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10CFR 50.90

ComEd

February 23, 2000

PSLTR-00-0057

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Request for an Amendment to Technical Specifications Section 3 /4.6.K, "Primary System Boundary" and Section 3 /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."

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

A001

In accordance with 10 CFR 50.90, we request a change to Technical Specifications (TS) of Facility Operating License Nos. DPR-19 and DPR-25, for the Dresden Nuclear Power Station, Units 2 and 3, respectively. The proposed change is to TS Section 3/4.6.K, "Primary System Boundary" and Section 3/4.12.C, "Special Test Exceptions." In support of this TS change request, we are also requesting exemption from 10CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," in accordance with 10CFR 50.12, "Specific exemptions."

The proposed change modifies the Pressure-Temperature (P-T) limits by revising the heatup, cooldown and inservice test limitations for the reactor pressure vessel (RPV) of each unit to a maximum of 32 Effective Full Power Years (EFPYs). The use of 32 EFPYs conservatively bounds both Units 2 and 3 that are currently at approximately 17 EFPYs. Furthermore, the proposed change deletes the Special Test Exception, which provides for pressure testing at greater than 212 degrees F in OPERATIONAL MODE 4, "COLD SHUTDOWN." This proposed change offers potential radiation dose savings by increasing the effectiveness of inspectors in the containment at lower ambient temperature; potential outage schedule savings; and a reduction of burden on operators by eliminating the requirement to maintain the RPV within a narrow temperature band above 212 degrees F during pressure testing.

This proposed change stems from recently approved change in American Society of Mechanical Engineers (ASME) methodology for determining allowable P-T limits. The basic methodology utilized for the generation of the proposed P-T curves is similar to that utilized for the current P-T limits which were approved in 1997. Several improvements to the methodology were made, including the incorporation of 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 Reference Fracture Toughness for Development of P-T Curves for ASME Section XI, Division 1."

This TS change request is supported by a request for an exemption in accordance with 10 CFR 50.12 from certain requirements of 10 CFR 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," to meet 10CFR 50, Appendix G, "Fracture Toughness Requirements." The requested exemption from 10 CFR 50.60(a) is to allow use of ASME Code Cases N-588 and N-640 as described below.

- ASME Code Case N-588 allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. The ASME code case was approved for use by the appropriate ASME B&PV Section XI committee on December 12, 1997.

- ASME Code Case N-640 provides an alternate method for determining the fracture toughness of reactor vessel materials for use in determining P-T Limits. The ASME code case was approved for use by the appropriate ASME B&PV Section XI committee on February 26, 1999.

Similar license amendment and exemption requests were submitted by Quad Cities Nuclear Power Station in Reference 1 and granted by the NRC in References 2 and 3. The revised P-T limits for the Dresden Nuclear Power Station reactor vessels were calculated using the same analytical methods that were used to develop the revised reactor vessel P-T limits for Quad Cities Nuclear Power Station. The material properties used with this methodology were specifically from the materials used to fabricate each vessel. These vessel material properties were the same as those used to generate the current vessel P-T limits, except for the beltline materials. The beltline material properties are consistent with our most recent responses to Generic Letter 92-01, "Reactor Vessel Structural Integrity," in Reference 4. The normal operating and anticipated transients used to define the vessel stresses were identical for both stations.

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

This request is subdivided as follows.

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

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

3. Attachment C describes our evaluation performed using the criteria in 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.

5. Attachment E gives justification for Exemption Request.

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

7. Attachment G provides GE Nuclear Energy Reports GE-NE-B13-02057-00-03 "Pressure-Temperature Curves for ComEd Dresden Unit 3," and GE-NE-B13-02057-00-04 "Pressure-Temperature Curves for ComEd Dresden Unit 2."

These proposed changes have been reviewed by the Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

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

Should you have any questions concerning this letter, please contact Mr. D. F. Ambler (815) 942-2920, extension 3800.

Respectfully,

Preston Swafford

Preston Swafford
Site Vice President
Dresden Nuclear Power Station

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B: Marked-Up Pages for Proposed Changes
Attachment C: Information Supporting No Significant Hazard Findings
Attachment D: Information Supporting An Environmental Assessment
Attachment E: Justification for Exemption Request
Attachment F: Technical Basis for Revised P-T Limit Core Methodology
Attachment G: GE-NE-B13-02057-00-03 "Pressure-Temperature Curves for ComEd Dresden Unit 3," and GE-NE-B13-02057-00-04 "Pressure-Temperature Curves for ComEd Dresden Unit 2."

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

bcc: Project Manager – NRR
Office of Nuclear Facility Safety – IDNS
Senior Reactor Analyst - IDNS
Manager of Energy Practice – Winston and Strawn
Director, Licensing and Compliance – Dresden Nuclear Power Station
Vice President, Regulatory Services– ComEd
ComEd Document Control Desk Licensing (Hard Copy)
ComEd Document Control Desk Licensing (Electronic Copy)
Regulatory Assurance Manager – Dresden Nuclear Power Station
Regulatory Assurance Manager – Quad Cities Nuclear Power Station
NRC Coordinator – Dresden Nuclear Power Station
NSRB Site Coordinator – Dresden Nuclear Power Station
Brendan Casey – Dresden Nuclear Power Station
ITS Project Lead - Dresden Nuclear Power Station
Dresden Regulatory Assurance, Subject File
SVP Numerical File – PSLTR #00-0057

STATE OF ILLINOIS	)	
COUNTY OF GRUNDY	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Numbers**
DRESDEN NUCLEAR POWER STATION UNITS 2 and 3	)	**50-237 and 50-249**

SUBJECT: Request for an Amendment to Technical Specifications Section 3 /4.6.K, "Primary System Boundary" and Section 3 /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."

AFFIDAVIT

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

Preston Swafford

Preston Swafford
Site Vice President

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

for the State above named, this 23rd day of

February, 20 00.

OFFICIAL SEAL
LINDA S. SANDERS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 12-29-2002

Linda S. Sanders
Notary Public

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is proposing changes to the Technical Specifications (TS) of Facility Operating Licenses DPR-19 and DPR-25, for Dresden Nuclear Power Station, Units 2 and 3, respectively. The proposed changes are to TS Section 3/4.6.K, "Primary System Boundary" and Section 3 /4.12.C, "Special Test Exceptions."

The proposed changes revise the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel (RPV) of each unit to a maximum of 32 Effective Full Power Years (EFPYs). The proposed changes also delete the TS Special Test Exception, which provides for pressure testing at greater than 212 degrees F in OPERATIONAL MODE 4, "COLD SHUTDOWN."

The proposed changes are described in detail in Section E of this Attachment. The marked up TS pages are shown in Attachment B. Also, marked up bases pages are provided for completeness.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS Limiting Conditions for Operation (LCO) and Surveillance Requirements (SRs) provide for the primary coolant system temperature and RPV metal temperature and pressure to be limited and monitored within the acceptable regions as shown on TS Figures 3.6.K-1 through 3.6.K-5.

TS Special Test Exception 3/4.12.C provides for operation in Mode 4 when Reactor Coolant System (RCS) temperature is in excess of 212 degrees F provided OPERATIONAL MODE 3, "HOT SHUTDOWN." LCO requirements for secondary containment isolation; secondary containment integrity; secondary containment automatic isolation dampers; and standby gas treatment are met.

C. BASES FOR THE CURRENT REQUIREMENT

All components in the RCS 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 3.9.1.1.1 of the Dresden Nuclear Power Station Updated Final Safety Analysis Report (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.

Four RPV regions are considered for the development of the pressure-temperature (P-T) curves: 1) the reactor core beltline region; 2) the non-beltline region (other than the closure flange region and the bottom head region); 3) the closure flange region, and 4) the bottom head region. The reactor core beltline region is defined as the region of the RPV that directly surrounds the effective height of the active core and is subject to a Reference Temperature Nil-Ductility Transition (RT_{NDT}) adjustment to account for radiation embrittlement. The non-beltline, closure flange, and bottom head regions receive insufficient neutron fluence to necessitate a RT_{NDT} adjustment. These regions contain components which include; the RPV nozzles, RPV closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that are not directly subjected to neutron radiation damage. Although the closure flange and bottom head regions are non-beltline regions, they are treated separately for the development of the P-T curves to address 10 CFR Part 50, Appendix G "Fracture Toughness Requirements," requirements.

The purpose of this Special Test Exception LCO is to allow inservice leak and hydrostatic tests of the RPV to be performed in OPERATIONAL MODE 4 when the metallurgical characteristics of the RPV require pressure testing at temperatures greater than 212 degrees F, which normally corresponds to OPERATIONAL MODE 3.

Pressure testing required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler & Pressure and Vessel (B&PV) Code "Rules for Inservice Inspection of Nuclear Power Plant Components," are performed prior to startup after a refueling outage. The minimum temperatures at the required pressures allowed for these tests are determined from the RPV P-T limits required by TS Section 3.6.K, "Pressure-Temperature Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased RPV neutron fluence over time, the minimum allowable RPV temperature increases at a given pressure. Pressure testing will eventually be required with minimum RCS temperatures that are greater than 212 degrees F.

D. NEED FOR REVISION OF THE REQUIREMENT

The proposed changes are a result of recently approved ASME methodology for determining allowable P-T limits.

The resultant benefits of the proposed changes include the following.

- Reduction in the challenges to operators in conducting pressure testing the RCS in excess of 212 degrees F and maintaining the RCS within a narrow temperature band.
- Personnel safety; conducting inspections at lower coolant temperatures, eliminates steam vapor hazards.
- 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 RCS temperature and RPV pressure requirements for testing.

- Improved leak detection afforded by observation of water leakage versus observation of steam vapor.
- Reduction in the potential to spread contamination in containment with the absence of steam vapor.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes revise TS Figures 3.6.K-1 through 3.6.K-5, and delete the Special Test Exception for Inservice Leak and Hydrostatic testing operation. The proposed TS Figures 3.6.K-1 through 3.6.K-3 are bounding P-T curves for Unit 2 and Unit 3 RPVs.

Bases changes that are affected by these proposed changes are also included for completeness.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The proposed changes to the P-T limits have been developed in accordance with the technical requirements of the ASME B&PV Code, Section XI, Appendix G as modified by ASME Code Cases N-588 and N-640.

<u>ASME Code Case N-588</u>

The current ASME B&PV Section XI, Appendix G approach requires the consideration of an axially oriented flaw in circumferential welds for purposes of calculating P-T limits. Postulating the ASME Code Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the RPV thickness, and is much longer than the width of the RPV girth (i.e., circumferential) welds. The fabrication of RPVs 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 procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual RPV welds, confirm that any remaining flaws are small, laminar in nature, and do not cross traverse to the weld bead orientation. Therefore, any defects potentially introduced during the fabrication process and not detected during the subsequent non-destructive examinations should only be oriented along the direction of weld fabrication. For circumferential welds, this indicates a postulated defect with a circumferential orientation.

Using ASME Code Case N-588 to determine P-T limits in conjunction with ASME B&PV Section XI Appendix G, provides appropriate and conservative procedures to determine limiting maximum postulated defects and to consider those defects in the determination of the P-T limits. The application of this code case maintains the margin of safety for circumferential welds equivalent to that originally contemplated for plates, forgings and axial welds.

ASME Code Case N-640

The proposed 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 of fracture toughness. The other margins involved with the ASME B&PV Section XI, Appendix G process to determine 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 is technically more correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the static initiation fracture toughness because the controlled heatup and cooldown process limits the rate at which stress is developed in the RPV wall to rates that are more appropriate for the static initation fracture toughness.

When the K_{Ia} curve was codified in 1974, the initial conservatism of the K_{Ia} curve was necessary due to limited experience and knowledge of RPV material fracture toughness. The conservatism also provided margin thought to be necessary to cover other uncertainties and the postulated effects of operating loads.

Since 1974, additional knowledge has been gained from examination and testing of RPVs that has reduced many of these uncertainties and resolved the postulated effects from operating loads. Since the original formulation of the ASME B&PV Code, Section XI, K_{Ia} and K_{Ic} curves in 1972, the fracture toughness database has been increased by orders of magnitude, and both remain lower bound curves. The additional data has significantly reduced the uncertainties associated with material fracture toughness. The new information indicates that the lower bound on fracture toughness provided by the K_{Ic} curve is extremely conservative. This lower bound on fracture toughness provides a greater margin of safety beyond that which is required to protect public health and safety from a potential RPV failure.

Details of the evaluations performed to calculate the P-T limits using this methodology are provided in Attachment G.

Similar license amendment and exemption requests were submitted by Quad Cities Nuclear Power Station in Reference 1 and granted by the NRC in References 2 and 3. The revised P-T limits for the Dresden Nuclear Power Station reactor vessels were calculated using the same analytical methods that were used to develop the revised reactor vessel P-T limits for Quad Cities Nuclear Power Station. The material properties used with this methodology were specifically from the materials used to fabricate each Dresden Nuclear Power Plant vessel. These vessel material properties were the same as those used to generate the current vessel P-T limits, except for the beltline materials. The beltline material properties are consistent with our most recent responses to Generic Letter 92-01. The normal operating and anticipated transients used to define the vessel stresses were identical for both stations.

As a conservative measure, the bounding 32 EFPYs neutron fluence value of 5.1×10^{17} n/cm^2 from the RPVs at both Dresden and Quad Cities Nuclear Power Stations was used to adjust the beltline material RT_{NDT} values.

G. IMPACT ON PREVIOUS SUBMITTALS

ComEd has reviewed the proposed changes regarding impact on any previous submittals and has determined that there is no impact on any outstanding previous submittals.

H. SCHEDULE REQUIREMENTS

We request approval of this amendment prior to August 1, 2000, to support activities in the Unit 3 refueling outage currently scheduled to begin September 2000.

I. REFERENCES

None.

ATTACHMENT B, Proposed Change to Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3

MARKED-UP PAGES FOR PROPOSED CHANGES

<u>REVISED PAGE(S)</u>

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Bases	**Pages B 3/4.6-6** **B 3/4.6-7**
Technical Specification	**Pages 3/4.12-3** **3/4.12-4**
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The primary system coolant system temperature and reactor vessel metal temperature and pressure shall be limited as specified below:

1. Pressure Testing:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figures e 3.6.K-1 ~~through 3.6.K-3~~ with the rate of change of the primary system coolant temperature ≤20°F per hour, or

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour when reactor vessel metal temperature and pressure is maintained within the Acceptable Regions as shown on Figure 3.6.K-[illegible]. 2

2. Non-Nuclear Heatup and Cooldown and low power PHYSICS TESTS:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-[illegible], and 2

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During non-nuclear heatup or cooldown, and pressure testing operations, at least once per 30 minutes,

   a. The rate of change of the primary system coolant temperature shall be determined to be within the heatup and cooldown rate limits, and

   b. The reactor vessel metal temperature and pressure shall be determined to be within the Acceptable Regions on Figures 3.6.K-1 through 3.6.K-[illegible]. 2

2. For reactor critical operation, determine within 15 minutes prior to the withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown,

   a. The rate of change of the primary system coolant temperature to be within the limits, and

   b. The reactor vessel metal temperature and pressure to be within the Acceptable Region on Figure 3.6.K-[illegible]. 3

3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-6 3, and

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4. The reactor vessel flange and head flange temperature ≥83°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

ACTION:

With any of the above limits exceeded,

1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes without exceeding the applicable primary system coolant temperature rate of change limit, and

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations within 72 hours, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

4. The reactor vessel flange and head flange temperature shall be verified to be ≥83°F:

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

      1) ≤113°F, at least once per 12 hours.

      2) ≤93°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.

~~FIGURE 3.6.K-1~~

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 18 EFPY

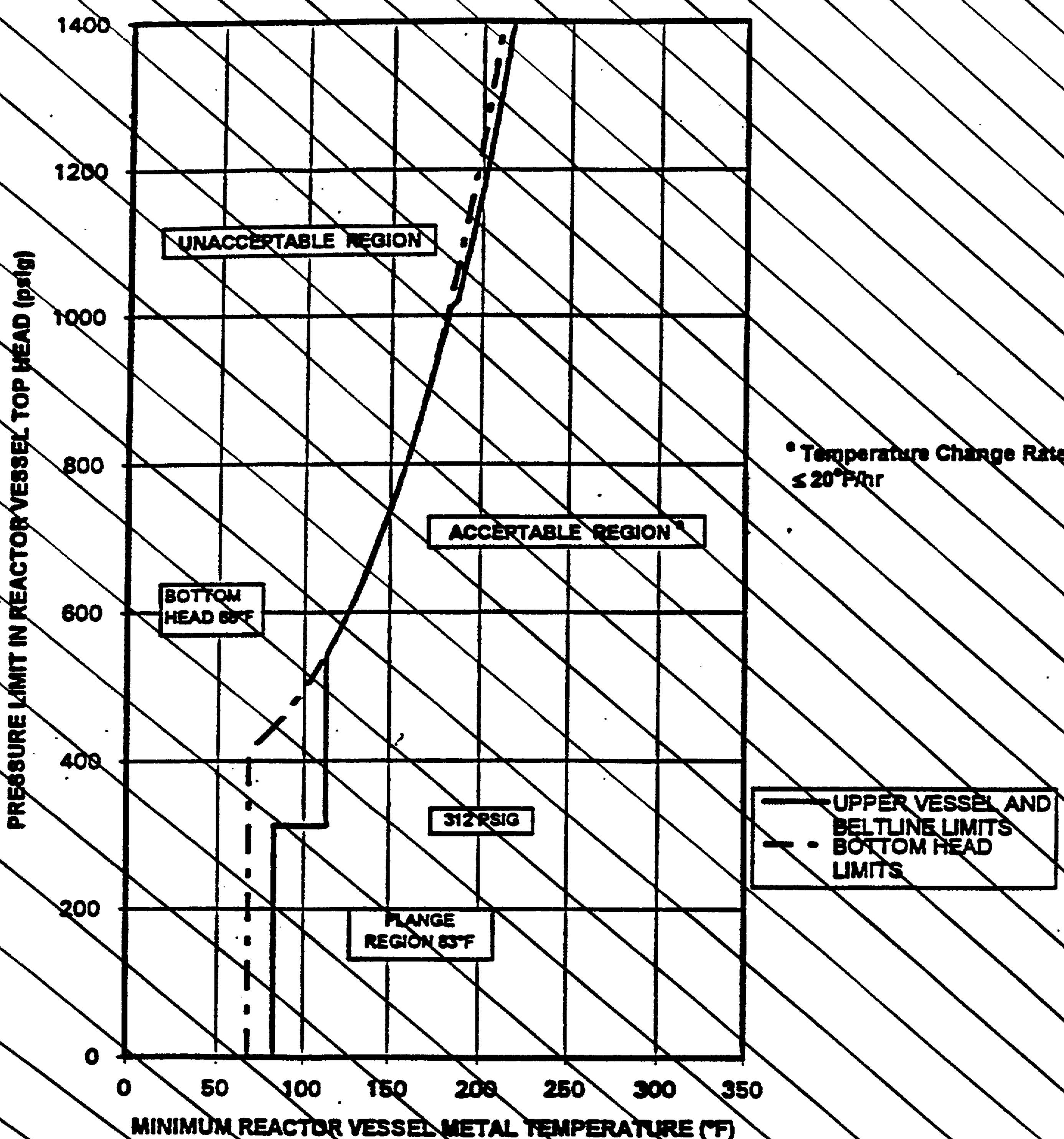


FIGURE 3.6.K-1

PRESSURE – TEMPERATURE LIMITS FOR PRESSURE TESTING – VALID TO 32 EFPY

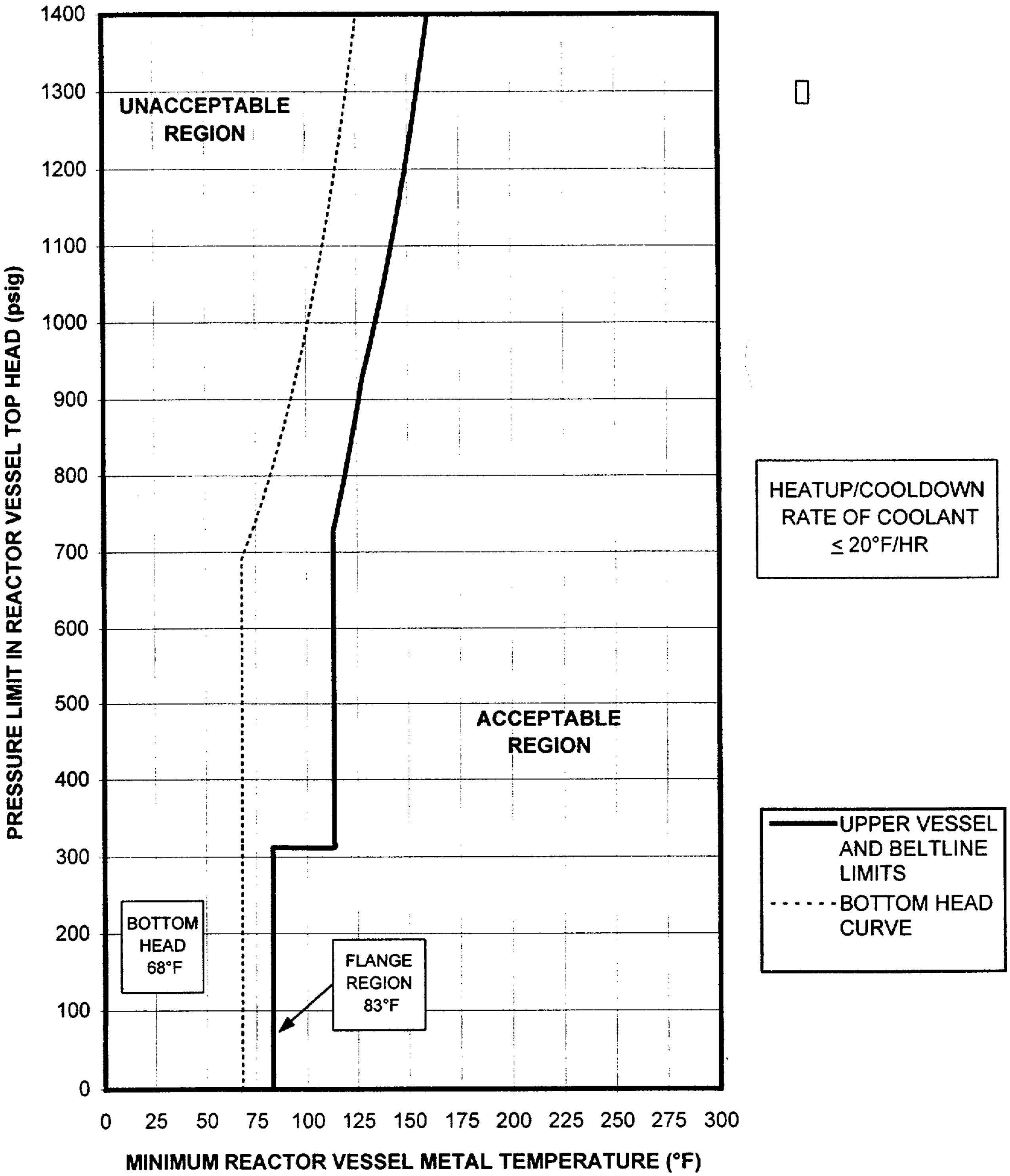


FIGURE 3.6.K-2

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 20 EFPY

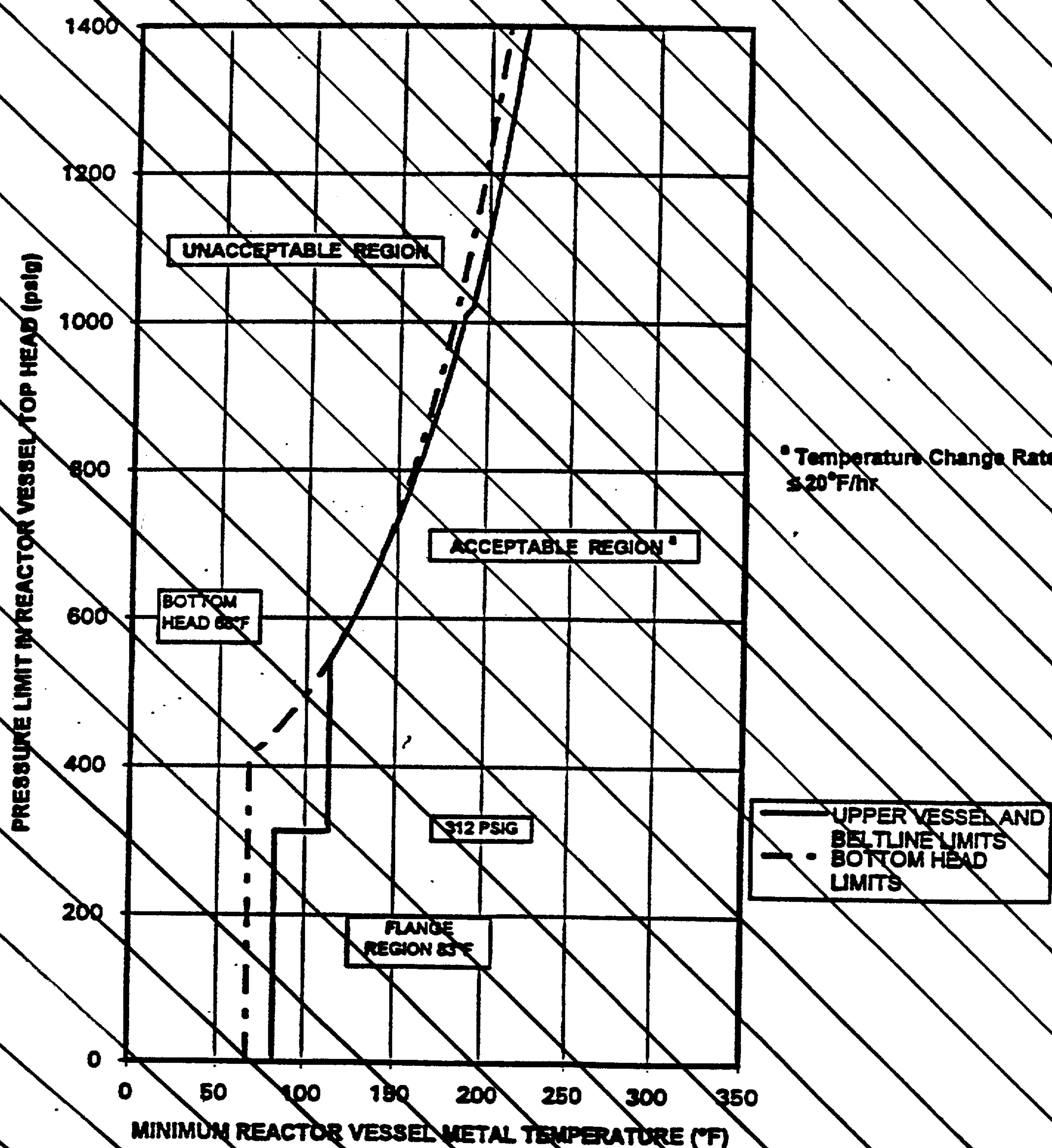
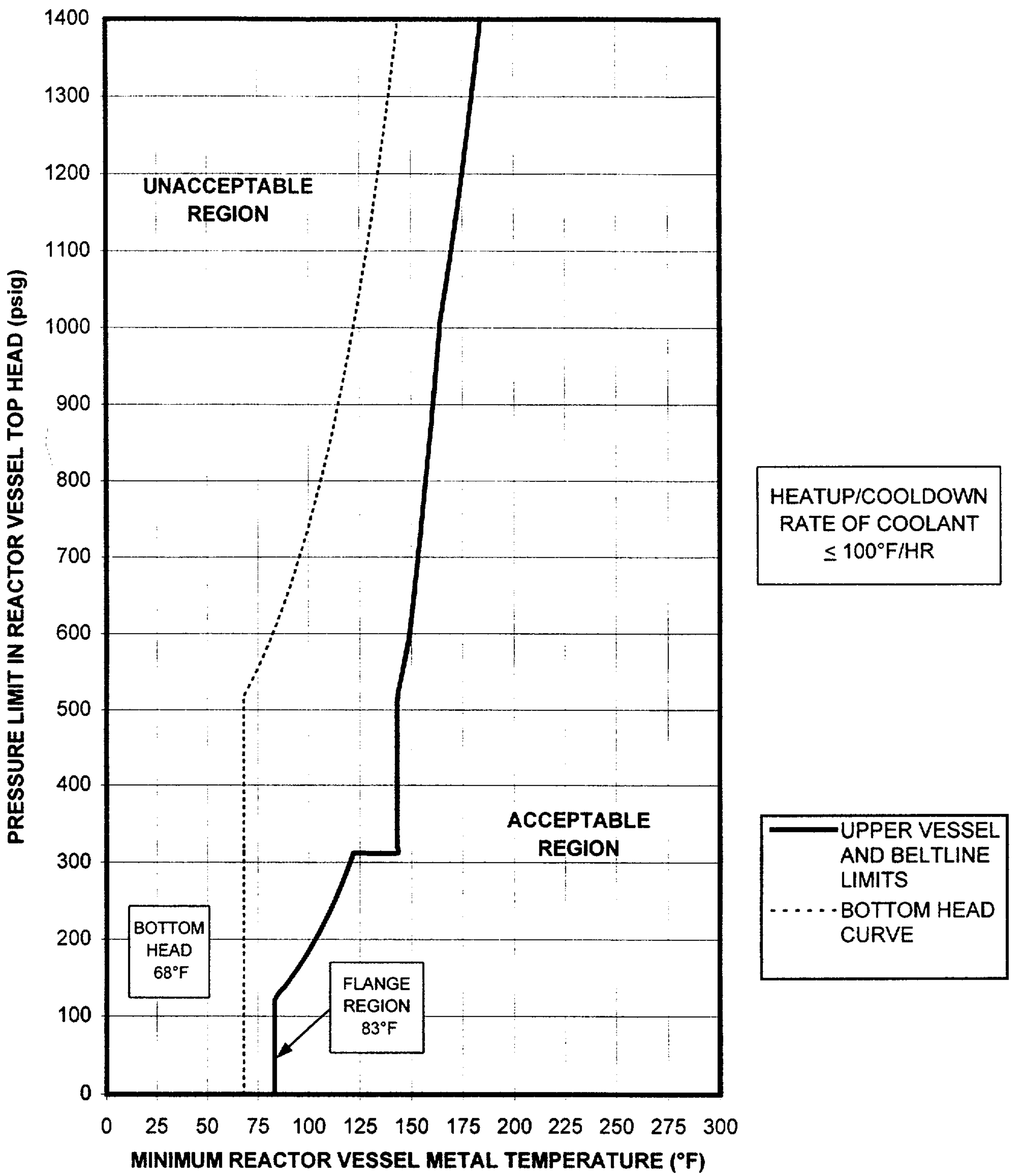


FIGURE 3.6.K-2

PRESSURE – TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN – VALID TO 32 EFPY



INSERT Figure 3.6.K-3

~~FIGURE 3.6.K-3~~

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 22 EFPY

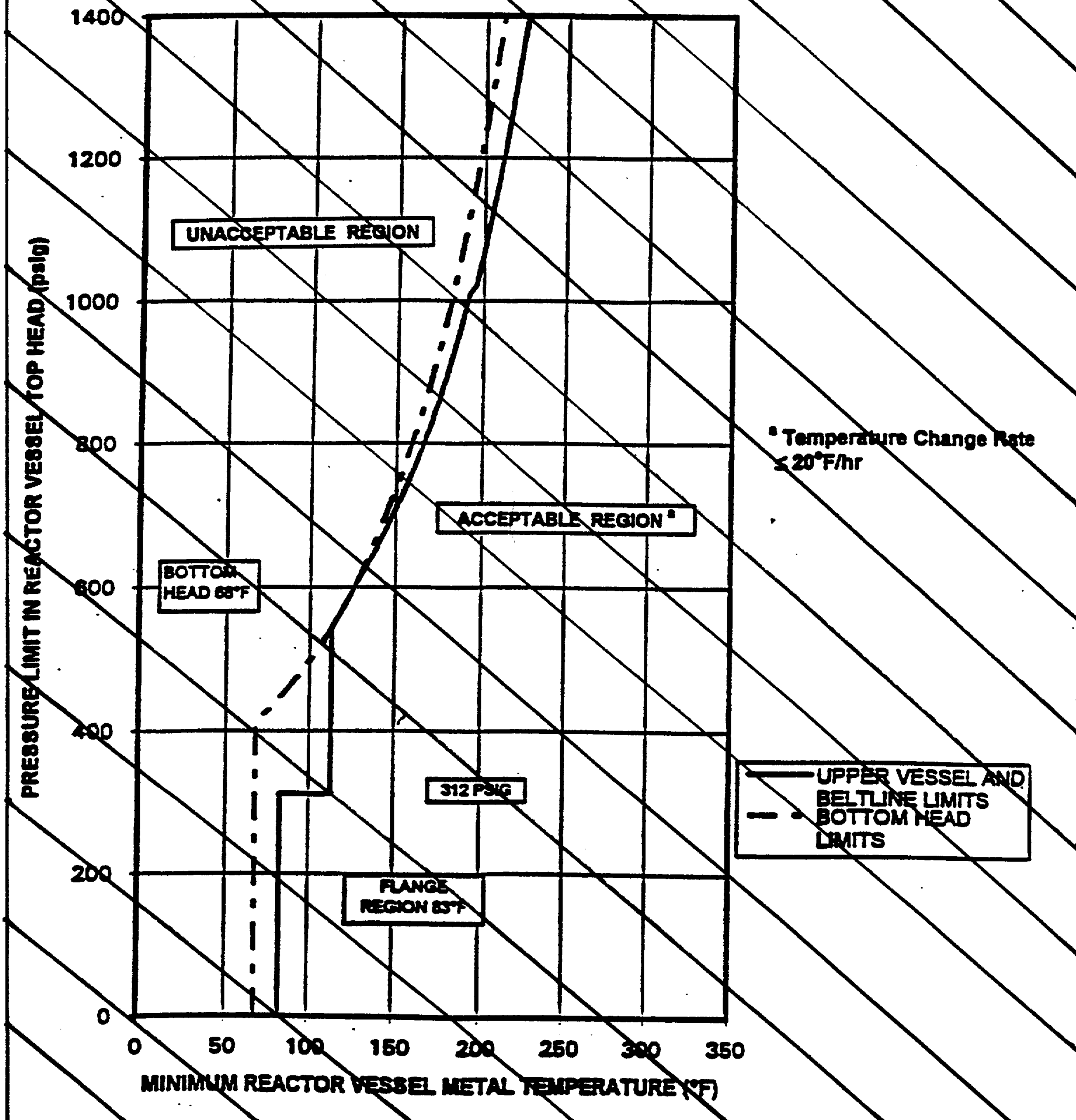


FIGURE 3.6.K-3

PRESSURE - TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 32 EFPY

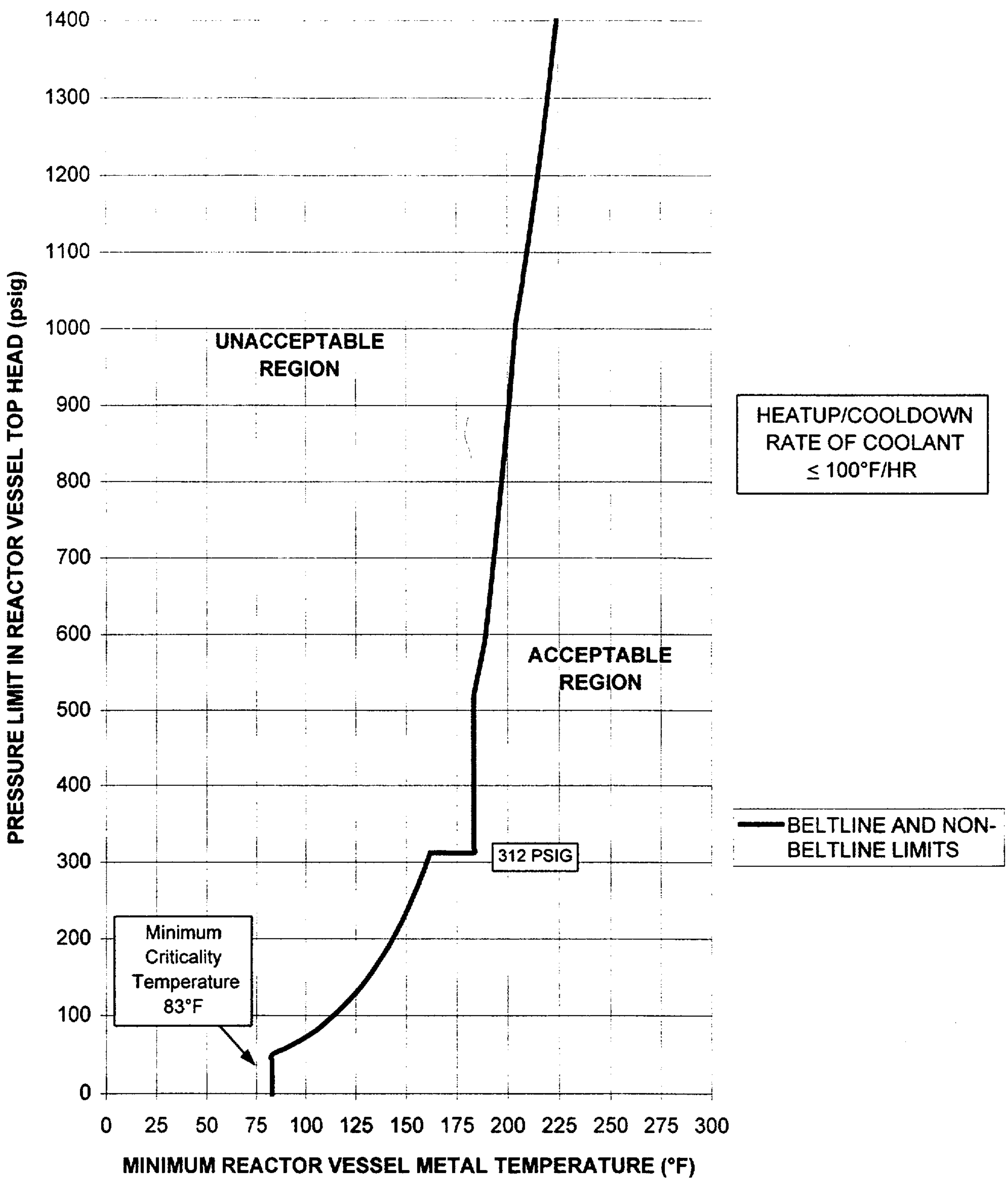


FIGURE 3.6.K-4

PRESSURE - TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN - VALID TO 22 EFPY

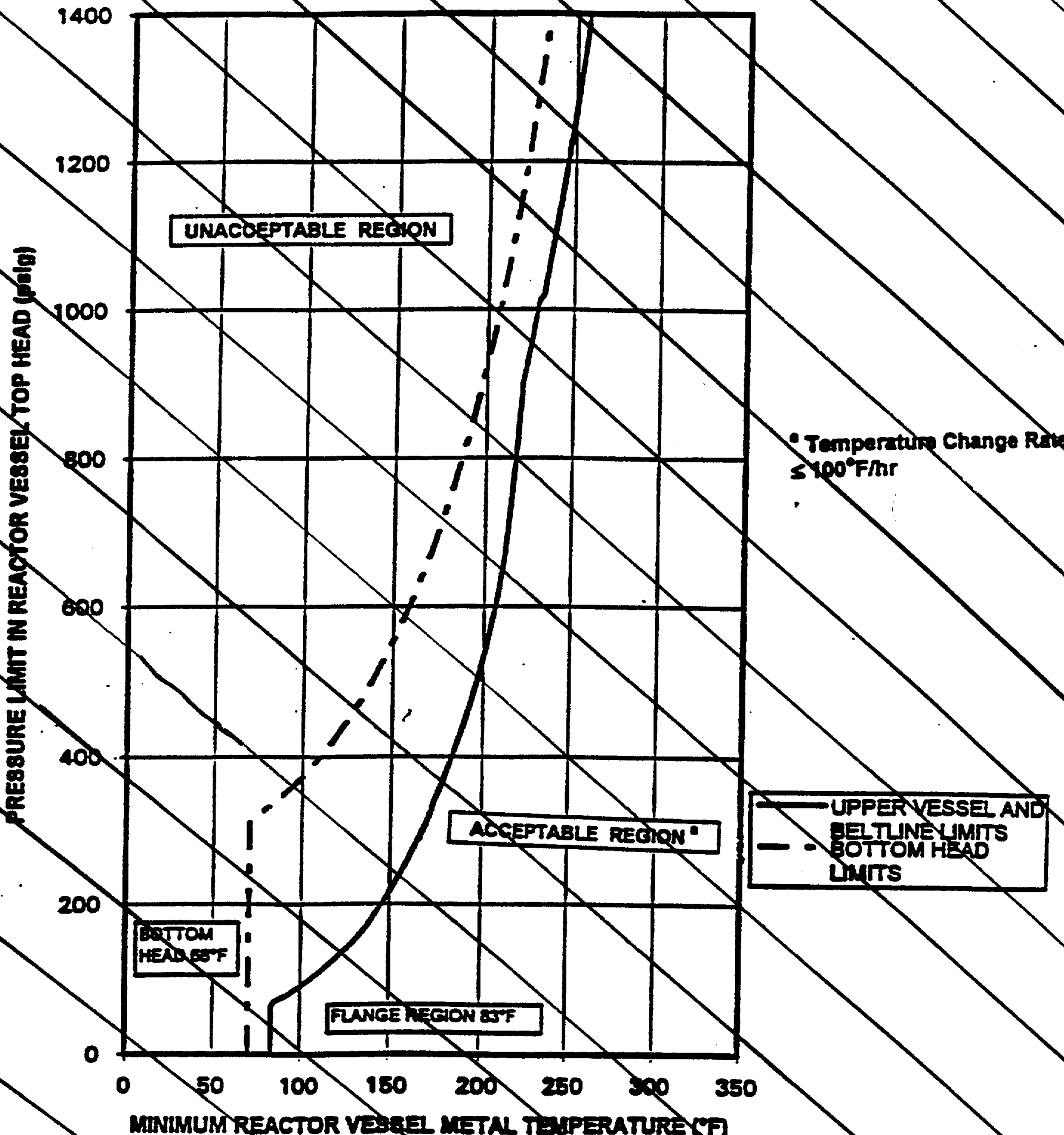
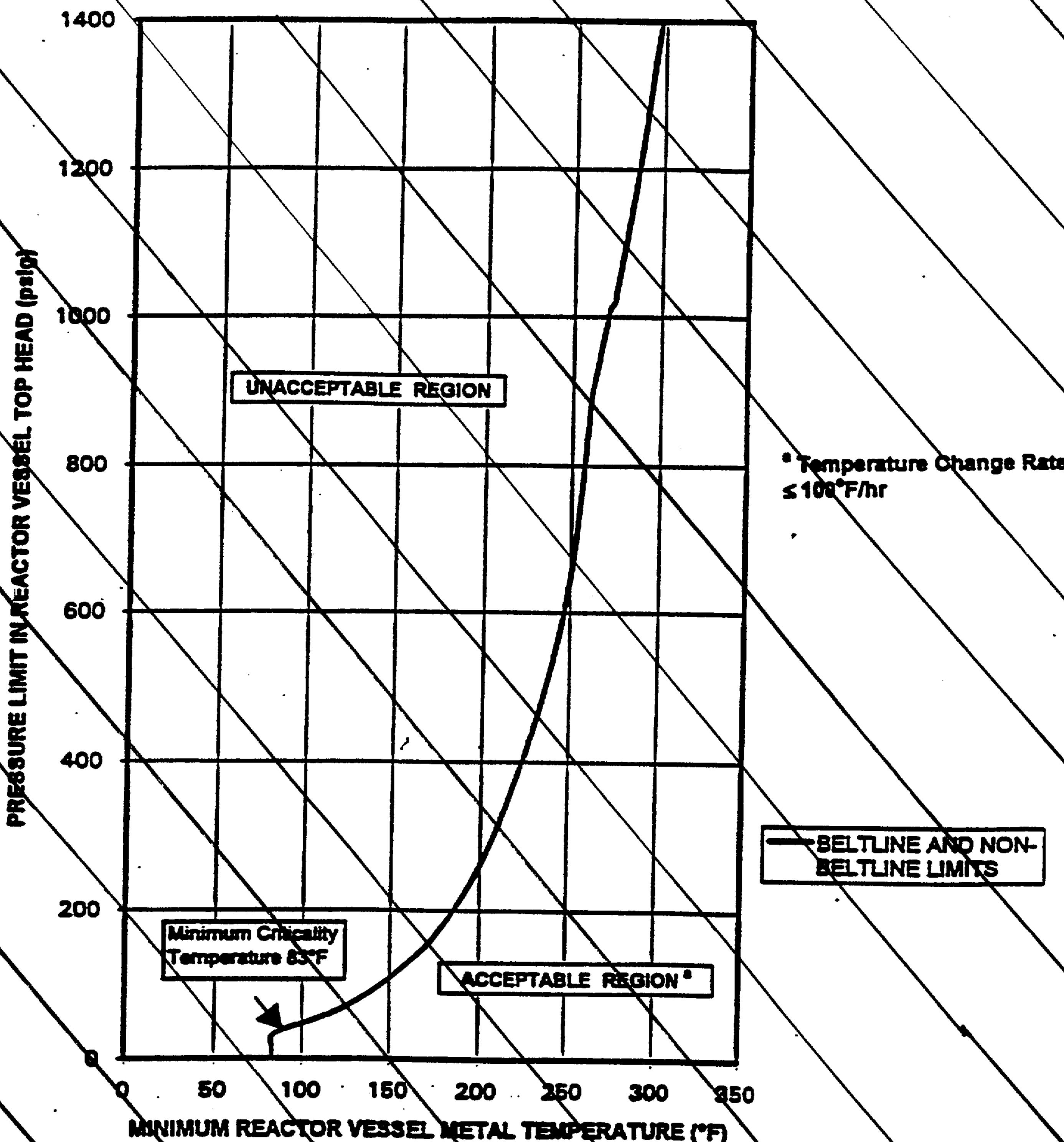


FIGURE 3.6.K-5

PRESSURE - TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 22 EFPY



BASES

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.

3 2

The pressure-temperature limit lines are shown, for operating conditions; Pressure Testing, Figures 3.6.K-1 ~~through 3.6.K-3~~, Non-Nuclear Heatup/Cooldown, Figure 3.6.K-4, and Core Critical Operation Figure 3.6.K-5. The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Four vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region and the bottom head region); 3) the closure flange region and 4) the bottom head region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline, closure flange, and bottom head regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange and bottom head regions are non-beltline regions, they are treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, connecting welds and the vertical electroslag welds which terminate immediately below the vessel flange is 23°F. Therefore, the minimum allowable boltup temperature is established as 83°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Figures 3.6.K-1 ~~through 3.6.K-3~~ - Pressure Testing

As indicated in Figure 3.6.K-1 ~~through 3.6.K-3~~ for pressure testing, the minimum metal temperature of the reactor vessel shell is 83°F for reactor pressures less than 312 psig. This 83°F minimum boltup temperature is based on a RT_{NDT} of 23°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. The bottom head region limit is established as 68°F, based on moderator temperature assumptions for shutdown margin analyses. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 113°F. The 113°F minimum temperature is based on a closure flange region RT_{NDT} of 23°F and a 90°F conservatism required by 10CFR Part 50 Appendix G. Beltline curves as a function of vessel exposure for ~~18, 20 and 22~~ effective full power years (EFPY) are presented~~, to allow the use of the appropriate curve up to 22 EFPY of operation~~.

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is

Figure~~s~~ 3.6.K-1 ~~through 3.6.K-3 are~~ governing for applicable pressure testing with a maximum heatup/cooldown rate of 20°F/hour.

2

Figure 3.6.K-4 - Non-Nuclear Heatup/Cooldown

2

Figure 3.6.K-4 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. The maximum heatup/cooldown rate of 100°F/hour is applicable.

3

Figure 3.6.K-5 - Core Critical Operation

3

The core critical operation curve shown in Figure 3.6.K-5, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any pressure testing or non-nuclear heatup/cooldown limits. ~~Since Figure 3.6.K-4 is more limiting, Figure 3.6.K-5 is Figure 3.6.K-4 plus 40°F. The maximum heatup/cooldown rate of 100°F/hour is applicable.~~

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens are used in predicting reactor vessel material embrittlement. The operating limit curves of Figures 3.6.K-1 through 3.6.K-5 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

3

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M 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, however, single failure considerations require that two

3.12 - LIMITING CONDITIONS FOR OPERATION

C. Inservice Leak and Hydrostatic Testing Operation

The average reactor coolant temperature specified in Table 1-2 for OPERATIONAL MODE 4 may be changed to "NA," and operation considered not to be in OPERATIONAL MODE 3; and the requirements of LCO 3.6.P, "Shutdown Cooling - COLD SHUTDOWN," may be suspended, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL MODE 3 LCOs are met:

1. LCO 3.2.A, "Isolation Actuation", Table 3.2.A-1, Functional Unit Number 2, "SECONDARY CONTAINMENT ISOLATION";
2. LCO 3.7.N, "SECONDARY CONTAINMENT INTEGRITY";
3. LCO 3.7.O, "Secondary Containment Automatic Isolation Dampers"; and
4. LCO 3.7.P, "Standby Gas Treatment System."

APPLICABILITY:

OPERATIONAL MODE 4 with average reactor coolant temperature >212°F.

ACTION:

With one or more of the above requirements[a] not met:

4.12 - SURVEILLANCE REQUIREMENTS

C. Inservice Leak and Hydrostatic Testing Operation

Perform the applicable surveillance requirements for the required OPERATIONAL MODE 3 LCOs in accordance with the frequency of the applicable surveillance requirements.

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a Separate ACTION entry is allowed for each requirement of the LCO.

3.12 - LIMITING CONDITIONS FOR OPERATION

4.12 - SURVEILLANCE REQUIREMENTS

1. Immediately enter the applicable ACTION of the affected LCO[b], or

2. Immediately suspend activities that could increase the average reactor coolant temperature or pressure, and reduce average reactor coolant temperature to ≤212°F within 24 hours.

---

b Required ACTIONs to be in OPERATIONAL MODE 4 include reduce average coolant temperature ≤212°F.

BASES

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS. Low power PHYSICS TESTS during OPERATIONAL MODE 2 may be required to be performed while still maintaining access to the primary containment and reactor pressure vessel. Additional requirements during these tests to restrict reactor power and reactor coolant temperature provide protection against potential conditions which could require primary containment or reactor coolant pressure boundary integrity.

3/4.12.B SHUTDOWN MARGIN Demonstrations

Performance of SHUTDOWN MARGIN demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO. SHUTDOWN MARGIN tests may be performed while in OPERATIONAL MODE 2 in accordance with Table 1-2 without meeting this Special Test Exception. For SHUTDOWN MARGIN demonstrations performed while in OPERATIONAL MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM, or must be verified by a second licensed operator or other technically qualified individual. To provide additional protection against inadvertent criticality, control rod withdrawals that are "out-of-sequence", i.e., do not conform to the Banked Position Withdrawal Sequence, must be made in individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATION(s) may be in progress. This Special Test Exception then allows changing the Table 1-2 reactor mode switch position requirements to include the Startup or Hot Standby position such that the SHUTDOWN MARGIN demonstrations may be performed while in OPERATIONAL MODE 5.

~~3/4.12.C Inservice Leak and Hydrostatic Testing Operation~~

~~The purpose of this Special Test Exception LCO is to allow certain reactor coolant pressure tests to be performed in OPERATIONAL MODE 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) require pressure testing at temperatures > 212°F, which normally corresponds to OPERATIONAL MODE 3.~~

~~Pressure Testing required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to startup after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.6.K, "Pressure/Temperature Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Pressure testing will eventually be required with minimum reactor coolant temperatures > 212°F.~~

BASES

Allowing the reactor to be considered in OPERATIONAL MODE 4 during pressure testing, when the reactor coolant temperature is >212°F, effectively provides an exception to OPERATIONAL MODE 3 requirements, including OPERABILITY of primary containment and the full complement of redundant Emergency Core Cooling Systems. Since the pressure tests are performed at low decay heat values, and near OPERATIONAL MODE 4 conditions, the stored energy in the reactor core will be low. Under these conditions, the potential for failed fuel and a subsequent increase in coolant activity above LCO 3.6.J, "Specific Activity," limits are minimized. In addition, secondary containment will be OPERABLE, in accordance with this Special Test Exception LCO, and will be capable of handling any airborne radioactivity or steam leaks that could occur during the performance of pressure testing. The required pressure testing conditions provide adequate assurance that the consequences of a steam leak will be conservatively bounded by the consequences of the postulated main steam line break outside of primary containment described in the UFSAR. Therefore, these requirements will conservatively limit radiation releases to the environment.

In the event of a large primary system leak, the reactor vessel would rapidly depressurize, allowing the low pressure core cooling systems to operate. The capability of the low pressure coolant injection and core spray subsystems, as required in OPERATIONAL MODE 4 by LCO 3.5.B, "ECCS Shutdown," would be more than adequate to keep the core flooded under this low decay heat load condition. Minor system leaks would be detected by leakage inspections before significant inventory loss occurred.

For the purposes of this Special Test Exception, the protection provided by normally required OPERATIONAL MODE 4 applicable LCOs, in addition to the secondary containment requirements required to be met by this Special Test Exception LCO, will ensure acceptable consequences during normal pressure test conditions and during postulated accident conditions.

Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases. Compliance with this Special Test Exception LCO is optional. Operation at reactor coolant temperatures >212°F can be in accordance with Table 1-2 for OPERATIONAL MODE 3 operation without meeting this Special Test Exception LCO or its ACTIONs.

If it is desired to perform these tests while complying with this Special Test Exception LCO, then the OPERATIONAL MODE 4 applicable LCOs and specified OPERATIONAL MODE 3 LCOs must be met. This Special Test Exception LCO allows changing Table 1-2 temperature limits for OPERATIONAL MODE 4 to "NA" and suspending the requirements of LCO 3.6.P, "Shutdown Cooling - COLD SHUTDOWN." The additional requirements for secondary containment LCOs to be met will provide sufficient protection for operations at reactor coolant temperatures >212°F for the purpose of performing pressure testing.

This LCO allows primary containment to be open for frequent unobstructed access to perform inspections, and for outage activities on various systems to continue consistent with the

BASES

OPERATIONAL MODE 4 applicable requirements that are in effect immediately prior to and immediately after this operation.

The OPERATIONAL MODE 4 requirements may only be modified for the performance of inservice pressure tests so that these operations can be considered as in OPERATIONAL MODE 4, even though the reactor coolant temperature is >212°F. The additional requirement for secondary containment OPERABILITY according to the imposed OPERATIONAL MODE 3 requirements provides conservatism in the response of the unit to any event that may occur. Operations in all other OPERATIONAL MODES are unaffected by this LCO.

Footnote (a) has been provided to modify the ACTIONs related to pressure testing operation. Footnote (a) allows a separate condition entry for each requirement of the LCO.

If an LCO specified in LCO 3.12.C is not met, the ACTIONs applicable to the stated requirements are entered immediately. ACTION 1 has been modified by Footnote (b) that clarifies the intent of another LCO's ACTION to be in OPERATIONAL MODE 4 which includes reducing the average reactor coolant temperature to ≤212°F.

ACTION 2 is an alternate action that can be taken instead of ACTION 1 to restore compliance with the normal OPERATIONAL MODE 4 requirements, and thereby exit this Special Test Exception LCO's Applicability. Activities that could further increase reactor coolant temperature or pressure are suspended immediately, in accordance with ACTION 2, and the reactor coolant temperature is reduced to establish normal OPERATIONAL MODE 4 requirements. The allowed completion time of 24 hours for ACTION 2 provides sufficient time to reduce the average reactor coolant temperature from the highest expected value to ≤212°F with normal cooldown procedures. The completion time is also consistent with the time provided in LCO 3.0.C to reach OPERATIONAL MODE 4 from OPERATIONAL MODE 3.

The applicable LCOs are required to have their Surveillances met to establish that this LCO is being met. A discussion of the applicable Surveillance Requirements is provided in their respective Bases.

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison (ComEd) Company has evaluated the proposed changes to the Technical Specification (TS) for Dresden Nuclear Power Station, Units 2 and 3, and determined that it involves 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.

ComEd is proposing changes to the Pressure-Temperature (P-T) limits by revising the heatup, cooldown, and inservice test limitations for the Reactor Pressure Vessel (RPV) specified in Technical Specifications Section 3/4.6.K "Primary System Boundary". Furthermore, the proposed changes delete the Special Test Exception of TS Section 3/4.12.C "Special Test Exception" which allows for pressure testing at greater than 212° F.

The information supporting the determination that the criteria set forth in 10 CFR 50.92 is 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 do not modify the reactor coolant pressure boundary, do not make changes in operating pressure, materials or seismic loading. The proposed changes adjust the reference temperature for the limiting beltline material to account for radiation effects and provide the same level of protection as previously evaluated. The proposed changes do not adversely affect the integrity of the Reactor Coolant System (RCS) such that its function in the control of radiological consequences is affected. Therefore, the proposed changes do not involve a significant increase in the probability or 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 do not create the possibility of a new or different kind of accident previously evaluated for Dresden Nuclear Power Station. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Use of the revised P-T curves will continue to provide the same level of protection as was previously reviewed and approved.

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

Further, the proposed changes to the P-T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate any accident sequence for Dresden Nuclear Power Station. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes reflect an update of the P-T curves to extend the RPV operating limit to 32 Effective Full Power Years (EFPYs). The revised curves are based on the latest American Society of Mechanical Engineers (ASME) guidance and actual operational data for the units. These proposed changes are acceptable because the ASME guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME Section XI Appendix G was approved in 1974. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

..

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company 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. ComEd has determined that these proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has 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.

Exemption Request

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 Code (B&PV) Code, Section XI, Code Case N-640 "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1" 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, "Fracture Toughness Requirements," paragraph IV.A.2.b.

Justification for Use of ASME Code Case N-640

10 CFR 50.12(a) Requirements

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

10 CFR 50.12 states that the NRC 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, Appendix G when an exemption is granted by the NRC under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety. The revised P-T limits being proposed for Dresden Nuclear Power Station Units 2 and 3 rely in part, on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME 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 is technically more correct than the K_{Ia} curve. The K_{IC} curve appropriately implements the static initiation fracture toughness because the controlled heatup and cooldown process limits the rate at which stress is developed in the RPV wall to rates that are more appropriate for the static initation fracture toughness.

   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 RPV material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of RPV materials and their fracture response to applied loads. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety.

Exemption Request

P-T curves based on the K_{Ic} fracture toughness limits will enhance overall plant safety by opening the P-T 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 excess of 212 degrees in a limited operating window and improvement in personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by approval of 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 requested exemption meets the special circumstances of the following paragraphs of 10 CFR 50.12.

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

   (a) (2) (iii) – would result in undue hardship or other costs that are significant if the regulation is enforced and;

   (a) (2) (v) – will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

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

   1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the RPV 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 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 RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME B&PV Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable margin of safety.

Exemption Request

10 CFR 50.12(a) (2) (iii): The RPV P-T 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 Dresden Nuclear Power Station Units 2 and 3, with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P-T operating window. This restriction challenges the operations staff during pressure tests to maintain a high temperature, in excess of 212° F within a limited operating window. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

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

10 CFR 50.12(a) (2) (v): The requested exemption provides only temporary relief from the applicable regulation and Dresden Nuclear Power Station Units 2 and 3 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 Code Case N-640 for use by the nuclear industry.

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

Exemption Request

Justification for Use of to ASME Code Case N-588

10CFR50.12 Requirements:

The requested exemption to allow use of ASME 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 NRC 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, Appendix G 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 RPV for the P-T. 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 (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 P-T limits.

   ASME Code Case N-588 provides benefits, in terms of calculating the P-T limits, by revising the Article G-2120 reference flaw orientation for circumferential welds in RPVs. 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 RPV 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 RPVs 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.

Exemption Request

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

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 P-T 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. <u>The requested exemption will not endanger the common defense and security</u>: The common defense and security are not endangered by 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>: Pursuant to 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 costs that are significant if the regulation is enforced and;
   (a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

Exemption Request

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 RPV boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and
2) P-T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

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

10 CFR 50.12(a)(2)(iii): The RCS P-T operating window is defined by the P-T operating and test curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of with these P-T curves without the relief provided by ASME Code Case N-588 would unnecessarily restrict the P-T operating window for Dresden Nuclear Power Station, Units 2 and 3. This restriction challenges the operations staff during pressure tests to maintain a high temperature in excess of 212°F within a limited operating window. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

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

10CFR50.12(a)(2)(v): The exemption provides only temporary relief from the applicable regulation and Dresden Nuclear Power 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 Code Case N-588 for use by the nuclear industry.

Exemption Request

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 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 Code Case N-588 for Dresden Nuclear Power 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 RPV 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 P-T limits specified in TS Section 3.4.3. Therefore, this exemption does not present an undue risk to the public health and safety.

(February 1999)

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 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° 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.

As 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 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.

Overall Plant Safety is Improved

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.

Impact on PWRs:

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 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.

4. McCabe, D.E., "Assessment of Metallurgical Effects that Impact Pressure Vessel Safe Margin Issues", Oak Ridge Report ORNL/NRC/LTR-94/26, October 1994.

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.

(February 1999)

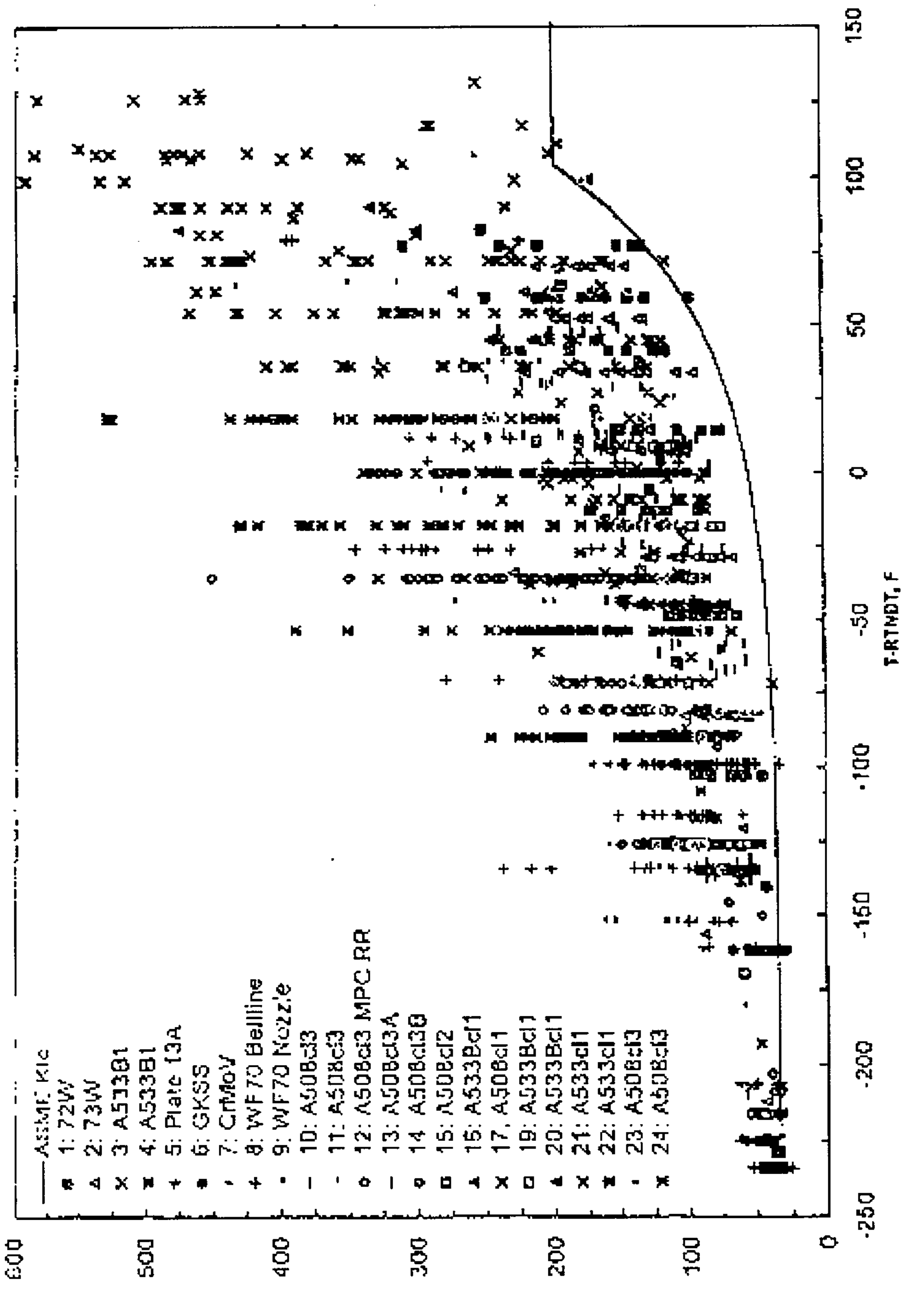
Table 1

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

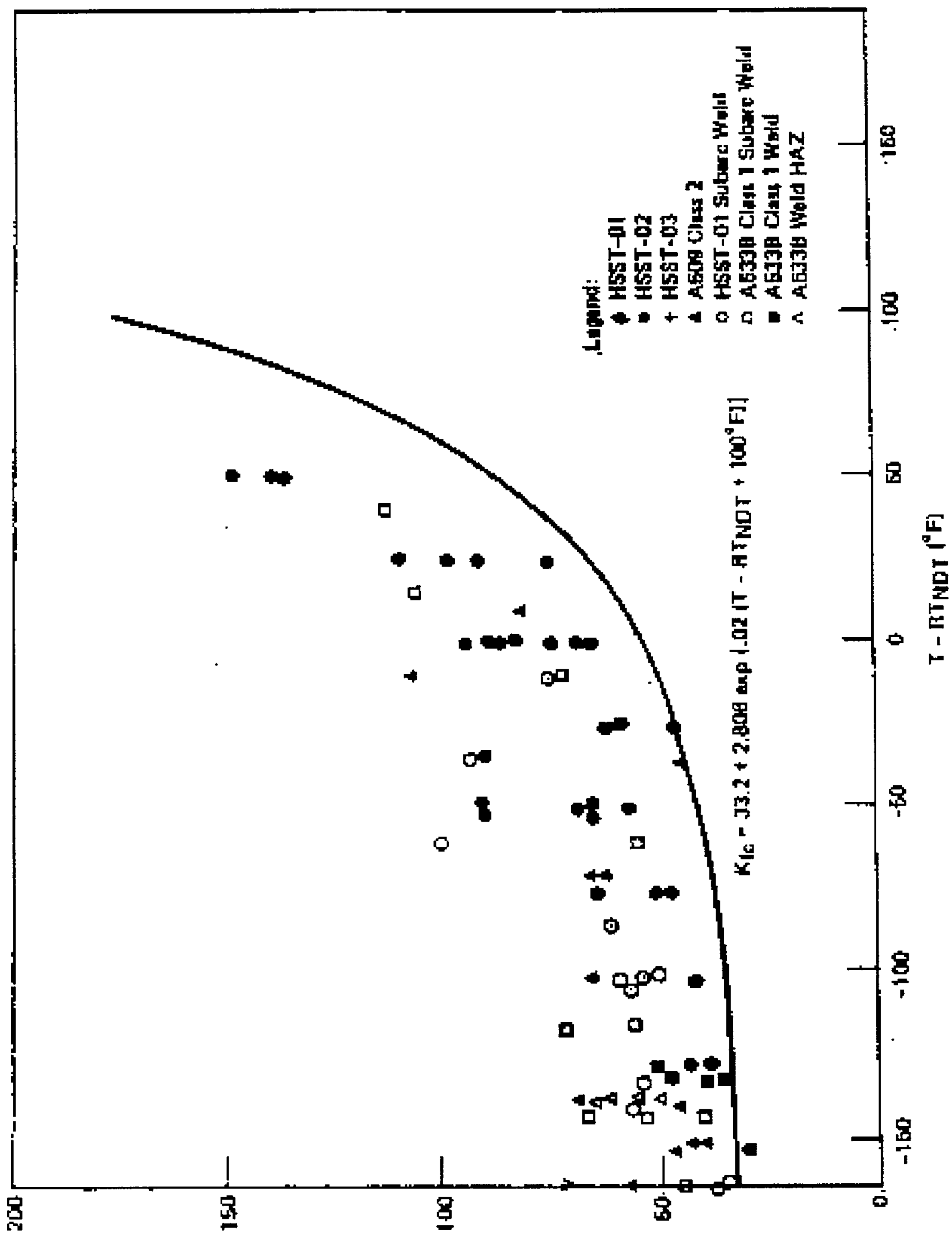
(February 1999)



Toughness, ksi √in

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

(February 1999)



K_{IC} (ksi √in)

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

(February 1999)

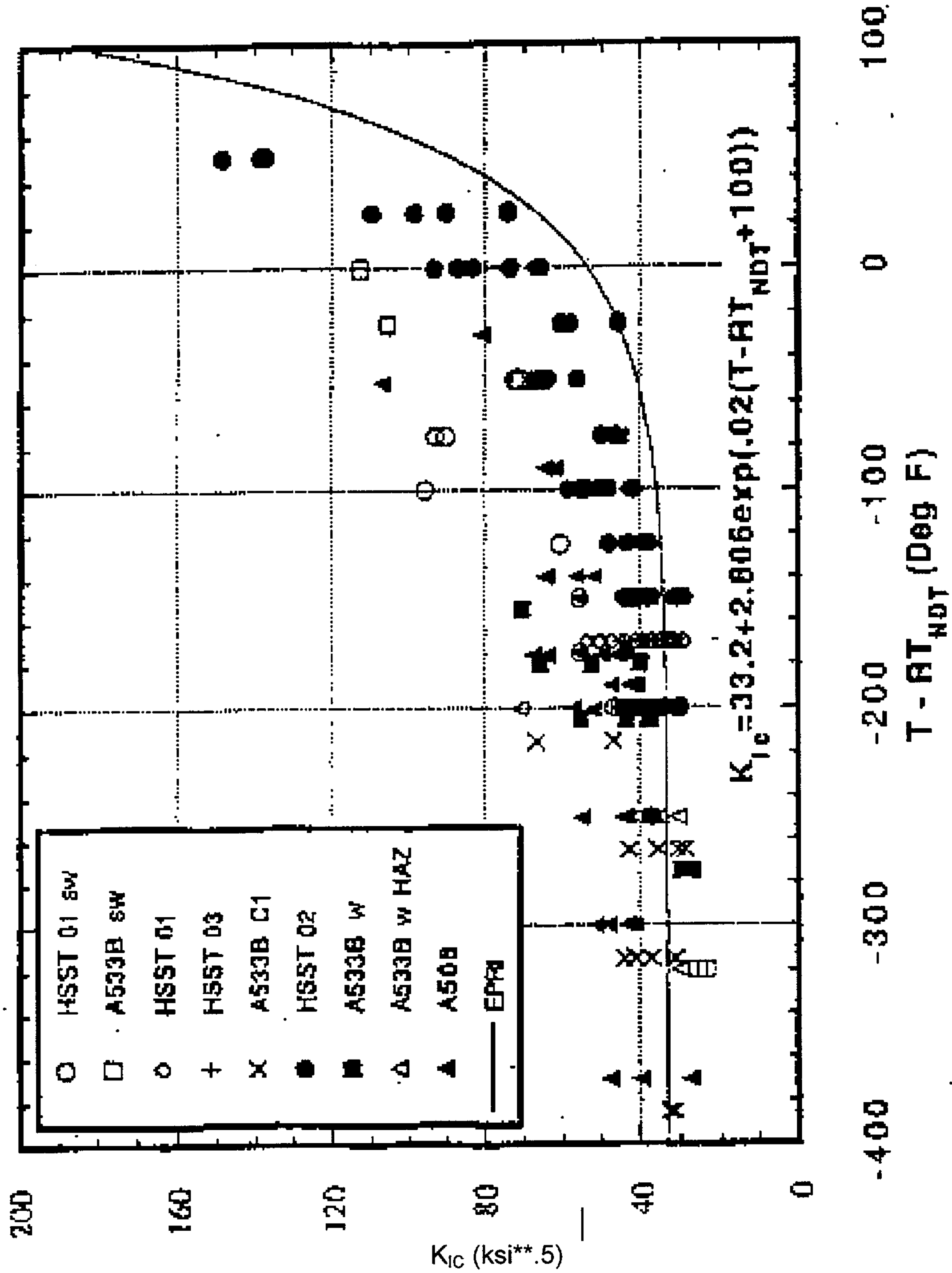


Figure 3. K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

(February 1999)

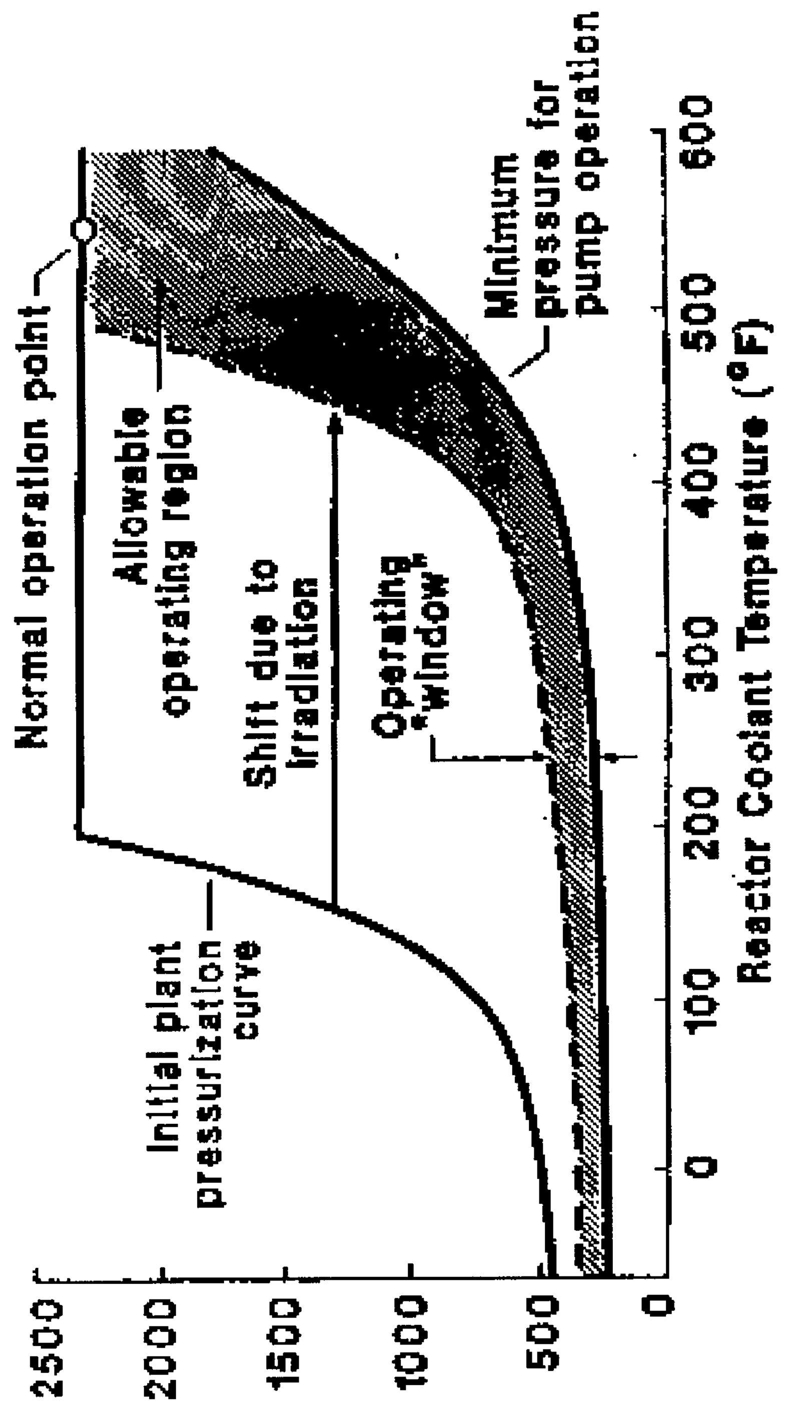


Figure 4. Operating Window From P-T Limit Curves [4]

(February 1999)

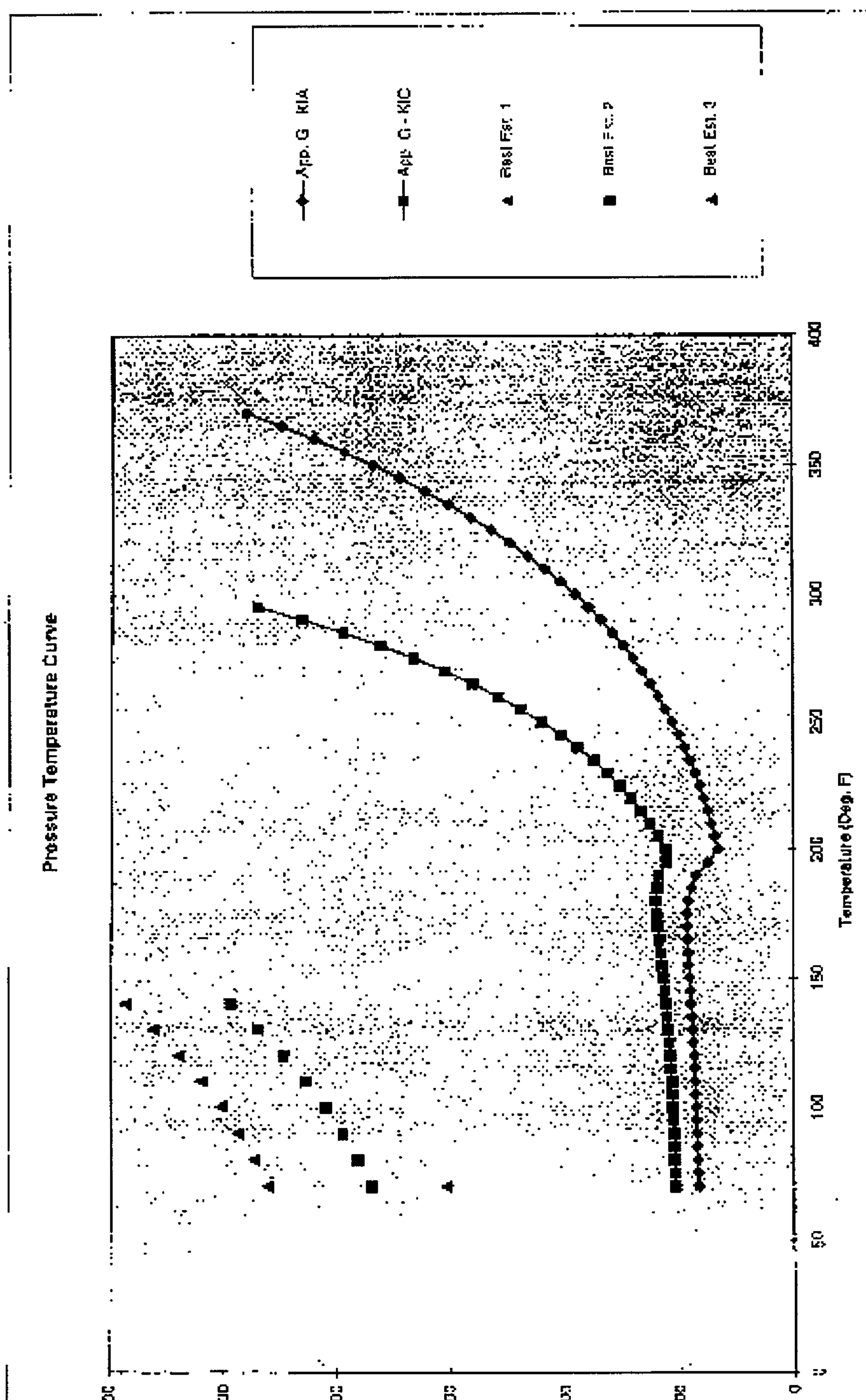


Figure 5. P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

(February 1999)

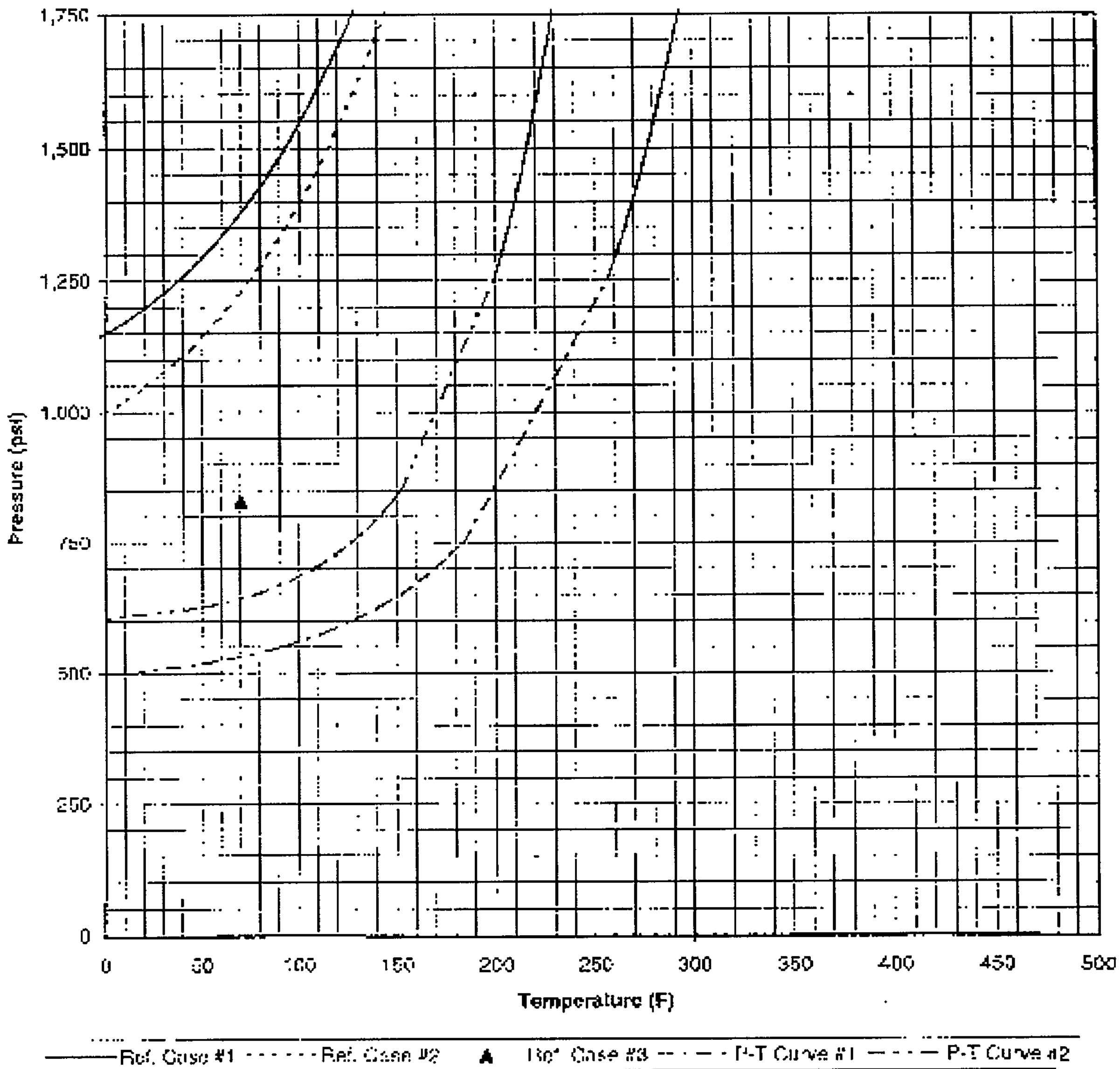


Figure 6. P-T Limit Curves Illustrating Deterministic Safety Factors
for a BWR Reactor Vessel

(February 1999)

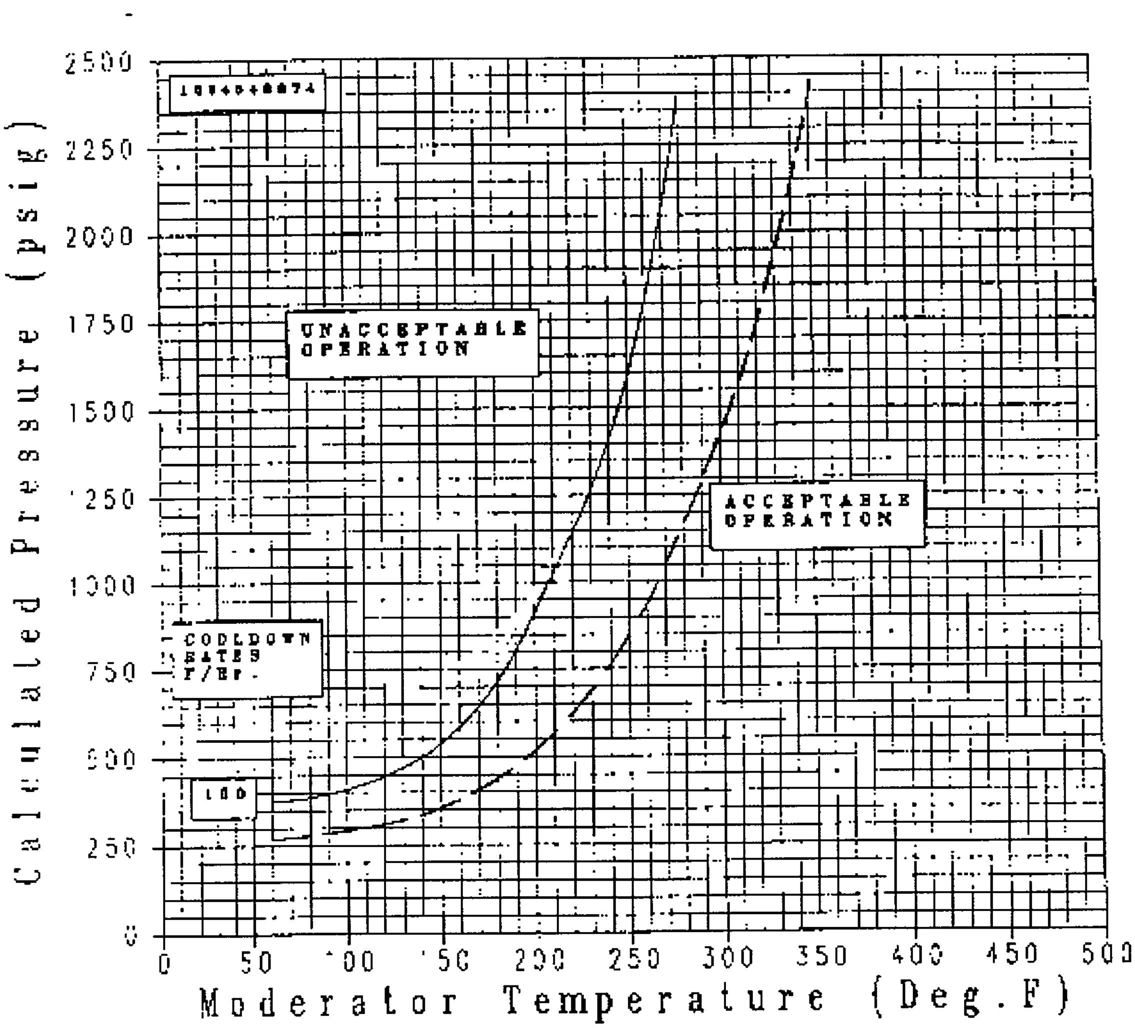


Figure 7. Comparison of Cool-Down Curves for the Existing and Proposed Methods - PWR [Dashed Curve = Existing (K_{Ia}) and Solid Curve = Proposed (K_{Ic})]

(February 1999)

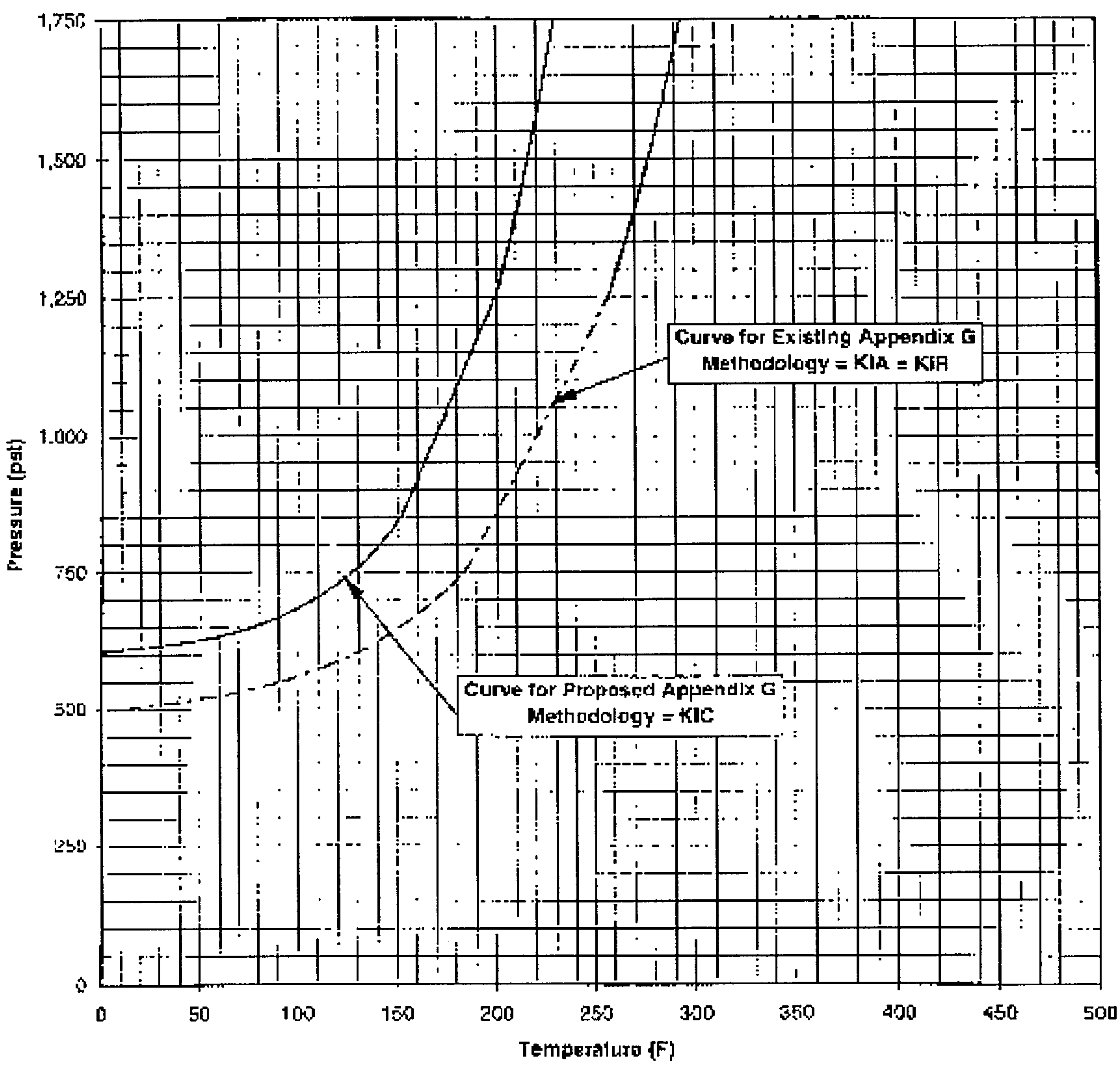


Figure 8. Comparison of Hydrotest P-T Curves for the Existing and Proposed Methods - BWR [Dashed Curve=Existing (K_{Ia}) and Solid Curve=Proposed (K_{Ic})]

(February 1999)

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.

(February 1999)

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.

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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.

(February 1999)

TABLE A-2: P-T Calculation Cases

<u>Calculation Case 1</u>

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.

<u>Calculation Case 2</u>

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

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

(February 1999)

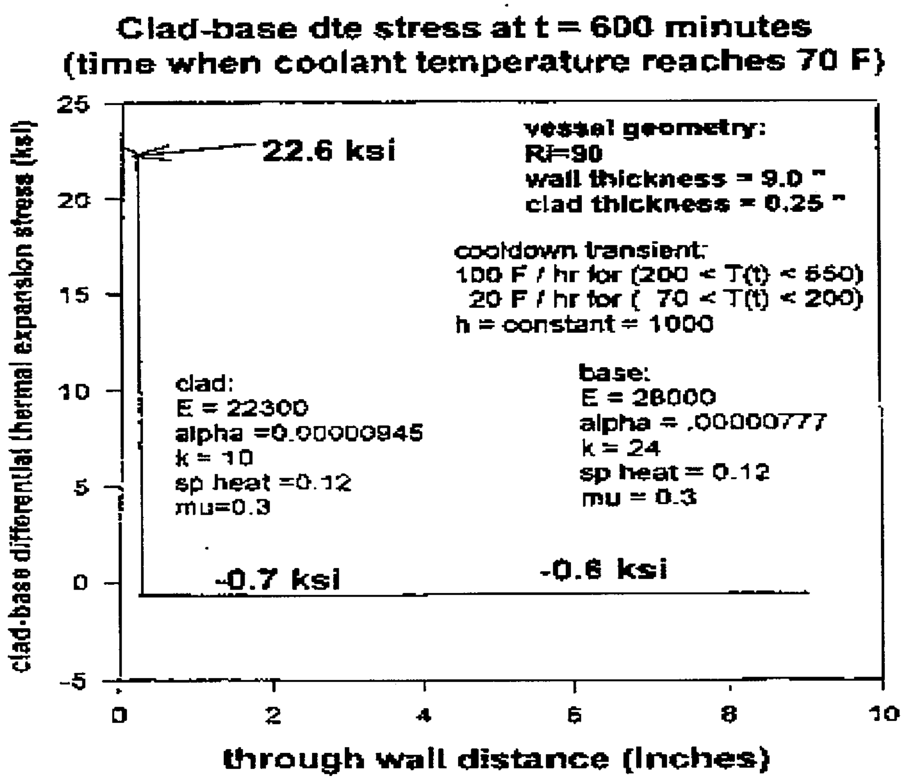

Figure A-1: Clad-base metal stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

(February 1999)

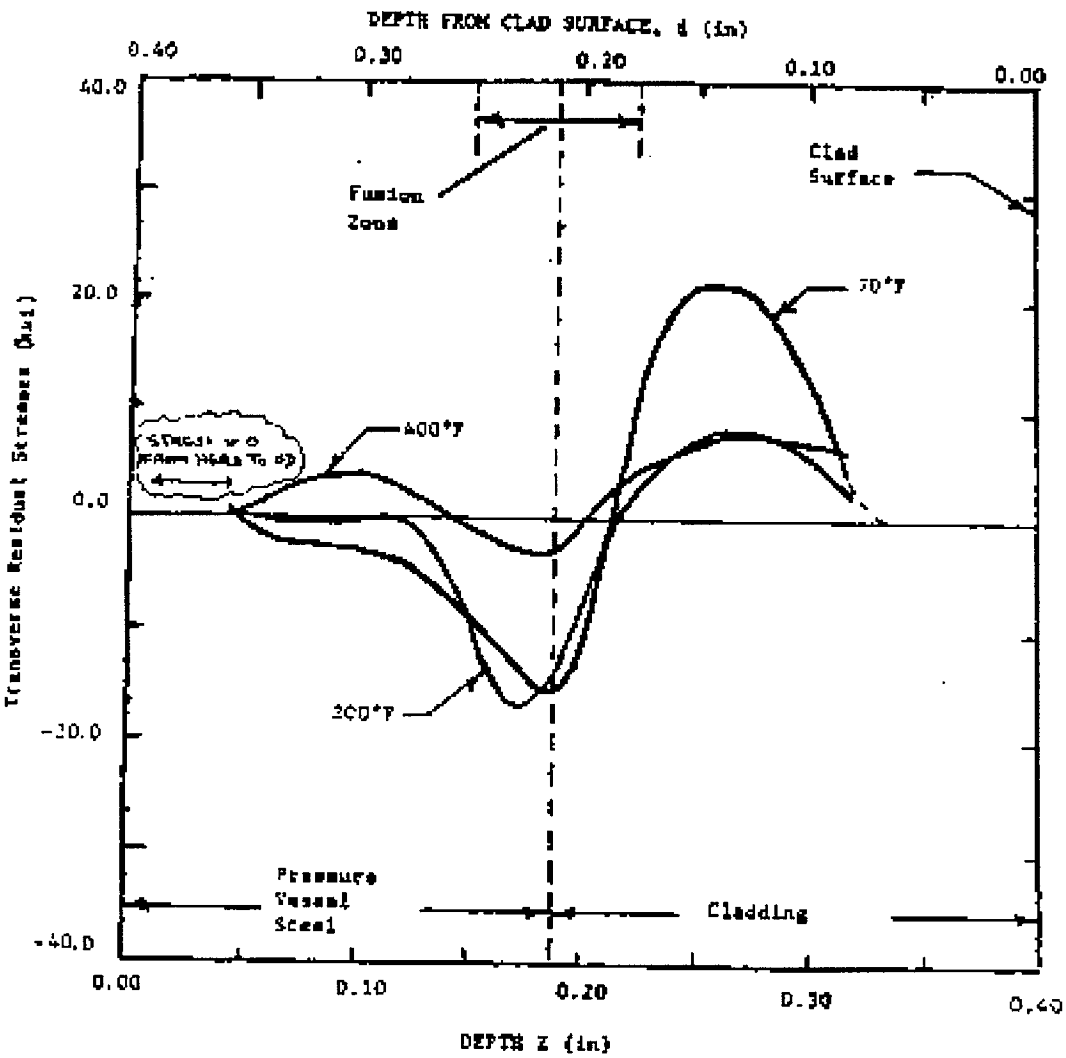

Figure A-2: Residual Stresses Transverse to Direction of Welding

SUMMARY OF PROPOSED LICENSE
AMENDMENT REQUEST

CHANGE TO THE PRESSURE- TEMEPERATURE CURVES AND DELETION OF SPECIAL TEST EXCEPTION

GE Nuclear Energy Reports

GE-NE-B13-02057-00-03

"Pressure-Temperature Curves for ComEd Dresden Unit 3"

and

GE-NE-B13-02057-00-04

"Pressure-Temperature Curves for ComEd Dresden Unit 2"

(February 1999)

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 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
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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° 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.

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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.

As 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 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.

Overall Plant Safety is Improved

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.

Impact on PWRs:

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 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.

4. McCabe, D.E., "Assessment of Metallurgical Effects that Impact Pressure Vessel Safe Margin Issues", Oak Ridge Report ORNL/NRC/LTR-94/26, October 1994.

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.

(February 1999)

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

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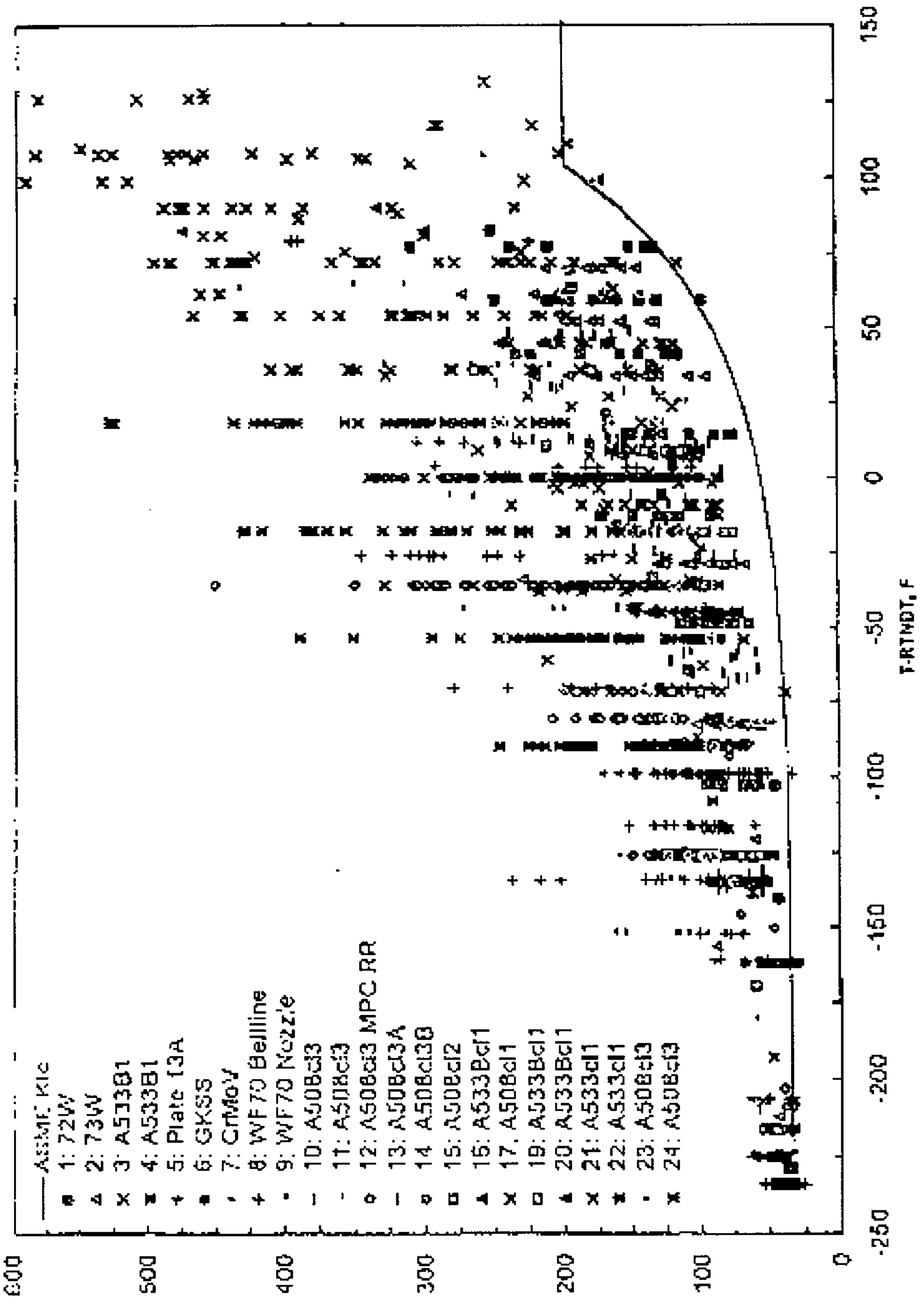

Toughness, ksi √in

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

(February 1999)

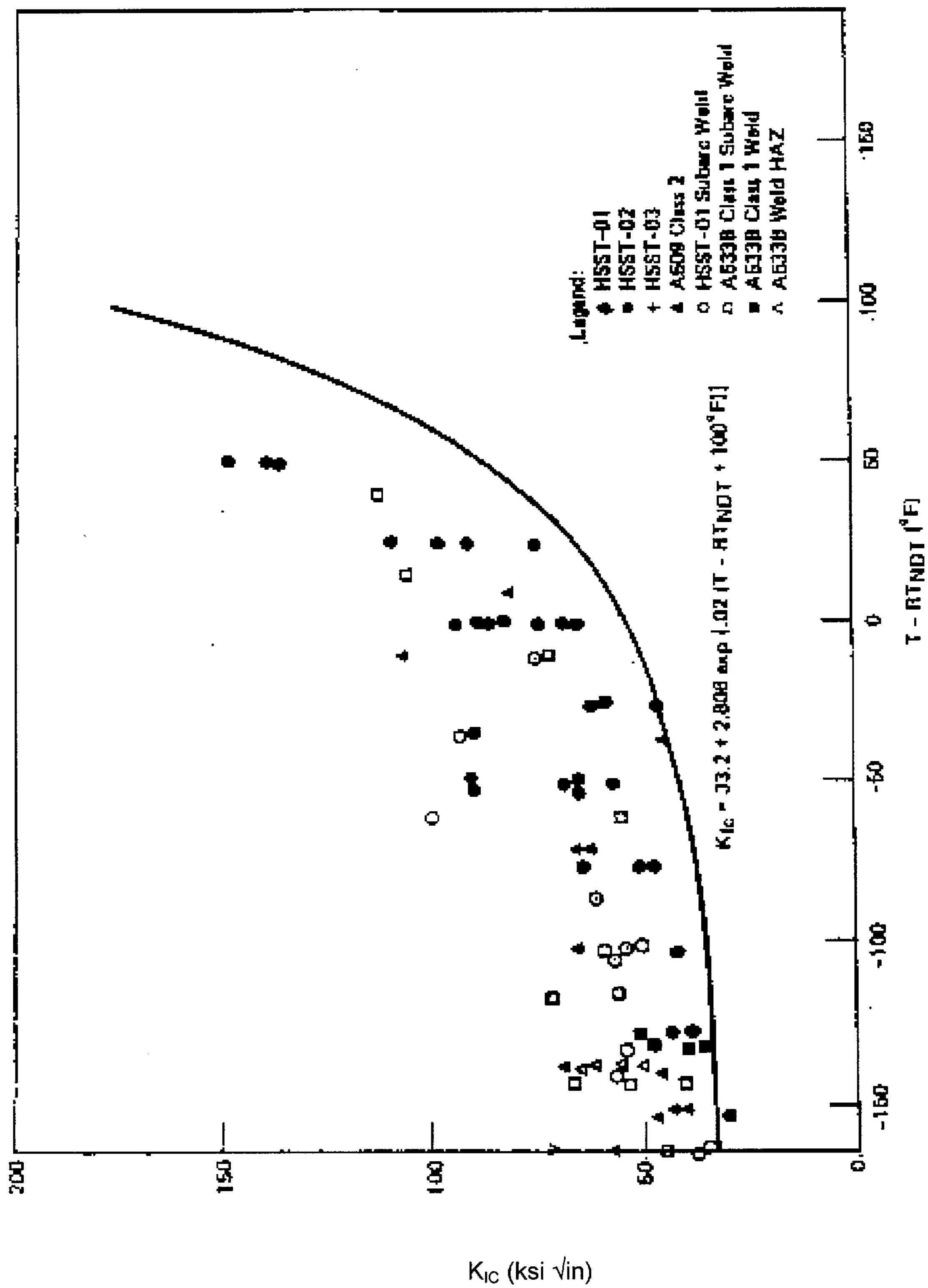

Figure 2. Original K_{IC} Reference Toughness Curve, with Supporting Data [2]

(February 1999)

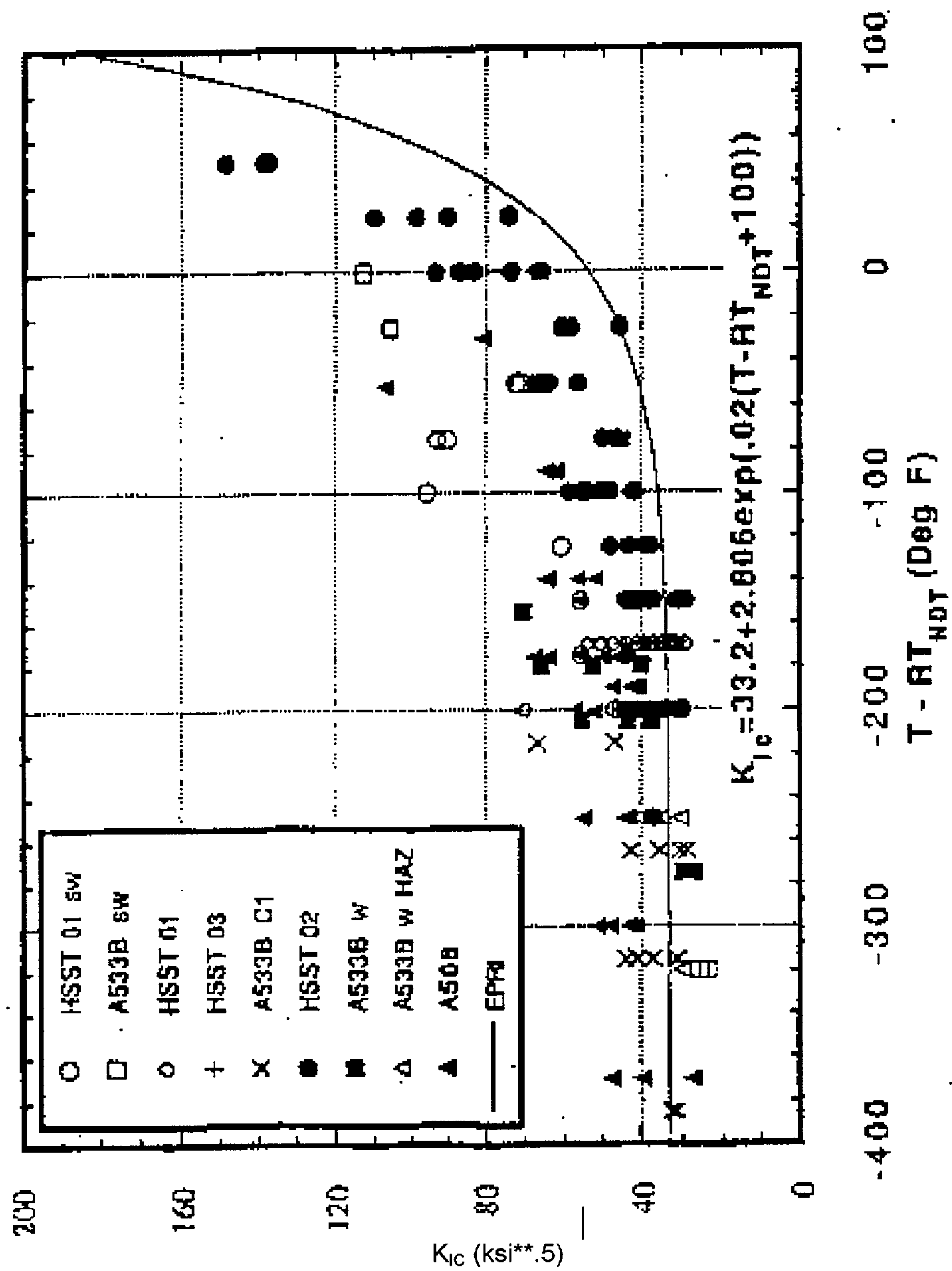

Figure 3. K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

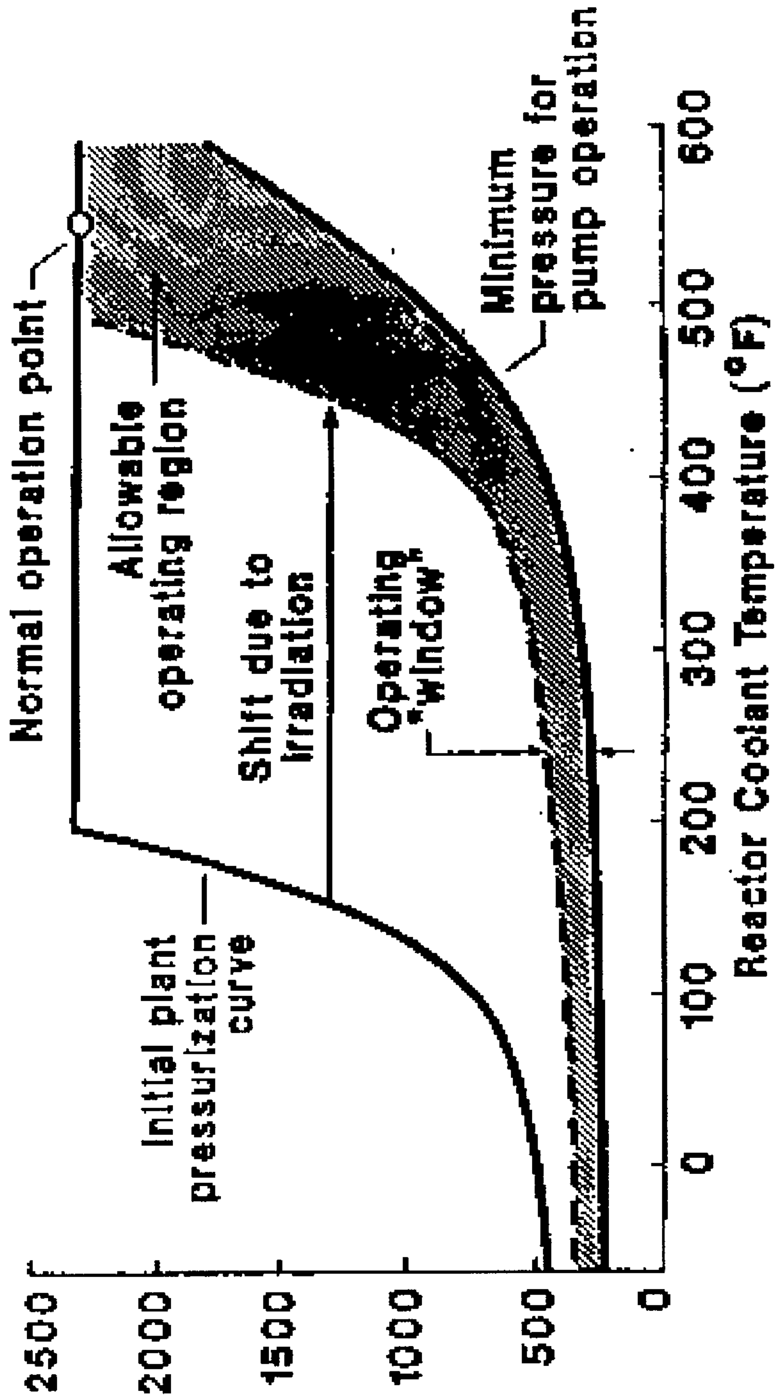

Figure 4. Operating Window From P-T Limit Curves [4]

(February 1999)

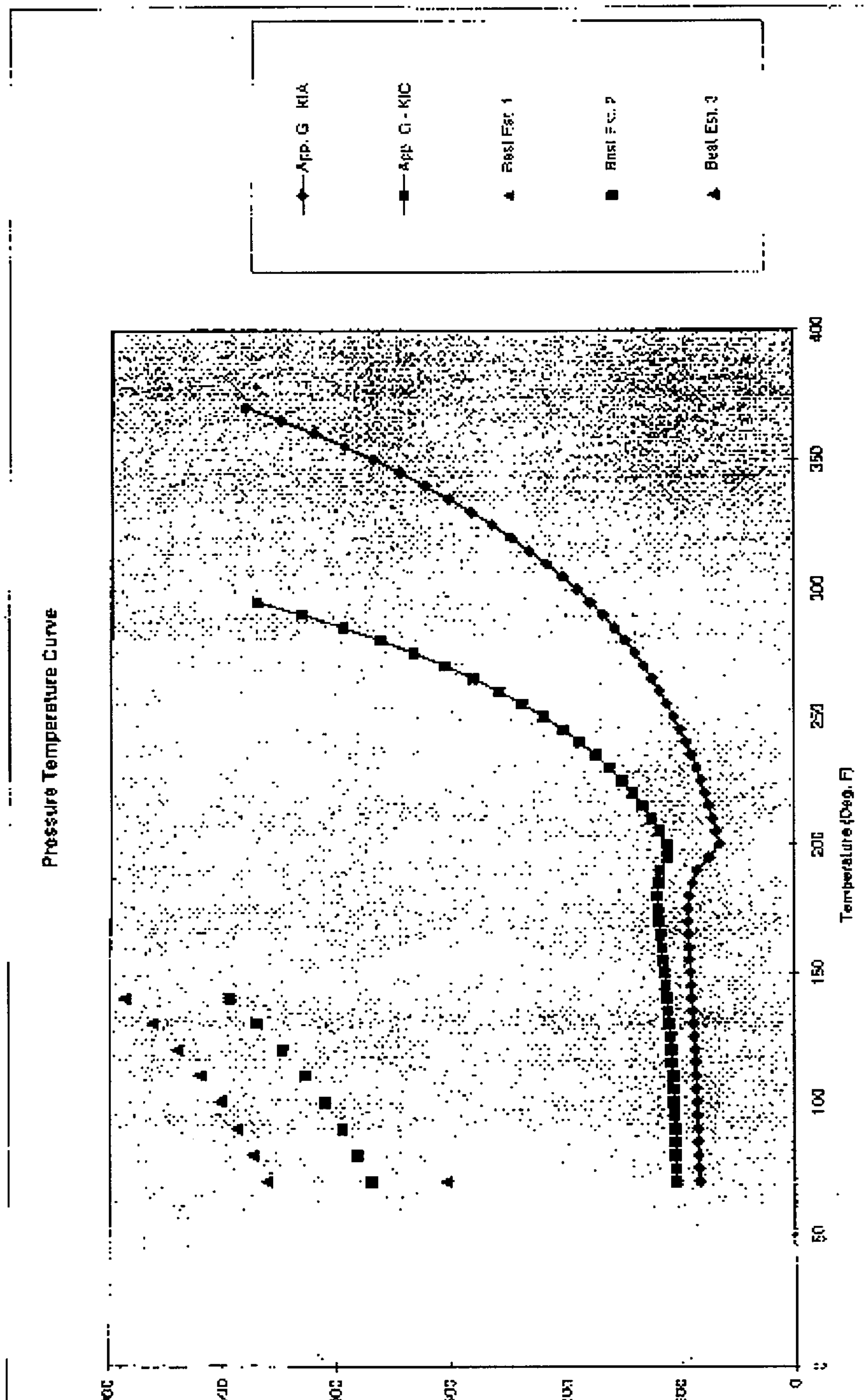

Figure 5. P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

(February 1999)

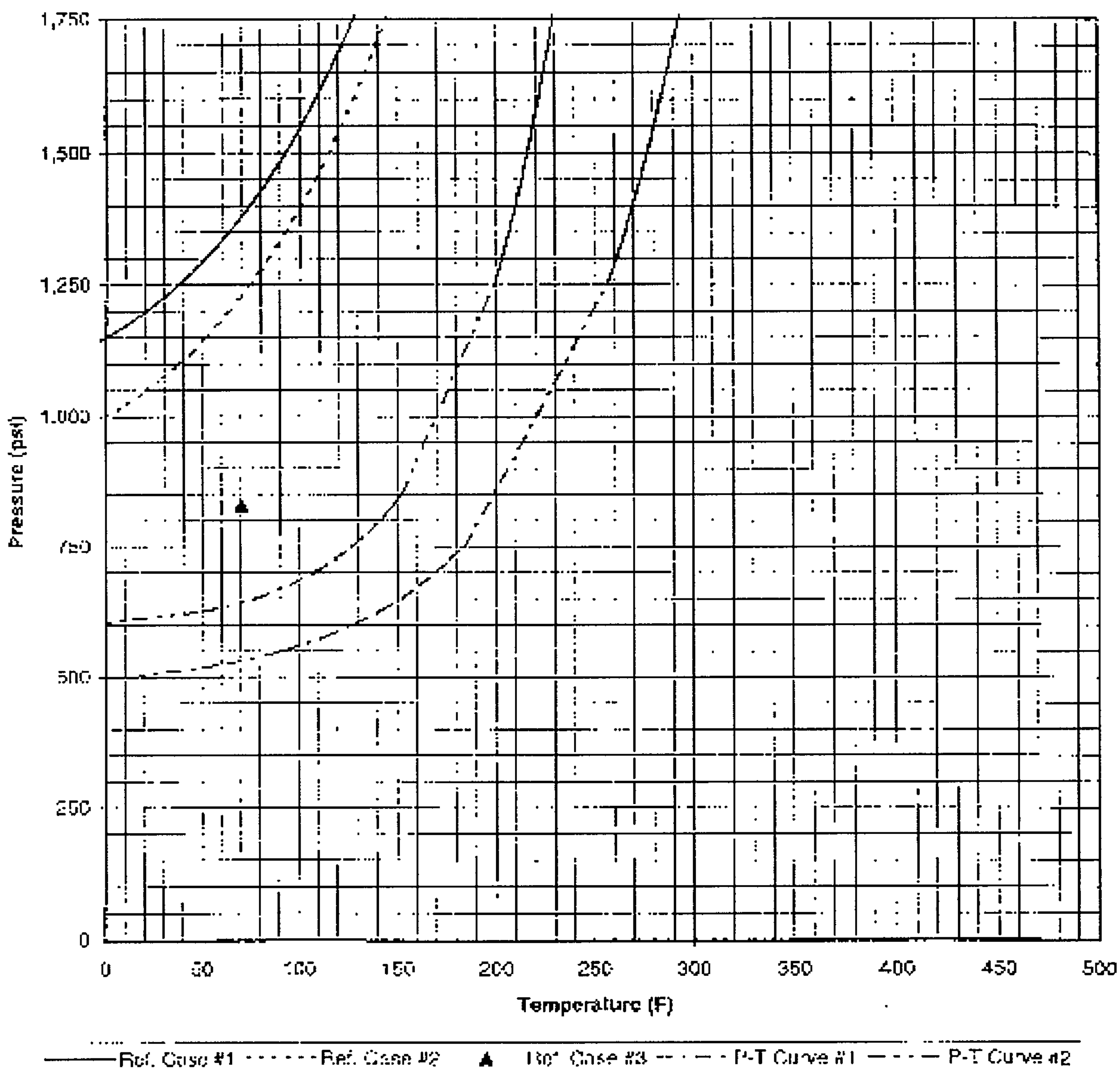

Figure 6. P-T Limit Curves Illustrating Deterministic Safety Factors
for a BWR Reactor Vessel

(February 1999)

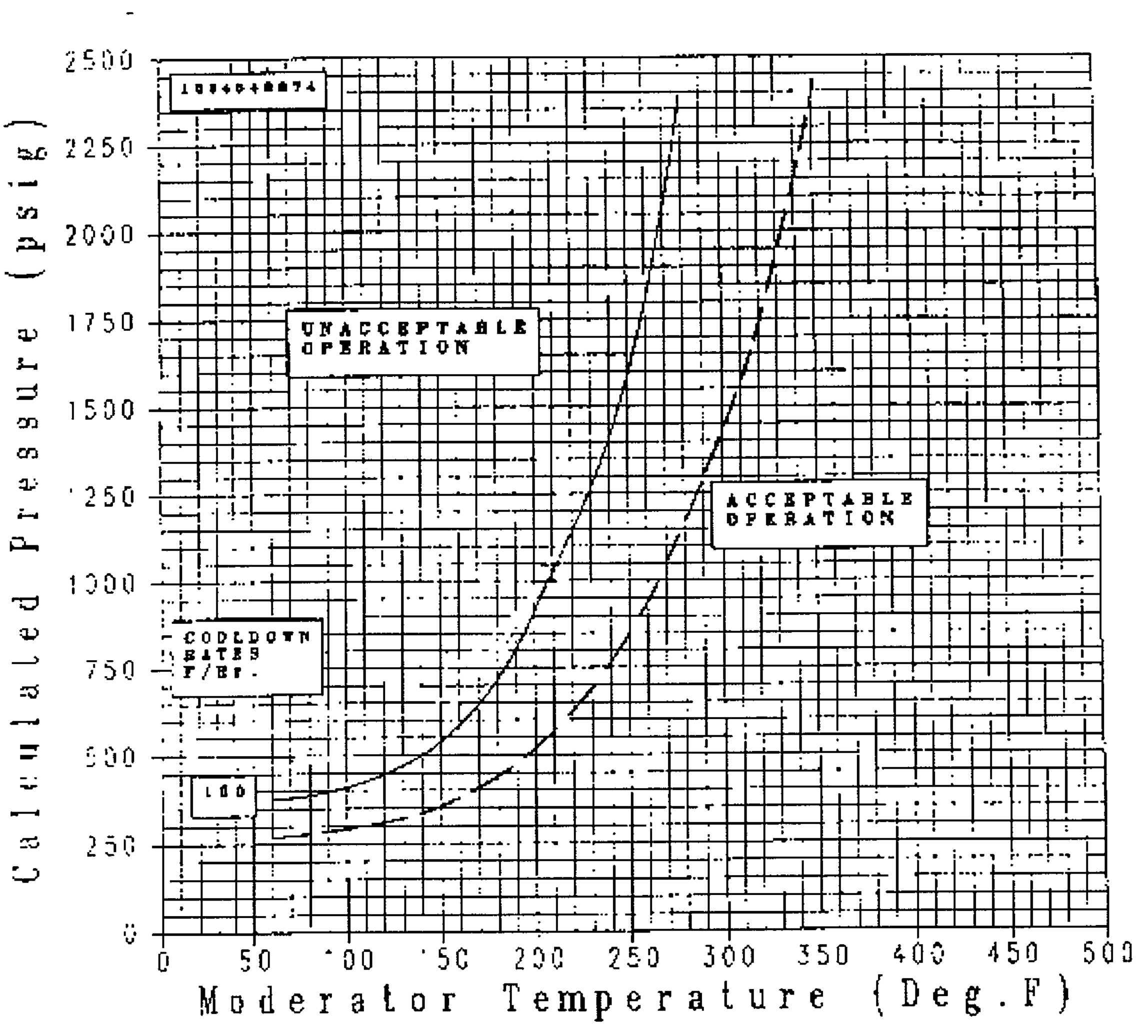

Figure 7. Comparison of Cool-Down Curves for the Existing and Proposed Methods - PWR [Dashed Curve = Existing (K_{Ia}) and Solid Curve = Proposed (K_{Ic})]

(February 1999)

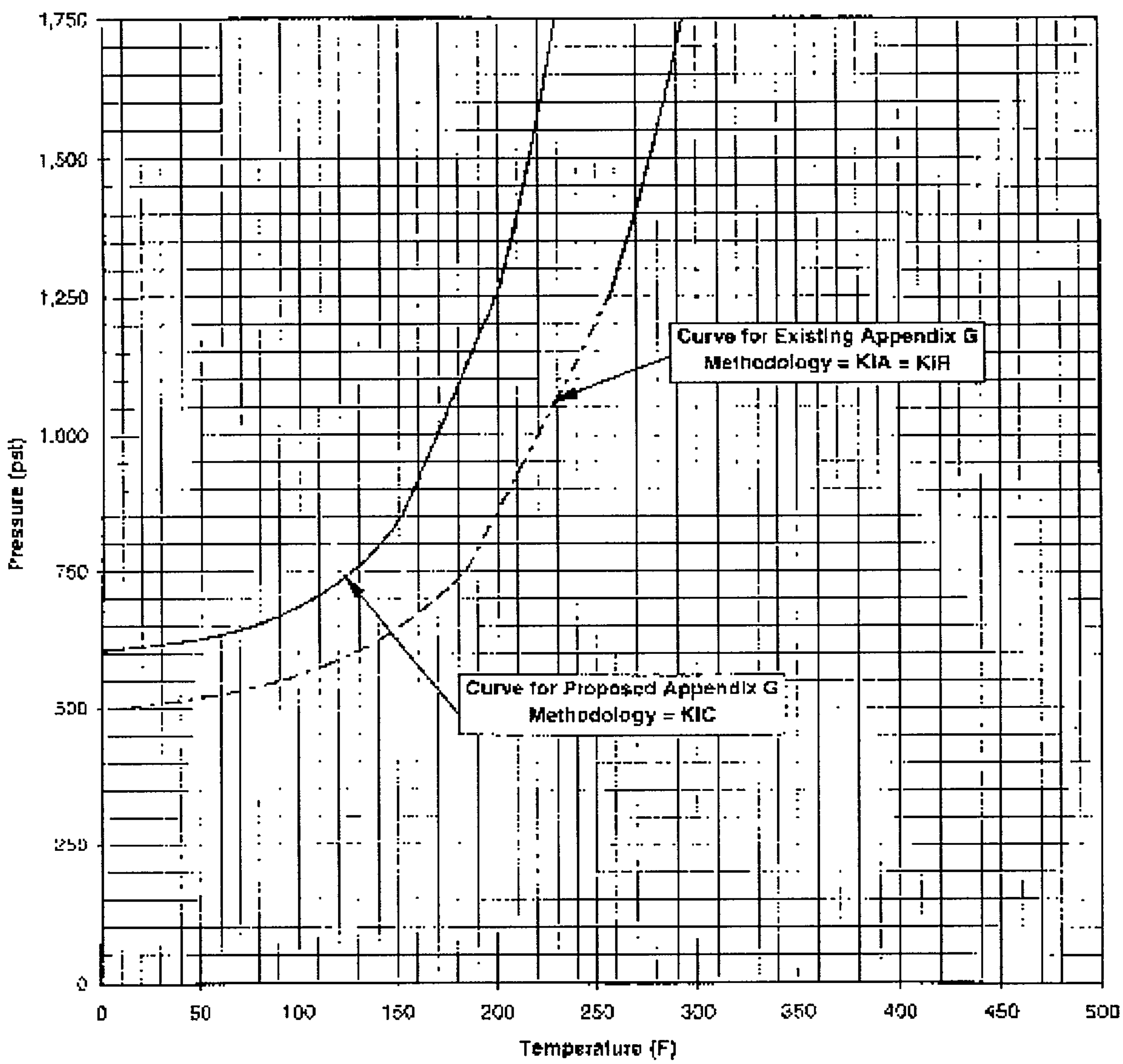

Figure 8. Comparison of Hydrotest P-T Curves for the Existing and Proposed Methods - BWR [Dashed Curve=Existing (K_{Ia}) and Solid Curve=Proposed (K_{Ic})]

(February 1999)

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.

(February 1999)

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

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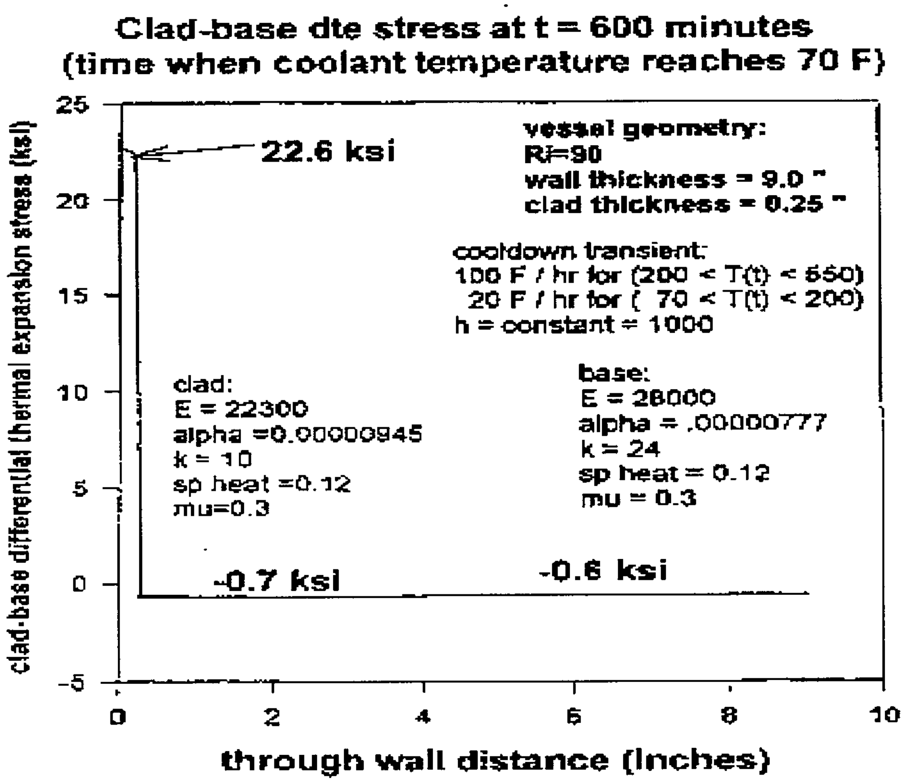

Figure A-1: Clad-base metal stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

(February 1999)

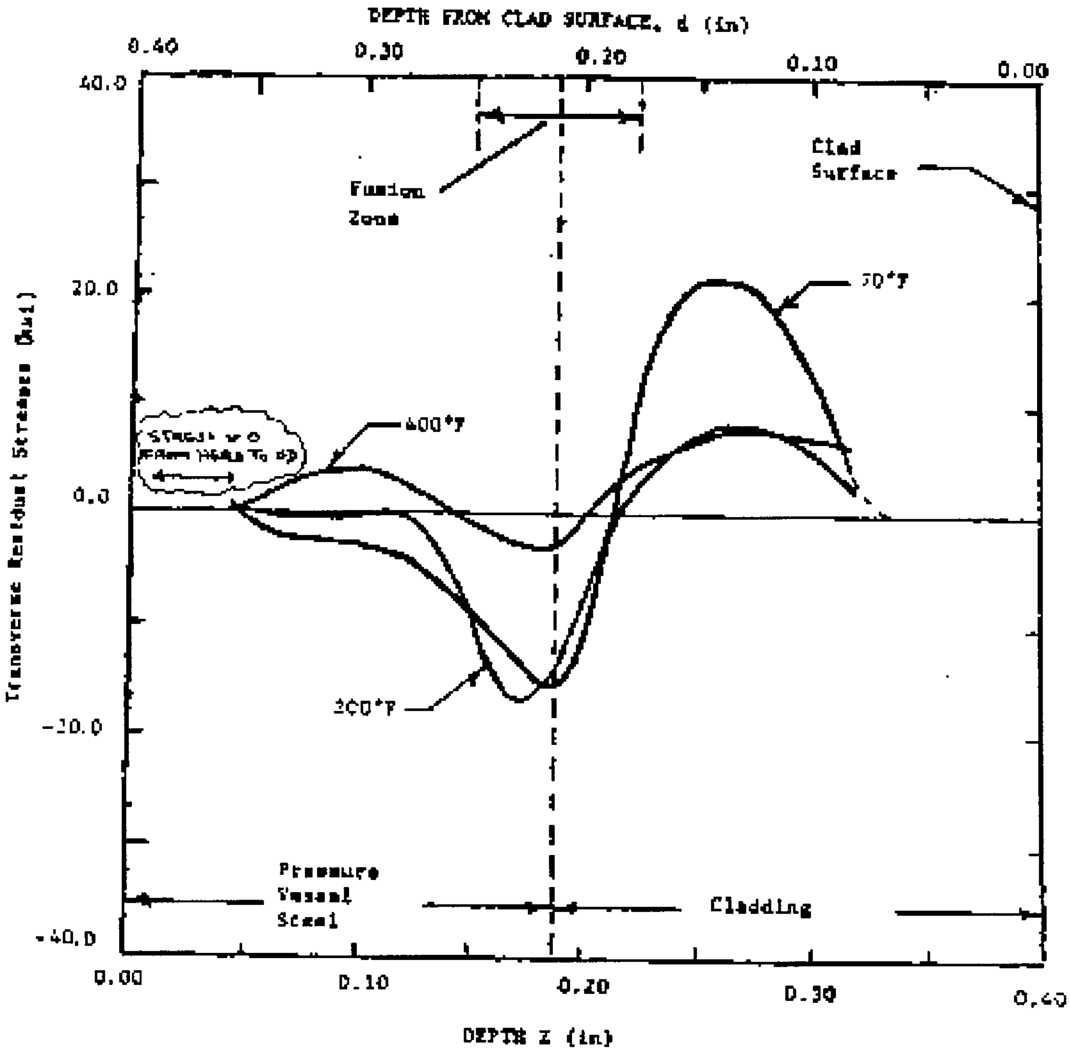

Figure A-2: Residual Stresses Transverse to Direction of Welding

ATTACHMENT G

GE Nuclear Energy Reports

GE-NE-B13-02057-00-03

"Pressure-Temperature Curves for ComEd Dresden Unit 3"

and

GE-NE-B13-02057-00-04

"Pressure-Temperature Curves for ComEd Dresden Unit 2"