, a reference P-T curve was developed, using a best estimate (mean) K_{Ic} curve, and no safety factor on stress, along with a flaw depth of one inch. These analyses all considered the K_I/K_{Ic} ratio at all points on the crack front located in the ferritic steel. Typical results are shown in Table 1 for a PWR. Comparing the reference or best estimate curve with the two P-T curves calculated using code requirements, we see that there is a large margin on the allowable pressure, whether one uses K_{Ia} or K_{Ic} limits in Appendix G.

For PWRs, another important contribution to the margin, which cannot be quantified, is the low temperature overpressure protection system (LTOP) which is operational in the low temperature range. The margins increase significantly for higher temperatures, as seen in Figure 5.

Impact of the Change on P-T Curves

To show the effect that the proposed change would produce, a series of P-T limit curves were produced for a typical plant. These curves were produced using identical input information, with one curve using K_{Ia} and the other using the proposed new approach, with K_{Ic}. Since the limiting conditions for the PWR (cooldown) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs.

The results are shown in Figure 7 for a typical PWR cool-down transient.

Summary and Conclusions

Technology developed over the last 25 years has provided a strong basis for revising the ASME B&PV Code Section XI pressure-temperature limit curve methodology. The safety margin that exists with the revised methodology is still very large.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures, need for HPSI systems lock-out, LTOP system challenges and/or fuel margin problems, and personnel hazards and exposure will all decrease.

References

1. VanderSluys, W.A. and Yoon, K.K., "Transition Temperature Range Fracture Toughness in Ferritic Steels and Reference Temperature of ASTM", prepared for PVRC and BWOG, BAW 2318, Framatome Technologies, April 1998.

2. Marston, T.U., "Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A", EPRI Special Report NP-719-SR, August 1978.

3. Nanstad, R.K. and Keeney, J.A., and McCabe, D.E., "Preliminary Review of the Bases for the K_{Ic} Curve in the ASME Code", Oak Ridge National Laboratory Report ORNL/NRC/LTR-93/15, July 12, 1993.

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5. Chirigos, J.N. and Meyer, T.A., "Influence of Material Property Variations on the Assessment of Structural Integrity of Nuclear Components", ASTM Journal of Testing and Evaluation, Vol. 6, No. 5, Sept. 1978, pp 289-295.

6. White Paper on Reactor Vessel Integrity Requirements for Level A and B conditions, prepared by Section XI Task Group on R.V. Integrity Requirements, EPRI TR-100251, January 1993.

Table 1
Summary of Allowable Pressures for
20 Degree/hour Cooldown of Axial Flaw at
70 Degrees F and RT_{PTS} of 270 F
(Typical PWR Plant)

Type of Evaluation	Allowable Pressure* (psi)	Pressure Ratio
Appendix G with t/4 flaw and K_{Ia} Limit	420	1.00
Appendix G with t/4 flaw and K_{Ic} Limit	530	1.26
Reference Case: 1 inch flaw For pressure, thermal, Residual and cladding loads	1520	3.61
Reference Case: 1 inch flaw for pressure, thermal and residual loads	1845	4.38
Reference Case: 1 inch flaw For pressure and thermal Loading only	2305	5.48

* Note: Comparable values of allowable pressure were calculated by the ASME Section XI Operating Plant Working Group Members from Westinghouse, Framatome Technologies and Oak Ridge National Laboratory

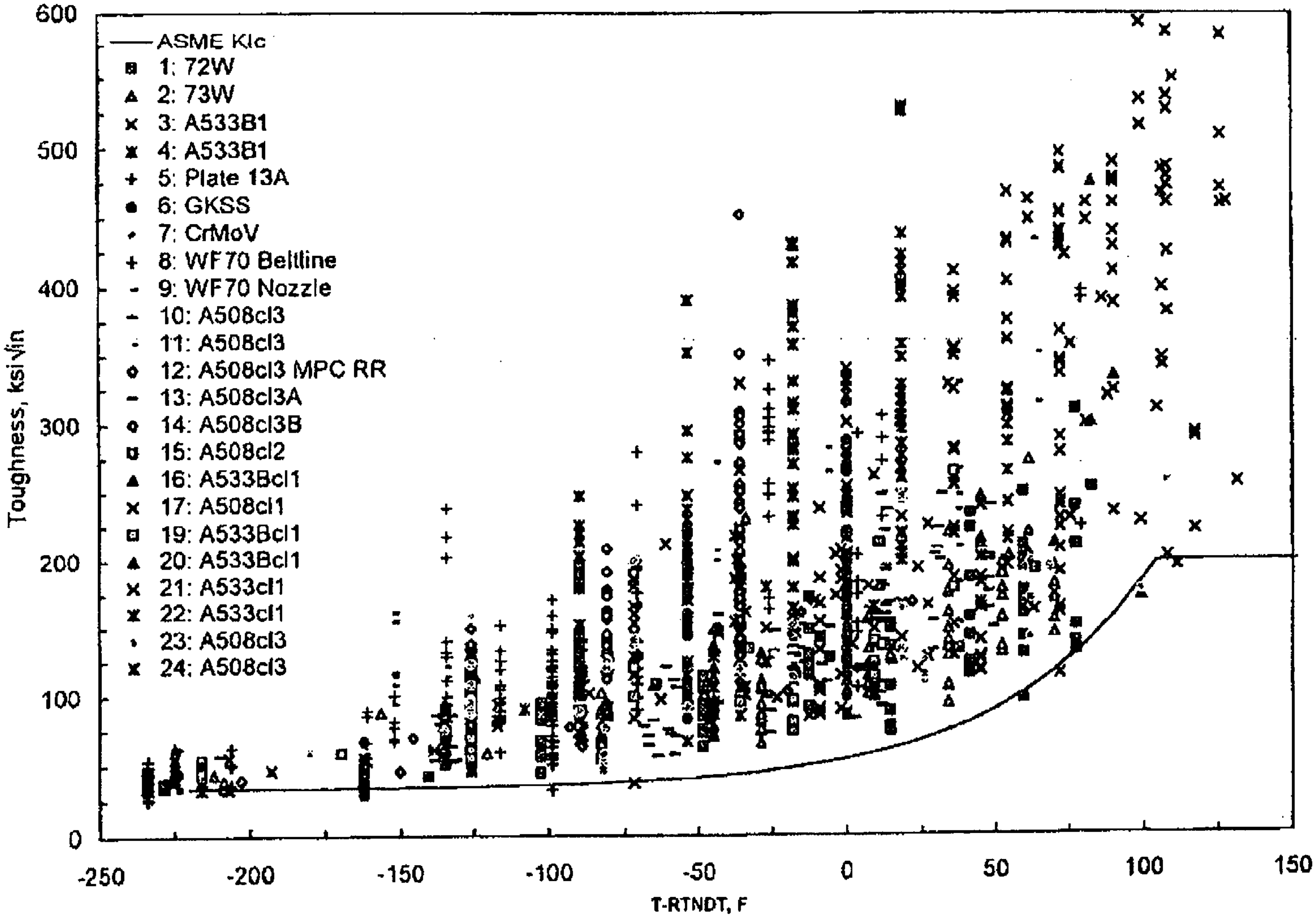

Figure 1. Static Fracture Toughness Data (K_{JC}) Now Available, Compared to K_{IC} [1]

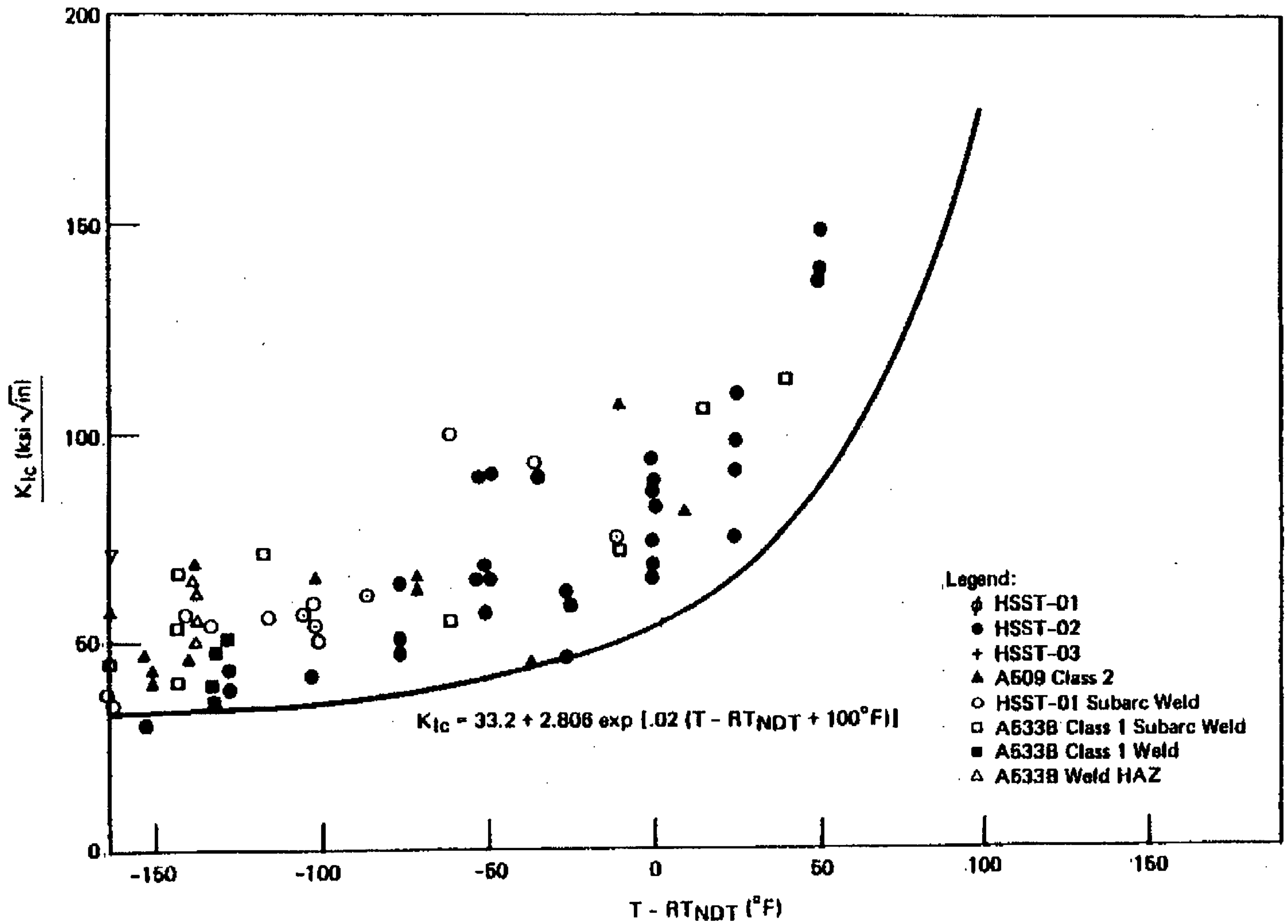

Figure 2. Original K_{IC} Reference Toughness Curve, with Supporting Data [2]

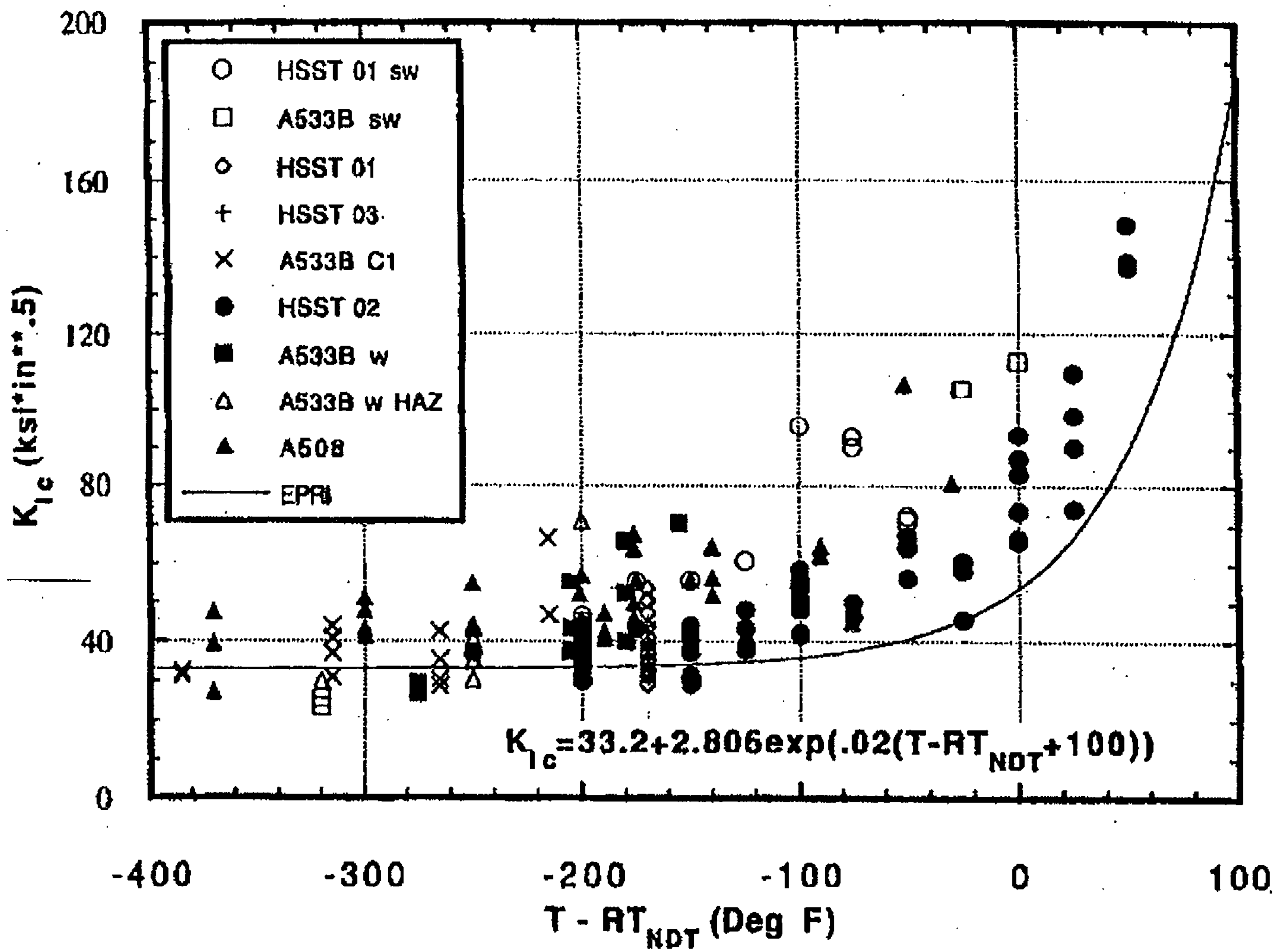

Figure 3. K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

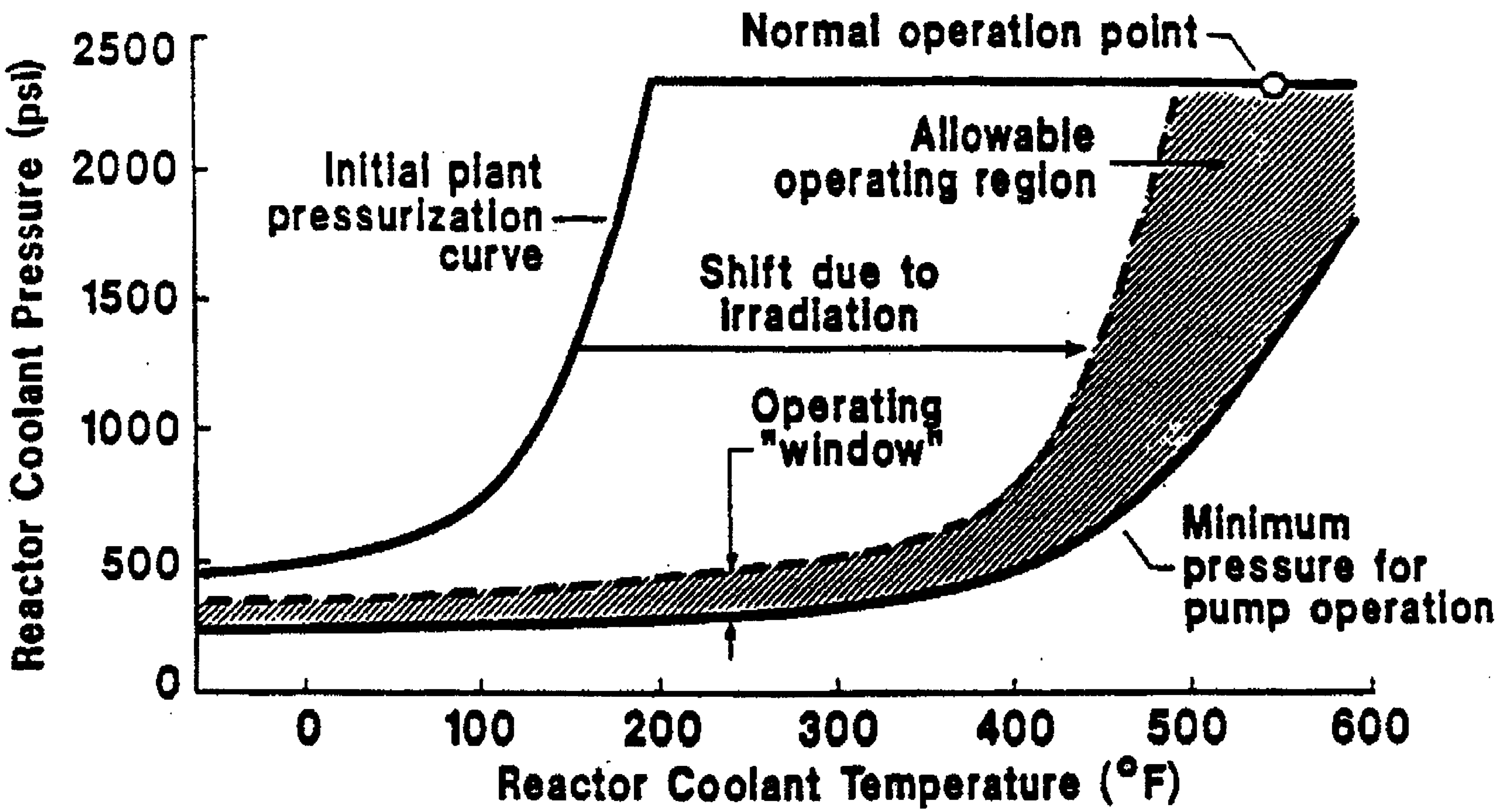

Figure 4. Operating Window From P-T Limit Curves [4]

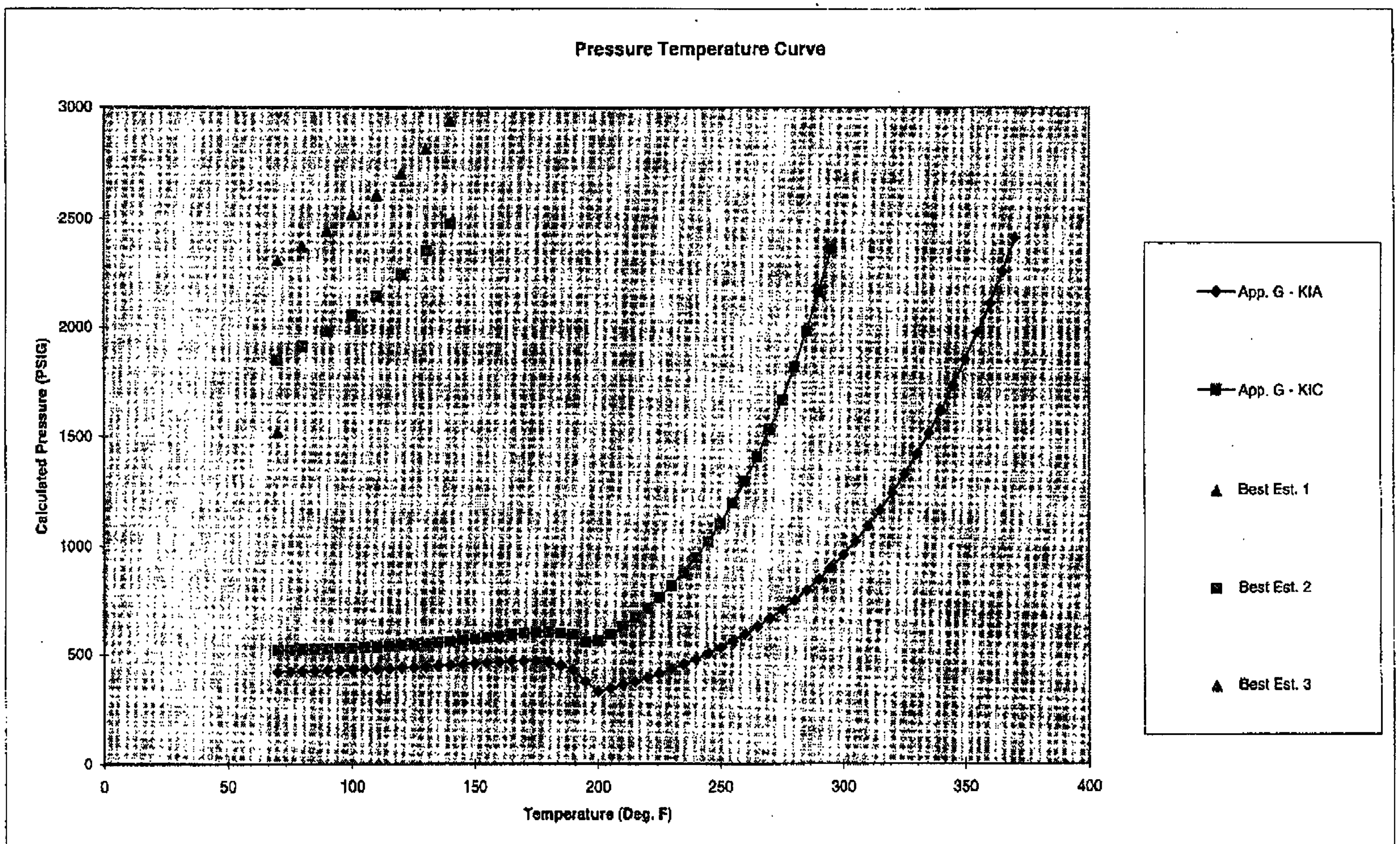

Figure 5. P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

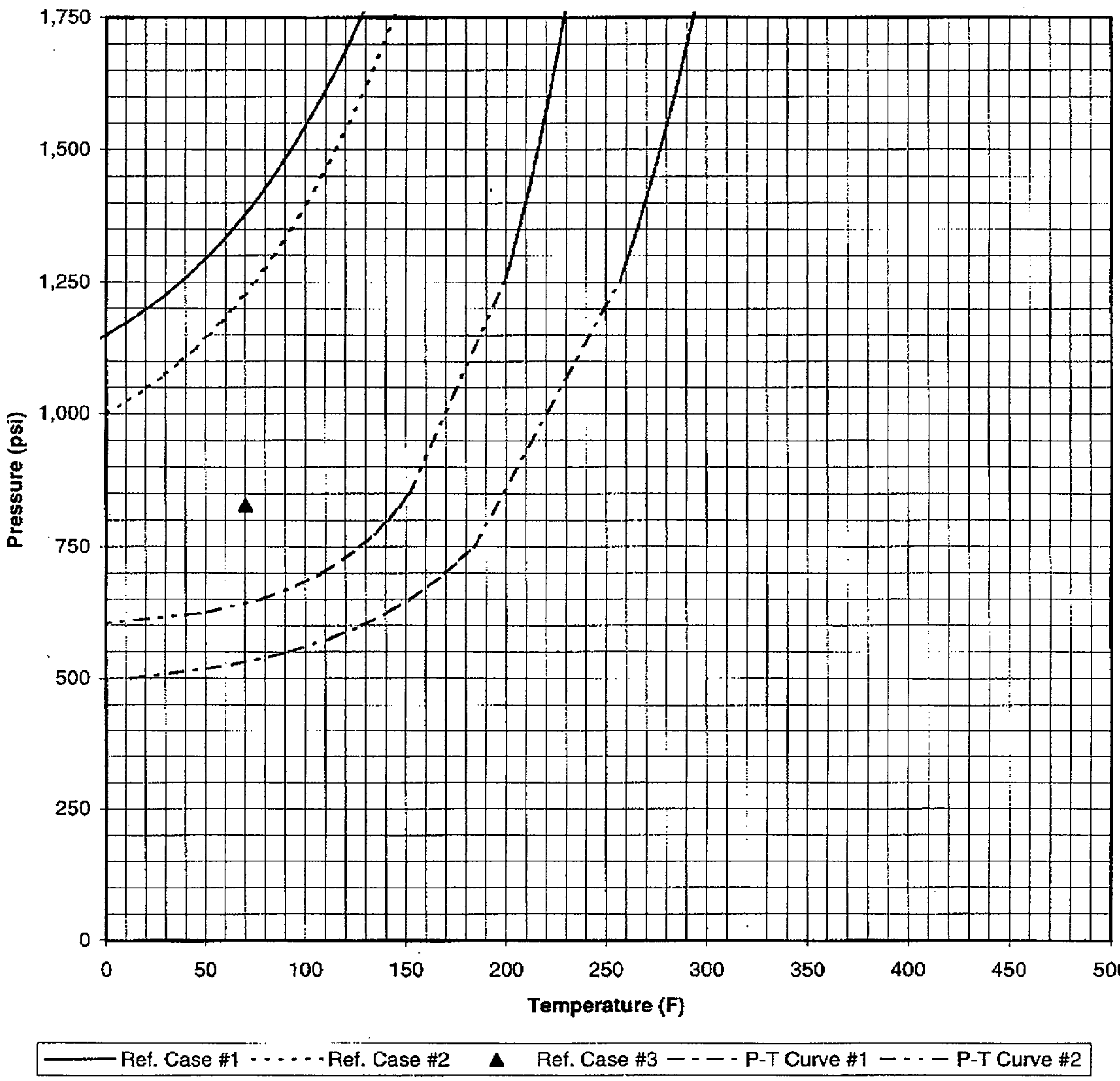

Figure 6. P-T Limit Curves Illustrating Deterministic Safety Factors for a BWR Reactor Vessel

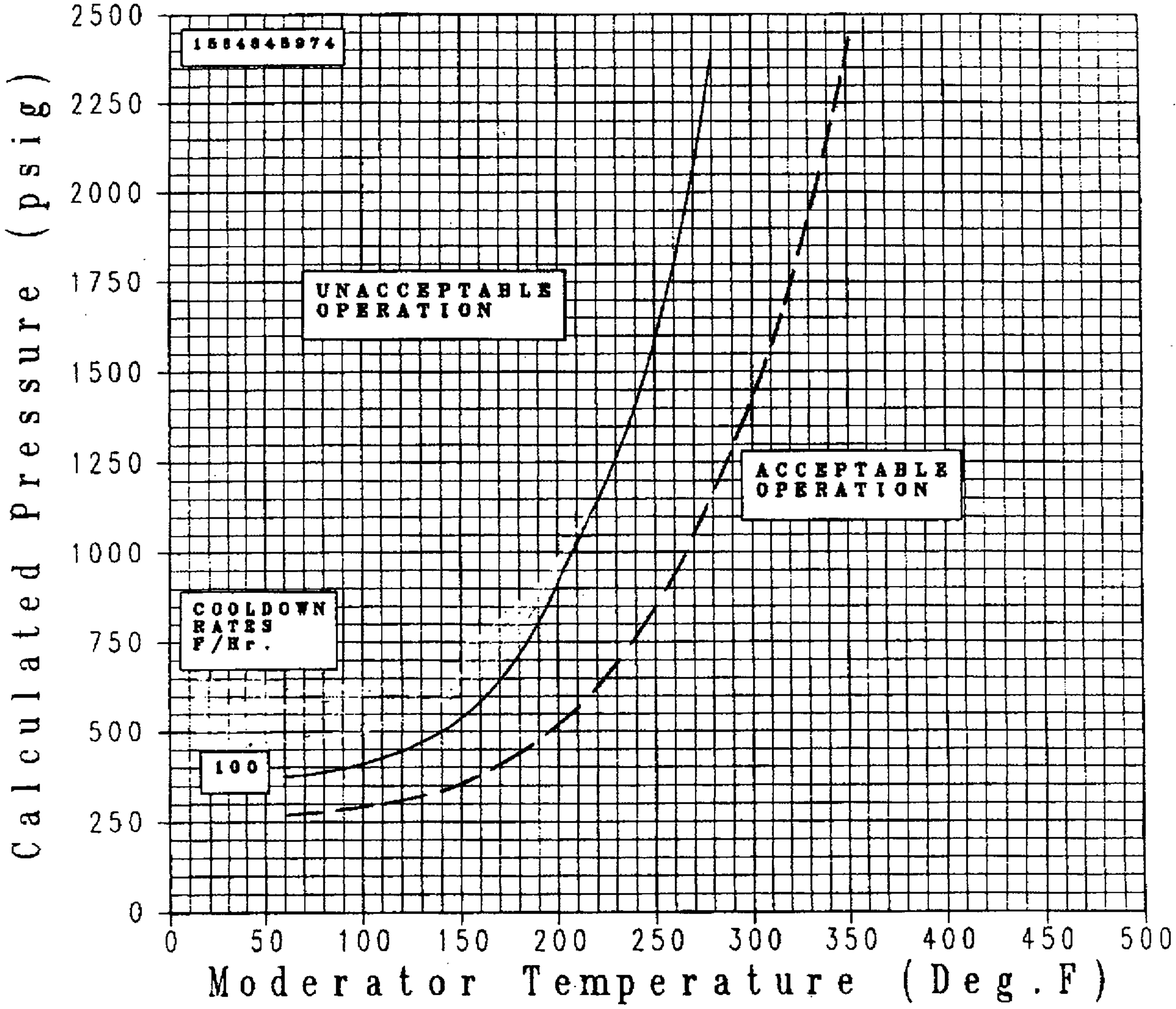

Figure 7. Comparison of Cool-Down Curves for the Existing and Proposed Methods - PWR [Dashed Curve = Existing (K_{Ia}) and Solid Curve = Proposed (K_{Ic})]

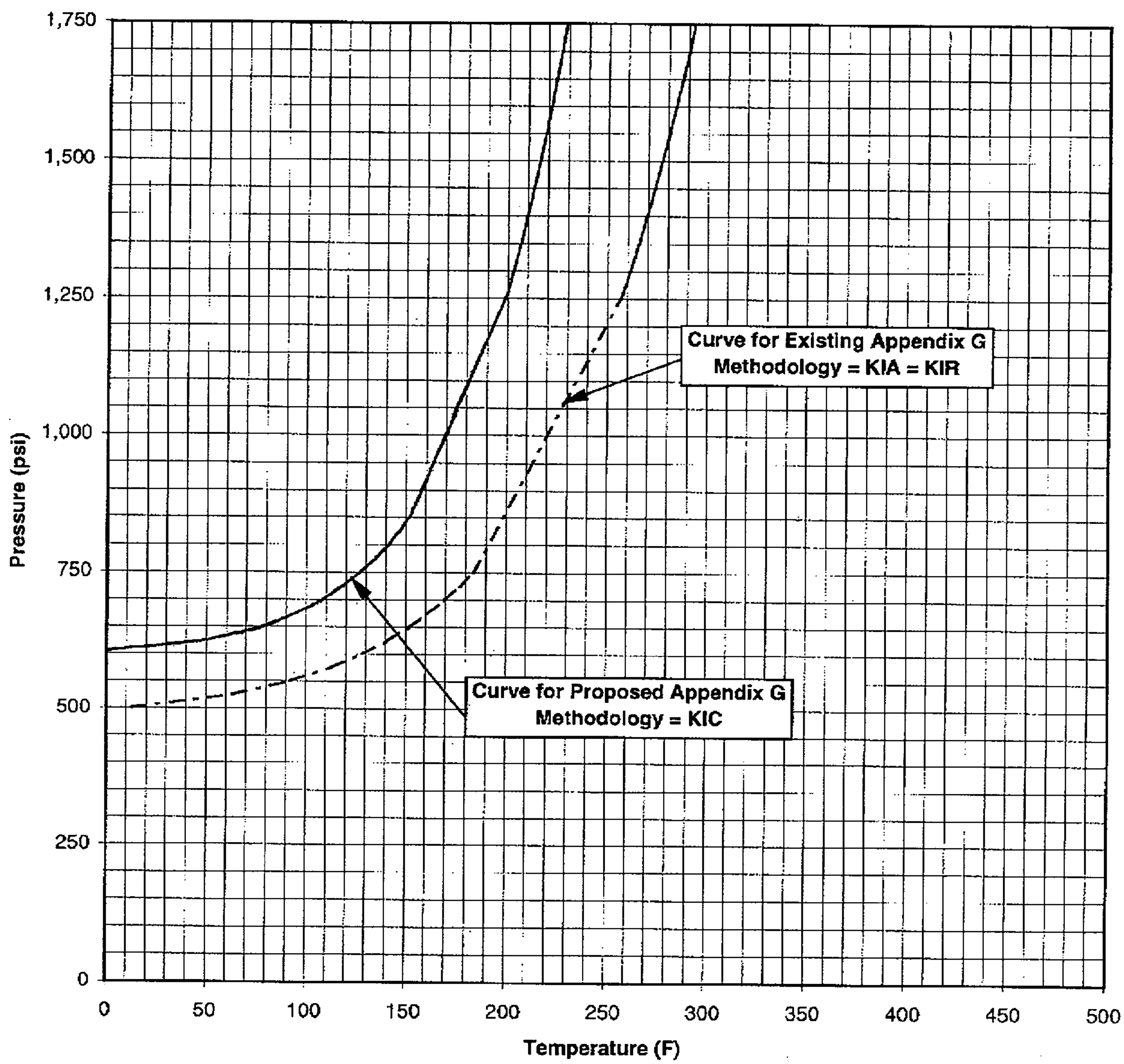

Figure 8. Comparison of Hydrotest P-T Curves for the Existing and Proposed Methods - BWR [Dashed Curve=Existing (K_{Ia}) and Solid Curve=Proposed (K_{Ic})]

Appendix A

Section XI P-T Limit Curve Sample Problems

Introduction

This series of sample problems was developed to allow comparison calculations to be carried out to support the proposed change from K-IA to K-IC in Appendix G of Section XI. These problems were developed in a meeting held on July 7, 1998, between the NRC staff, Westinghouse, ORNL, and Framatome Technologies. Later, a variation on the sample problems was developed for application to BWRs.

The sample problems involve a tightly specified reference case, with two variations, and then two P-T Limit curve calculations whose input is also tightly specified, one using K-IA and the second using K-IC. The goal of the problems is to determine the margin on pressure which exists using the K-IA approach, and the margin which exists with the proposed K-IC approach.

The problem input variables are contained in the attached tables. The problem statement is given below. As will be seen there are two problem types, the first being a best-estimate, or reference problem, and the second being standard P-T limit curves determined using code-type assumptions, with safety factors.

Reference Cases (Best Estimate)

Determine a best estimate P-T Cooldown Curve for a typical reactor vessel, over the entire temperature range of operation, starting at 70F. For BWR plants, also calculate a hydrotest pressure versus temperature curve. The problem input is defined in Table 1. This problem is meant to be a best estimate curve with no specific safety factors, and best estimate values for each of the variables. Only pressure and thermal stresses are used for case R1. Although these stresses are the only ones presently considered in P-T limit curve calculations, other stresses can exist in the vessel, and two other cases were constructed to obtain additional information on these issues. These other two cases treat stresses which are at issue regardless of which toughness is used for the calculations, but are provided for information.

Reference case R2. This case is similar to case R1, but the weld residual stresses are added for a longitudinal weld in the reactor vessel.

Reference case R3. This case is similar to case R2, but now the clad residual stresses are added. Since the clad residual stresses are negligible at higher temperatures, this calculation is only performed at room temperature, or 70F.

The stress intensity factor results for the reference cases may not always result in the maximum value at the deepest point of the flaw, so care should be taken to check this. If the maximum value is not at the deepest point, the calculated ratio of K / K-IC should be calculated around the periphery, and reported. The resulting allowable pressure would then be determined from the governing result at each time step. The calculation method could use either Section XI Appendix A, or the ORNL method, as documented in Table A-1.

P-T Curve Cases

Case 1 is a classic P-T Curve calculation done according to the existing rules in Section XI Appendix G, using the K-IA curve and the code specified safety factors. The input values are provided in Table A-2, for both PWR and BWR plants.

Case 2 is the same as case 1, except that the fracture toughness curve K-IC is used. This is the proposed Code change.

In each case a full P-T limit curve should be calculated, but there is no need to calculate leak test temperature, bolt-up temperature, or any other parameters. For BWR plants, a hydrotest pressure versus temperature curve is also required.

TABLE A-1: REFERENCE CASE VARIABLES

Reference Case 1

Vessel Geometry:	Thickness = 9.0 inch (PWR) or 6.0 inches (BWR) Inside Radius = 90 inch (PWR) or 125 inches (BWR) Clad Thickness = 0.25 inch
Flaw:	Semi-elliptic Surface Flaw, Longitudinal Orientation Depth = 1.0 inch Length = 6 x Depth
Toughness:	Mean K_{IC} , from report ORNL/NRC/LTR/93-15, July 12, 1993 $K_{IC} = 36.36 + 51.59 \exp [0.0115 (T-RT_{NDT})]$
Loading:	100F/Hr cooldown from 550F to 200F 20F/Hr cooldown from 200F to 70F
Film Coefficient:	h = 1000B/hr-ft-F
Stress Intensity Factor Expression:	Section XI, Appendix A, or ORNL Influence Coefficients, from ORNL/NRC/LTR-93-33 Rev. 1, Sept. 30, 1995
Irradiation Effects:	RT_{NDT} = 236°F(PWR) or 168°F (BWR) @ inside surface = 220°F(PWR) @ depth = 1.0 in. = 200°F(PWR) @ depth = T/4 = 133°F(PWR) @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of coolant temperature and for BWR plants, calculate hydrotest pressure as a function of coolant temperature.

Reference Case 2

Same as Reference Case 2, but for the loadings, add a weld residual stress distribution.

	Location (a/t)	Stress(ksi)	Location (a/t)	Stress(ksi)
Inner Surface	0.000	6.50	0.045	5.47
	0.067	4.87	0.101	3.95
	0.134	2.88	0.168	1.64
	0.226	-0.79	0.285	-3.06
	0.343	-4.35	0.402	-4.31
	0.460	-3.51	0.510	-2.57
	0.572	-1.70	0.619	-1.05
	0.667	-0.46	0.739	0.35
	0.786	0.87	0.834	1.41
	0.881	1.96	0.929	2.55
	0.976	3.20	1.000	3.54

Reference Case 3

Same as Reference Case 2, but add clad residual stress distribution, and calculate allowable pressure only at 70°F.

For the clad residual stress distribution, choose either distribution 1 or distribution 2, from the attached figures. Figure A-1 was calculated from the ORNL Favor Code, and Figure A-2 was taken from a technical paper which presents results of residual stresses measured on nozzle drop-out materials.

TABLE A-2: P-T Calculation Cases

Calculation Case 1

Vessel Geometry: Thickness - 9.0 inch (PWR), 6.0 inches (BWR)
Inside Radius = 90 inch (PWR), 125 inches (BWR)
Clad Thickness = 0.25 inch

Flaw: Semi-elleptic Surface Flaw, Longitudinal Orientation
Depth = 1.0 inch
Length = 6 x Depth

Toughness: K_{Ia}

Loading: 100F/hr cooldown, 550 to 200 F
20F/hr cooldown, 200 to 70F

Stress Intensity Factor Expression: Latest Section XI App G expression (from ORNL/NRC/LTR-93-33, Rev. 1)

Irradiation Effects: ART = 236F(PWR) or 168°F(BWR) @ inside surface
= 220F(PWR) @ depth = 1.0 inch
= 200F(PWR) @ depth = T/4
= 133F(PWR) @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of temperature, and for BWRs calculate hydrotest pressure as a function of temperature.

Calculation Case 2

Same parameters as Case 1, but Toughness = K_{IC}

From ORNL Favor Code, per Terry Dickson, 7/9/98

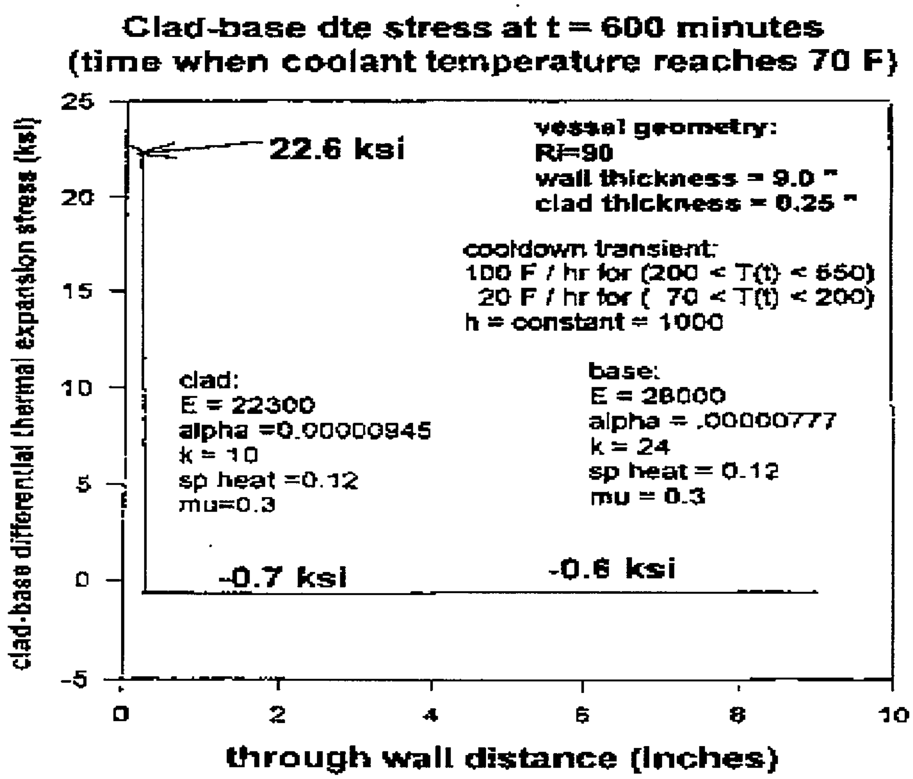

Figure A-1: Clad-base dte stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

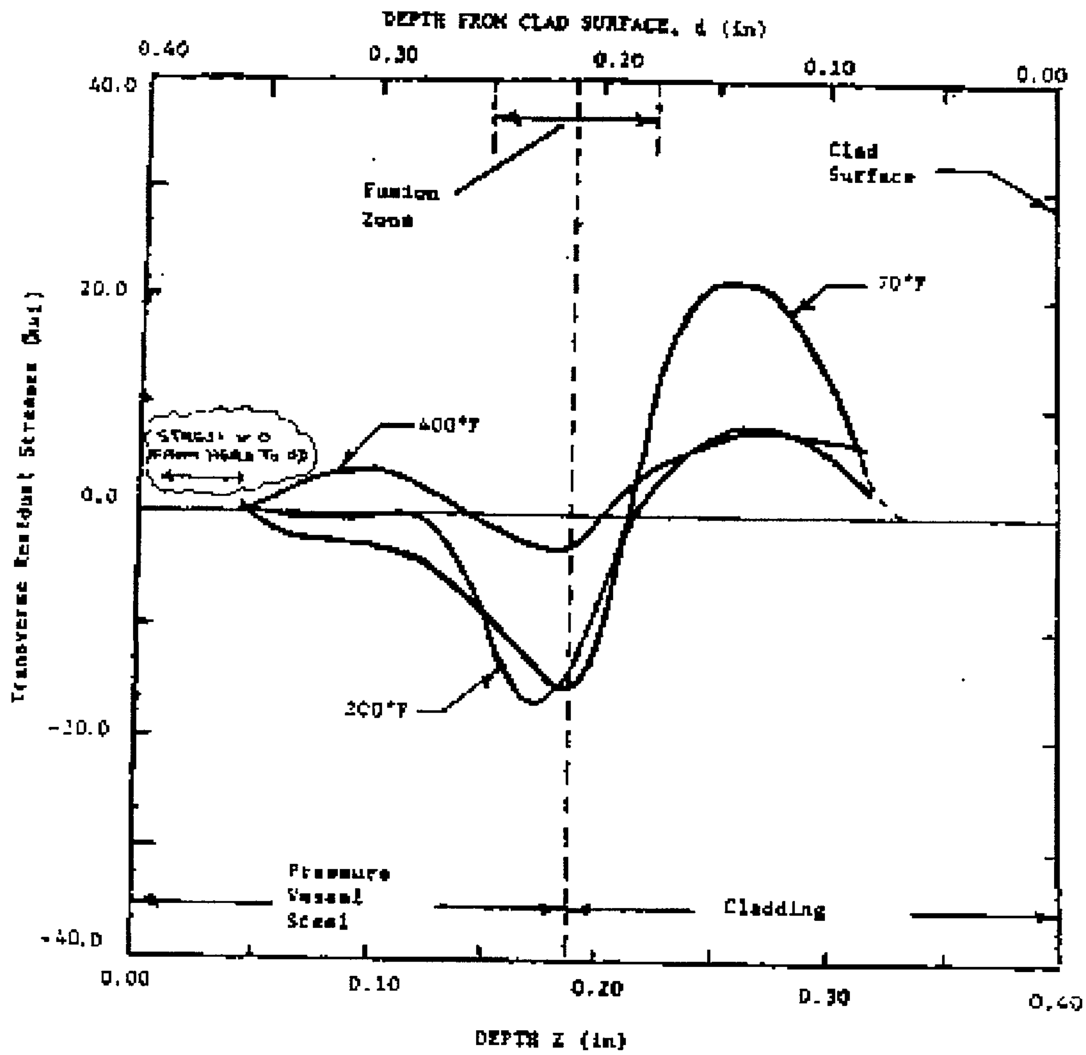

Figure A-2: Residual Stresses Transverse to Direction of Welding