Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

September 20, 1996

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

SUBJECT: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Request for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications (TS),
Changes to Pressure - Temperature (P-T) Curves
NRC Docket Nos. 50-237/249 and 50-254/265

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specification 3/4.6.K, of Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30. The purpose of this amendment request is to amend the P-T curves, updating them to 22 Effective Full Power Years (EFPYs), and to amend both the Technical Specification and its Bases.

The proposed Technical Specification Amendment is subdivided as follows:

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

2. Attachment B provides the proposed changes to the Technical Specification pages, including marked-up versions of the current pages.

3. Attachment C describes ComEd's evaluation performed in accordance with 10 CFR 50.92 (c), which confirms that no significant hazards consideration is involved. In addition, ComEd's Environmental Assessment Applicability Review is included.

4. Attachment D provides the technical report which developed the P-T Curves.

This proposed Technical Specification amendment has been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures.

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ComEd requests NRC approval of this request within five months of receipt of this submittal to be effective no later than 30 days following approval.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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

Please direct any questions you may have concerning this submittal to this office.

Sincerely,

Peter L. Piet

Peter L. Piet
Nuclear Licensing Administrator

Subscribed and Sworn to before me
on this 20th day of
September, 1996.
Betty Fox
Notary Public

OFFICIAL SEAL
BETTY FOX
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:08/28/00

Attachments:

A. Description and Safety Analysis of the Proposed Changes
B. Affected Technical Specification Pages
C. Evaluation of Significant Hazards Considerations and Environmental Assessment Applicability Review
D. GE Report "Pressure-Temperature Curves for Dresden and Quad Cities Stations" GE-NE-B11-00707-01R1 - July 1996

cc: A.B. Beach, Regional Administrator - RIII
C.G. Miller, Senior Resident Inspector - Quad Cities
C. L. Vanderniet, Senior Resident Inspector - Dresden
R. M. Pulsifer, Project Manager - NRR
J. F. Stang, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

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PROPOSED CHANGE TO TECH SPECS RE CHANGES TO PRESSURE - TEMPERATURE (P-T) CURVES

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ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

ATTACHMENT A
DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Description of the Proposed Change

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specifications 3/4.6.K "P-T Limits" of Facility Operating Licenses, DPR-19, DPR-25, DPR-29 and DPR-30. Corresponding changes to the TS Bases are also proposed. The proposed changes for 3/4.6.K are consistent with the requirements of Section 3/4.4.6 of the Improved Standard Technical Specifications (NUREG-1433).

The proposed changes are consistent with the criteria specified by the NRC in the "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," (58 FR 39132), which determines the design conditions and associated surveillances that should be located in the Technical Specification limiting conditions for operation.

The proposed changes include reformatting of the text for Specification 3.6.K. This consists primarily of expansion of the Limiting Condition for Operation (LCO) from a single, 3-part paragraph into 3 separate paragraphs corresponding to the evolutions which utilize separate primary coolant system temperature and pressure limits curves. This ensures clarity for the use of the curves, which have different limits for the rate of change limits for system temperatures.

In addition to the reformatting, an additional restriction is implemented in the ACTION statement for Specification 3.6. This change specifies the time limitation for completion of engineering evaluations required when the LCO requirements are not met. Because the existing Specification does not prescribe a time limitation for this evaluation, a time limit was implemented to ensure clarity of action for the plant staff. The value used (72 hours) is the same as used in the Improved Technical Specification, and is consistent with the verification of Operability for similar issues.

Description and Bases of the Current Operating License/Technical Specification Requirement

Pressure and Temperature limits are imposed on the reactor coolant system as shown on Figure 3.6.K-1. Curve A deals with system hydrostatic or leak testing and shows irradiation effects for 12,14 and 16 effective full power years (EFPY); Curve B with non-nuclear heatup/cooldown limit; and Curve C with nuclear limits. Both Curves B and C are valid to 16 EFPY. These curves meet the method used to account for irradiation embrittlement as described in Regulatory Guide 1.99, Revision 2.

Description of the Need and Bases for Amending the Technical Specifications

Background

The new pressure-temperature (P-T) curves prepared by General Electric have been developed to present steam dome pressure versus minimum vessel metal temperature criteria incorporating irradiation embrittlement effects in the beltline and appropriate non-beltline limits. These P-T curves were developed for 22 EFPY. The methodology used to generate the P-T curves in this report is an NRC approved methodology.

The pressure-temperature (P-T) curves are established to the requirements of 10CFR50, Appendix G to assure that brittle fracture of the reactor vessel is prevented. Part of the analysis involved in developing the P-T curves is to account for irradiation embrittlement effects in the core region, or beltline. The method used to account for irradiation embrittlement is described in Regulatory Guide 1.99, Revision 2.

In addition to beltline considerations, there are non-beltline discontinuity limits such as nozzles, penetrations, and flanges which affect the P-T curves. The non-beltline limits are based on generic analyses which are adjusted to the maximum reference temperature of nil ductility transition (RT_{NDT}) for the applicable Dresden or Quad Cities vessel components. The non-beltline limits are also governed by requirements in the existing technical specifications.

Furthermore, curves are included to allow monitoring of the non-beltline regions of the vessel separate from the beltline region. This refinement could minimize heating requirements prior to pressure testing.

GENERATION OF NEW CURVES

Operating Limits for pressure and temperature are required for three categories of operation: (a) pressure tests; (b) non-nuclear heatup/cooldown and low-level physics tests; and (c) core critical operation limits. The new limits focused in four areas - the closure flange region, the core beltline region, and the remainder of the vessel, or non-beltline regions with the exclusion of the bottom head region, and finally the bottom head region itself. The closure flange region limits are controlling at lower pressures primarily because of 10CFR50 Appendix G requirements. The non-beltline and beltline region operating limits are evaluated according to procedures in 10CFR50 Appendix G, ASME Code Appendix G and Welding Research Council (WRC) Bulletin 175 with the beltline region minimum temperature limits adjusted to account for vessel irradiation.

The material chemical and mechanical properties utilized for all regions of the vessels incorporate the most recent best-estimate values for welds available. Where appropriate, the methodology of GENE Report for BWR Owners Group NEDC - 32399-P, "Basis for GE RT_{NDT} Estimation Method," September 1994, has been applied to vessel materials in all regions in evaluating RT_{NDT}. As described in Attachment D (Attachment D reference [6]), the NRC has issued a SER approving the generic methodology used for this revision.

The revised curves result in portions of the new P-T limits being less restrictive than the previous curves. This is due to the revised limiting initial RT_{NDT} value of 23°F (which was previously 40°F for the existing Specification), determined by the electroslag weld (ESW) evaluation results. This is included in the Attachment D evaluation.

Figures 3.6.K-1 through 3.6.K-3 respectively represent the P-T limits for pressure tests for Effective Full Power Years (EFPYs) of 18, 20 and 22 years. Figure 3.6.K-4 represents the P-T limits for the non-nuclear and low-level physics tests for a heatup/cooldown rate of 100°F/hour. Figure 3.6.K-5 represents critical core operation limits.

The requirements for each vessel region influencing the P-T curves are discussed below.

Non-Beltline Regions

Non-beltline regions are those locations that receive too little fluence to cause any RT_{NDT} increase. Non-beltline components include the nozzles, the closure flanges, some shell plates, the top and bottom head plates and the control rod drive (CRD) penetrations. Detailed stress analyses, specifically for the purpose of fracture toughness analysis, of the non-beltline components were performed for the BWR/6. Plots were developed for the two most limiting BWR/6 components; the feedwater nozzle and the CRD penetration. All other components in the non-beltline regions are categorized under one of these two components.

The BWR/6 results have been applied to earlier BWR non-beltline vessel components based on the facts that earlier vessel component geometry's are not significantly different from BWR/6 configurations, and mechanical and thermal loading are comparable.

Under certain conditions, the minimum bottom head temperature can be significantly cooler than the beltline or closure flange region. These conditions can occur when the recirculation pumps are operating at low speed (or off), and during water injection through the control rod drives. To account for these circumstances, individual temperature limits for the bottom head were established.

For pressures below 20% of preservice hydrostatic test pressure (312 psig) and with full bolt preload, the closure flange region metal temperature is required to be at or greater than RT_{NDT}. The limiting flange region RT_{NDT} is 23.1°F. At low pressure, ASME Code Appendix G allows the beltline and bottom head regions to experience even lower metal temperatures than the flange region RT_{NDT}. However, these temperatures should not be reached as described below.

The shutdown margin for the Dresden and Quad Cities plants is calculated for a water temperature of 68°F. Shutdown margin is the quantity of reactivity needed for a reactor core to reach criticality with the strongest-worth control rod fully withdrawn and all other control rods fully inserted. Although it may be possible to safely allow water temperature to fall below this 68°F limit, extensive further calculations would be required to justify a lower temperature. Because the water temperature is currently limited to a minimum of 68°F, the metal temperature should not fall below this limit while fuel is in the vessel. (When fuel has been removed from the vessel, and the pressure is below 40 psi, the limiting vessel temperature is 60°F, the limiting RT_{NDT} of the vessel materials.)

ATTACHMENT A
DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Core Beltline Region

The pressure-temperature (P-T) limits for the beltline region are determined according to the methods in ASME Code Appendix G. As the beltline fluence increases during operation, these curves shift. These "Shift values" were used to adjust the RT_{NDT} values for the P-T limits. For the Dresden and Quad Cities vessels, the beltline curves were more limiting through 22 EFPY at typical operating pressures.

Closure Flange Region

10CFR50 Appendix G sets several minimum requirements for pressure and temperature, in addition to those outlined in the ASME Code, based on the closure flange region RT_{NDT}. In some cases, the results of analysis for other regions exceed these requirements and they do not affect the shape of the P-T curves. However, some closure flange requirements do impact the curves.

The approach used for Dresden and Quad Cities for the boltup temperature was based on a conservative value of (RT_{NDT} + 60°F). This conservatism is appropriate because boltup is one of the more limiting operating conditions (high stress and low temperature) for brittle fracture.

10CFR50 Appendix G sets minimum temperature requirements for pressure above 20% hydrotest pressure based on the RT_{NDT} of the closure region. The Pressure Test temperature must be no less than (RT_{NDT} + 90°F) and non-nuclear heatup/cooldown temperature no less than (RT_{NDT} + 120°F). The pressure test requirement causes a 30°F shift at 20% hydrotest pressure (312 psig) as shown in Figures 3.6.K-1 through 3.6.K-3. The heatup/cooldown (Figure 3.6.K-4) requirement has essentially no impact on the figure because the analytical results for the non-beltline regions exceed the temperature of (RT_{NDT} + 120°F) at 312 psig.

Core Critical Operation Requirements of 10CFR50, Appendix G

The core critical operation curve shown in Figure 3.6.K-5, is generated from the requirements of 10CFR50 Appendix G, Table 1. Essentially Table 1 requires that core critical P-T limits be 40°F above any pressure test or heatup/cooldown limits when pressure exceeds 20% of the preservice system hydrotest pressure. The heatup/cooldown curve (Figure 3.6.K-4) is more limiting than the appropriate pressure test curve, so limiting the core critical operation curve values must be at least the corresponding Figure 3.6.K-4 value plus 40°F whenever the pressure is above 312 psig.

10CFR50 Appendix G , Table 1 indicates that for BWR's with water level within normal range for power operation, the allowed initial criticality at the closure flange region is (RT_{NDT} + 60°F) at pressures below 312 psig. This requirement makes the minimum criticality temperature 83.1°F, based on a RT_{NDT} of 23.1°F. Above 312 psig, the temperature must be at least that required for the hydrostatic pressure test curve at 1120 psig.

The implementation of the calculation results utilized standard accuracy (rounding), which is reflected in final temperature limits which round the calculated RT_{NDT} of 23.1 °F to 23 °F. This rounded value is the final value utilized in the Attachment D (GE) report, and is standard industry practice, being well within the accuracy of temperature monitoring capability.

ATTACHMENT B

AFFECTED TECHNICAL SPECIFICATION PAGES

LICENSE DPR-19/25	LICENSE DPR-29/30
VIII	VIII
3/4.6-19	3/4.6-19
3/4.6-20	3/4.6-20
3/4.6-21	3/4.6-21
B 3/4.6-6	B 3/4.6-5
B 3/4.6-7	B 3/4.6-6
B 3/4.6-8	B 3/4.6-7

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.K-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

1. A maximum reactor coolant heatup of 100°F in any one hour period,
2. A maximum reactor coolant cooldown of 100°F in any one hour period,
3. A maximum reactor coolant temperature change of ≤20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
4. The reactor vessel flange and head flange temperature ≥100°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

INSERT 'A'

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the required heatup and cooldown limits and to the right of the limit lines of Figure 3.6.K-1 curves A, or B, as applicable, at least once per 30 minutes.
2. The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.6.K-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.
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.
4. The reactor vessel flange and head flange temperature shall be verified to be ≥100°F:
   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:
      1) ≤130°F, at least once per 12 hours.
      2) ≤110°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.

Insert A

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 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-4.

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-4, and

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

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-4.

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-5.

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.

INSERT A (CONT)

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-5, 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.

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.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION:

With any of the above limits exceeded,

~~1. Restore the temperature and/or pressure to within the limits within 30 minutes, and~~

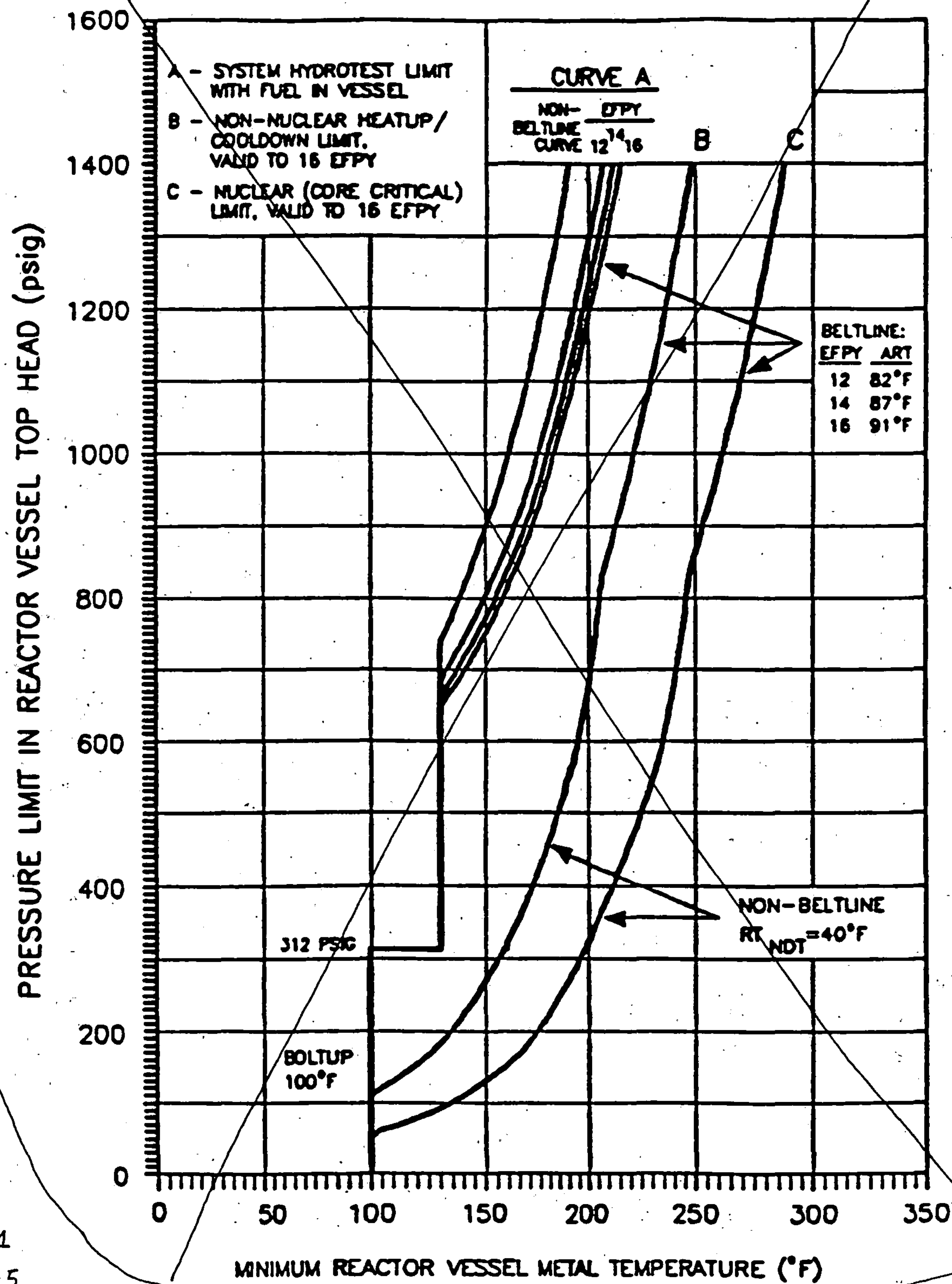
1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes without exceeding the applicable primary system coolant temperature rate of change limit, and

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.

FIGURE 3.6.K-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE



Replace with Figures 3.6.K-1 through 3.6.K-5

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.

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

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

Figures 3.6.K-1 through 3.6.K-3, are

~~The pressure-temperature~~ limit lines shown ~~in Figure 3.6.K-1~~, for operating conditions; ~~Inservice Hydrostatic~~ Pressure Testing ~~(curve A)~~, Non-Nuclear Heatup/Cooldown ~~(curve B)~~ and Core Critical Operation ~~(curve C)~~. 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.

Figure 3.6.K-5 Figure 3.6.K-4 and the bottom head region

Four ~~Three~~ 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~~ 3) the closure flange 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 ~~and~~ closure flange 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 region ~~is~~ non-beltline region, ~~it is~~ treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

(5) are (2) they are and bottom head

~~In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.~~

and 4) the bottom head region

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Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, the and shell and head materials is 23°F connecting to these flanges, ~~and~~ connecting welds ~~is 10°F; however,~~ the vertical electroslag welds which terminate immediately below the vessel flange ~~have an RT_{NDT} of 40°F~~. Therefore, the minimum allowable boltup temperature is established as ~~100°F~~ 83°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

~~Curve A - Hydrotesting~~

Insert B

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

~~Curve B~~ Figure 3.6.K-4 - Non-Nuclear Heatup/Cooldown

~~Curve B of~~ Figure 3.6.K-~~1~~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. As indicated by the vertical 100°F line, the boltup stresses at the ~~closure flange~~ region ~~are most~~ limiting for reactor pressures below ~~approximately 110~~ psig. For reactor pressures greater than ~~approximately 110 psig,~~ ~~pressurization~~ and thermal stresses become more limiting than the ~~boltup stresses~~, which is ~~reflected by the~~ nonlinear portion of curve B. The ~~non-linear~~ portion of the curve is dependent on non-beltline and ~~beltline regions~~, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a

The maximum heatup/cooldown rate of 100°F/hour is applicable.

INSERT B

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

As indicated in Figure 3.3.6.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.

Figures 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.

Figure 3.6.K-5

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vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

pressure testing or non-nuclear heatup / cooldown

~~Curve C~~ - Core Critical Operation

~~Curve C,~~ T~~t~~he core critical operation curve shown in Figure 3.6.K-~~4~~ 5, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any ~~curve A or B~~ Figure 3.6.K-4 limits. Since ~~curve B~~ Figure 3.6.K-4 is more limiting, ~~(curve C)~~ Figure 3.6.K-5 is ~~curve B~~ 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-~~73~~ 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 ~~can be used with confidence~~ are used in predicting reactor vessel material ~~transition temperature shift~~ embrittlement. The operating limit curves of Figure 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.

3/4.6.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 valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.K-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

1. A maximum reactor coolant heatup of 100°F in any one hour period,
2. A maximum reactor coolant cooldown of 100°F in any one hour period,
3. A maximum reactor coolant temperature change of ≤20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
4. The reactor vessel flange and head flange temperature ≥100°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

INSERT 'A'

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the required heatup and cooldown limits and to the right of the limit lines of Figure 3.6.K-1 curves A, or B, as applicable, at least once per 30 minutes.
2. The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.6.K-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.
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.
4. The reactor vessel flange and head flange temperature shall be verified to be ≥100°F:
   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:
      1) ≤130°F, at least once per 12 hours.
      2) ≤110°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.

Insert A

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 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-4.

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-4, and

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

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-4.

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-5.

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.

INSERT A (CONT)

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-5, 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.

<u>APPLICABILITY:</u>

At all times.

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.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION:

With any of the above limits exceeded,

1. ~~Restore the temperature and/or pressure to within the limits within 30 minutes, 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

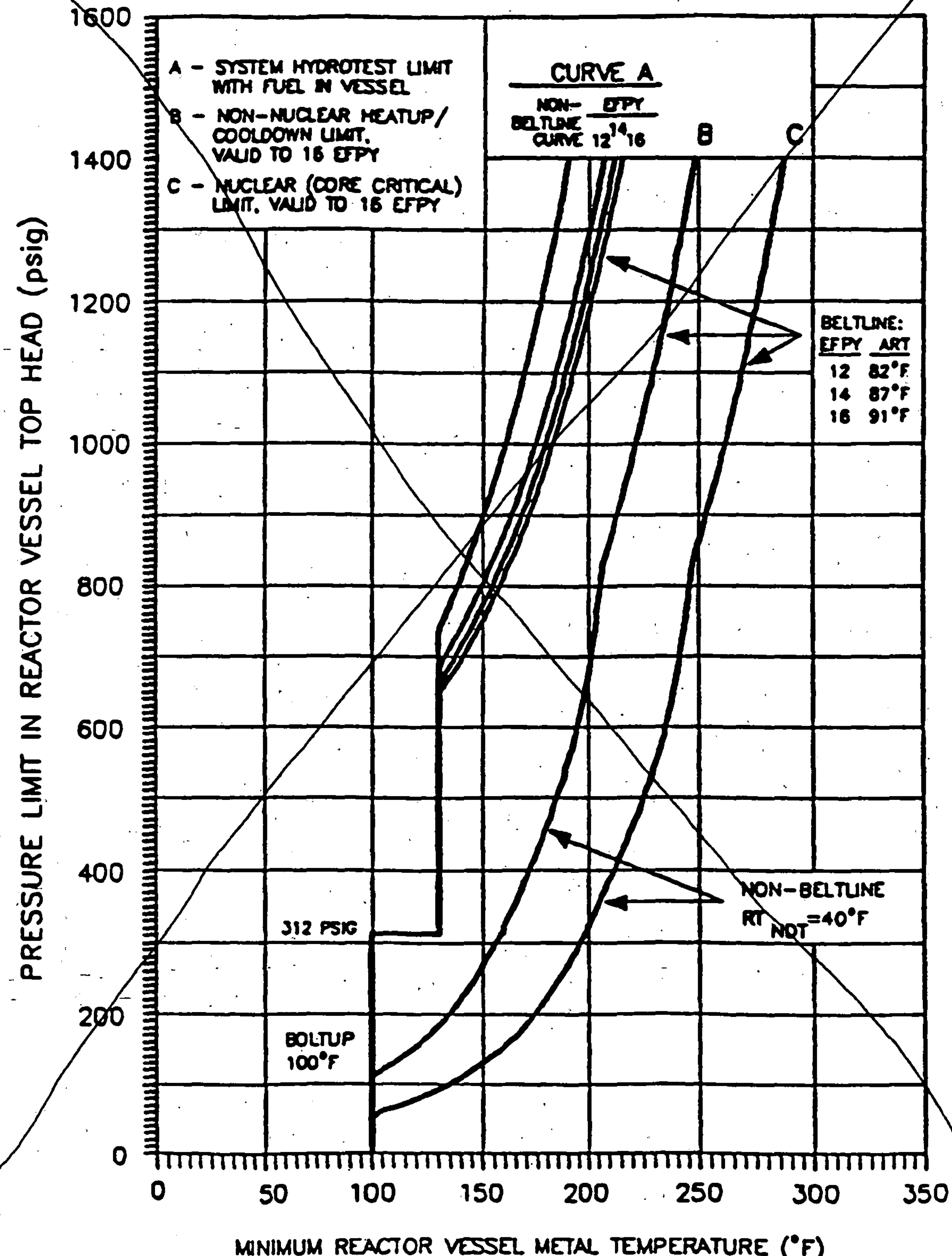
3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

Replace with Figures 3.6.K-1 through 3.6.K-5

FIGURE 3.6.K-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE



6

BASES

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1.1 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

~~The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for~~

BASES

the heatup ~~of the inner wall~~ cannot be defined. Subsequently, for ~~the cases in which~~ the outer wall of the vessel becomes the ~~stress controlling location, each heatup~~ rate of interest must be ~~analyzed on an individual basis.~~

Figures 3.6.K-1 through 3.6.K-3 are

The pressure-temperature limit lines shown ~~in Figure 3.6.K-1~~, for operating conditions; ~~Inservice Hydrostatic~~ Pressure Testing ~~(curve A)~~, Non-Nuclear Heatup/Cooldown ~~(curve B)~~, and Core Critical Operation ~~(curve C)~~. 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.

Figure 3.6.K-5 , Figure 3.6.K-4 and the bottom head region

~~Three~~ 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~~ 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, ~~and~~ closure flange, region(s) receive(s) 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 region(s) ~~is a~~ non-beltline region(s), ~~it is~~ they are treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

and bottom head

~~In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.~~

Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, and the shell and head materials connecting to these flanges, ~~and~~ connecting welds ~~is 10°F; however,~~ the vertical electroslag welds which terminate immediately below the vessel flange ~~have an RT_{NDT} of 40°F~~ is 23°F. Therefore, the minimum allowable boltup temperature is established as ~~100°F~~ 83°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

~~Curve A - Hydrotesting~~

~~As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G~~

Insert B

INSERT B

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

As indicated in Figure 3.3.6.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.

Figures 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.

~~for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.~~

~~A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stablized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.~~

~~Curve B~~ Figure 3.6.K-4 - Non-Nuclear Heatup/Cooldown

Pressure testing / non-nuclear heatup/cooldown

~~Curve B of~~ Figure 3.6.K-~~1~~ 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. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

~~Curve C~~ Figure 3.6.K-5 - Core Critical Operation

~~Curve C,~~ The core critical operation curve shown in Figure 3.6.K-~~1~~ 5 is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any ~~curve A or B~~ Figure 3.6.K-4 limits. Since ~~curve B~~ Figure 3.6.K-4 is more limiting, ~~(curve C~~ Figure 3.6.K-5 is ~~curve B~~ Figure 3.6.K-4 plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-~~73~~ 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 ~~can be used with confidence~~ are used in predicting reactor vessel material ~~transition temperature shift.~~ embrittlement. The operating limit curves of Figure 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.

The maximum heatup/cooldown rate of 100°F/hour is applicable.

ATTACHMENT B (Cont'd)

PROPOSED TECHNICAL SPECIFICATION PAGES

ATTACHMENT C
EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT APPLICABILITY REVIEW

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

ComEd proposes to amend Appendix A, Technical Specifications, Section 3/4.6.K of Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30. The amendment request changes the pressure temperature (P-T) curves, Figure 3.6.K-1 and associated Bases.

ComEd has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of Dresden Units 2 and 3 or Quad Cities Units 1 and 2 in accordance with the proposed amendment will not:

1) **Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:**

   The proposed changes merely adjust the reference temperature for the limiting beltline material to account for irradiation effects and provide the same level of protection as previously evaluated. The adjusted reference temperature calculations were performed utilizing the guidance contained in Regulatory Guide 1.99, Revision 2. The change is administrative in nature to reflect the extension of the operating limits to 22 EFPY. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

2) **Create the possibility of a new or different kind of accident from any accident previously evaluated because:**

   The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. The revised operating limits are merely an updated of the old limits by taking into account the effects of irradiation on the limiting reactor vessel material. Use of the revised P-T curves will continue to provide the same level of protection as was previously reviewed and approved. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

   The associated change to the P-T curves related to this proposed amendment does not affect any activities or equipment and are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3) Involve a significant reduction in the margin of safety because:

The proposed amendment reflect an update of the P-T curves to extend the operating limit to 22 EFPY. The revised curves are based on the latest NRC guidance along with actual data for the units. The new limits retain the margin of safety to the level expected for a new vessel, adjusted for irradiation effects as required by 10CFR, Appendix G, thereby maintaining a conservative margin of safety.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

ComEd has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT B

AFFECTED TECHNICAL SPECIFICATION PAGES

LICENSE DPR-19/25		LICENSE DPR-29/30	
Remove	Insert	Remove	Insert
VIII	VIII	VIII	VIII
3/4.6-19	3/4.6-19	3/4.6-19	3/4.6-19
3/4.6-20	3/4.6-20	3/4.6-20	3/4.6-20
3/4.6-21	3/4.6-21a	3/4.6-21	3/4.6-21a
--	3/4.6-21b	--	3/4.6-21b
--	3/4.6-21c	--	3/4.6-21c
--	3/4.6-21d	--	3/4.6-21d
--	3/4.6-21e	--	3/4.6-21e
B 3/4.6-6	B 3/4.6-6	B 3/4.6-5	B 3/4.6-5
B 3/4.6-7	B 3/4.6-7	B 3/4.6-6	B 3/4.6-6
B 3/4.6-8	B 3/4.6-8	B 3/4.6-7	B 3/4.6-7
B 3/4.6-9	--		

SECTION PAGE

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 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-4.

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-4, and

   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-4.

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-5.

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-5, 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

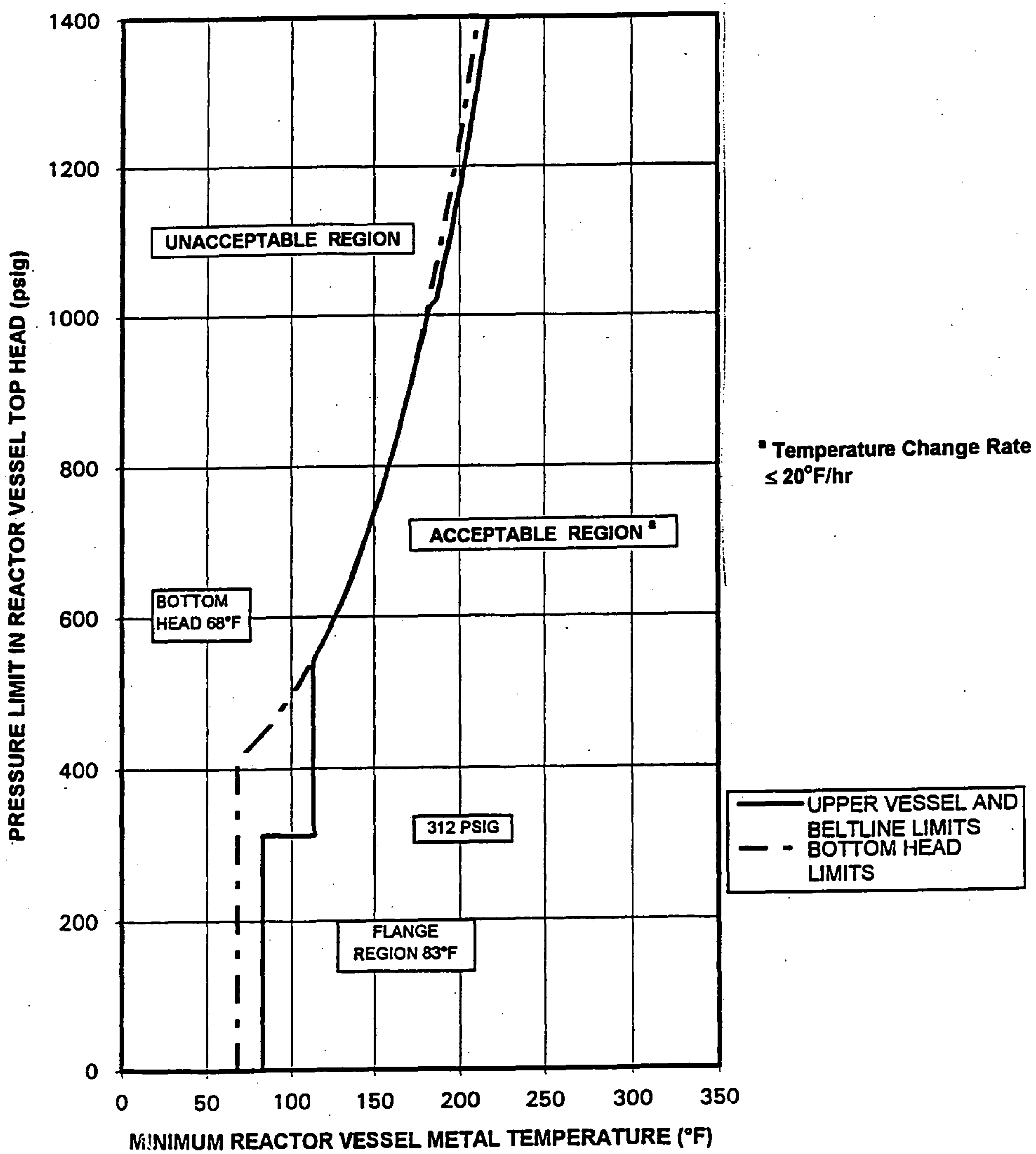


FIGURE 3.6.K-2

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

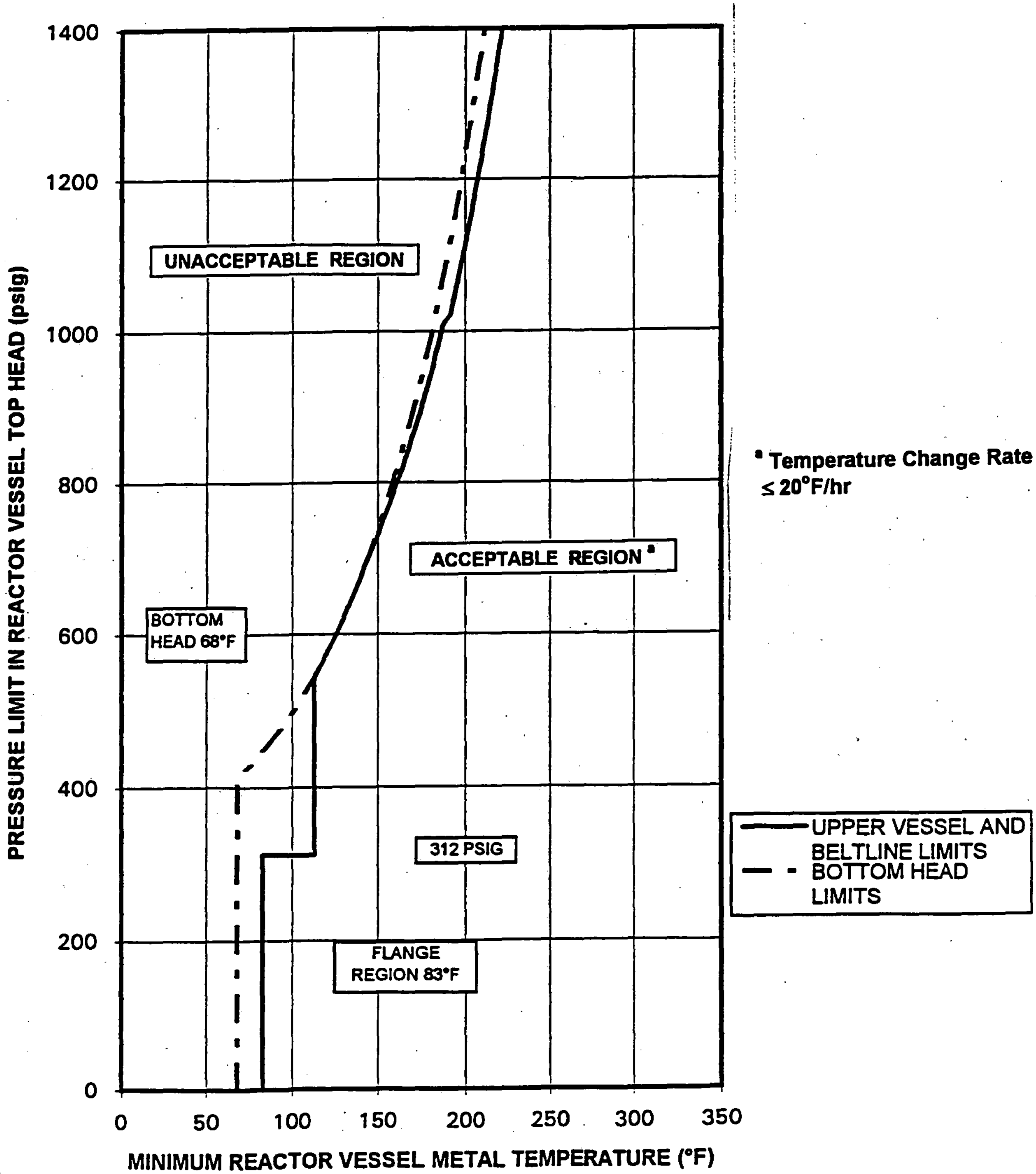
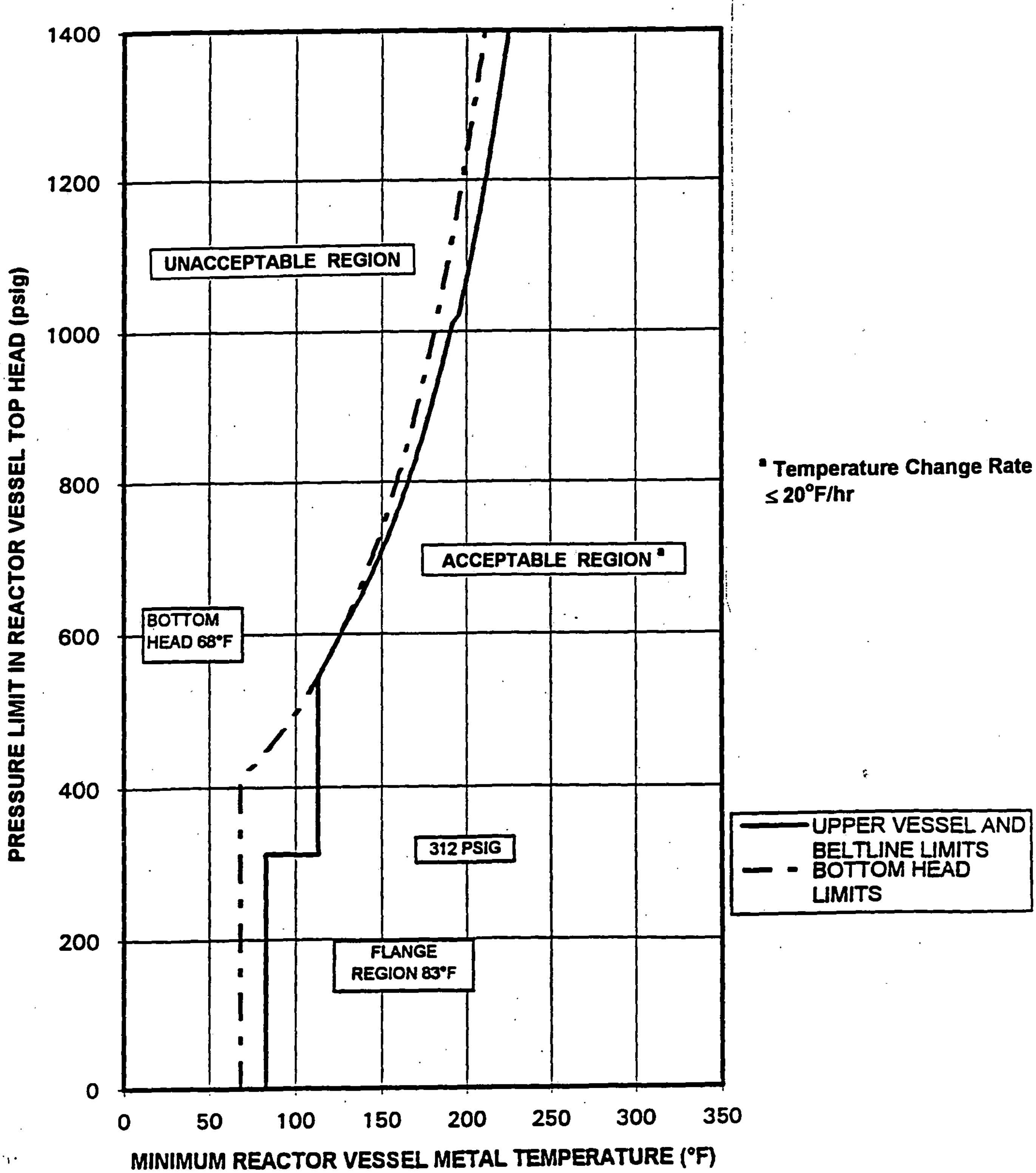


FIGURE 3.6.K-3

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



[a] Temperature Change Rate ≤ 20°F/hr

FIGURE 3.6.K-4

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

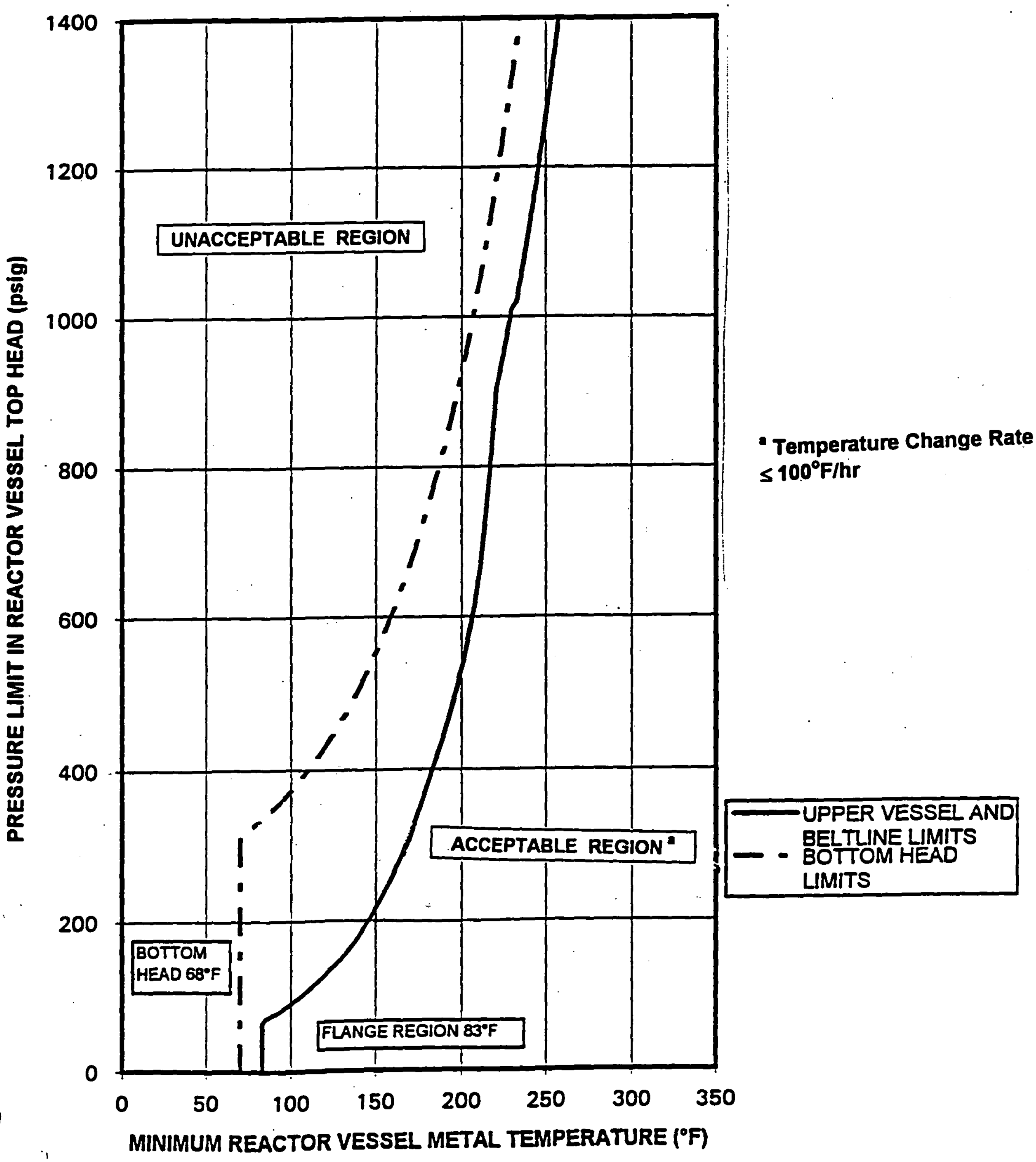
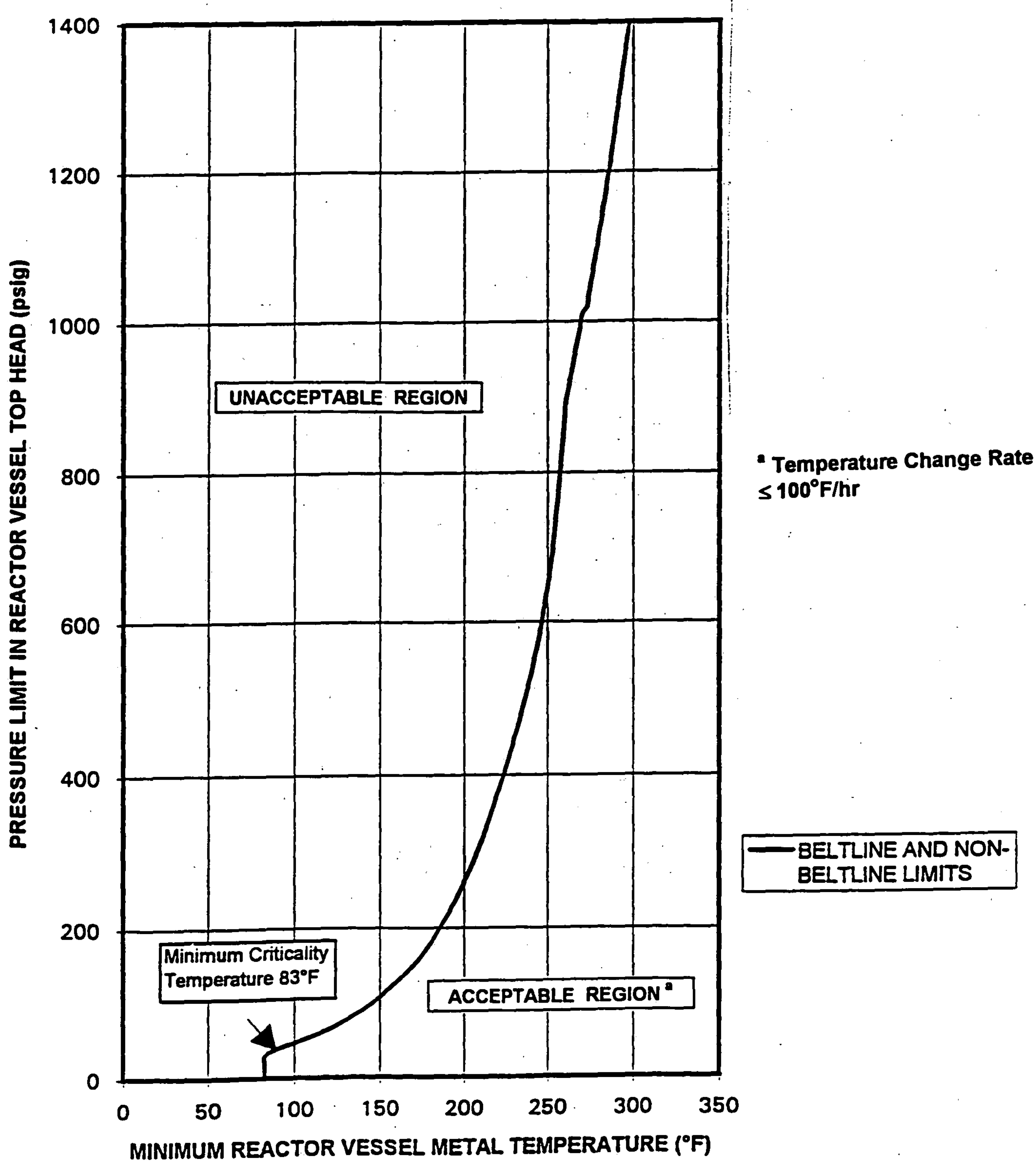


FIGURE 3.6.K-5

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.

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.

BASES

Figures 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.

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

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.

Figure 3.6.K-5 - Core Critical Operation

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 ASTI 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.

3/4.6.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 valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not

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uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

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

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

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

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

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 3.6.K-1 through 3.6.K-3 with the rate of change of the primary system coolant temperature ≤ 20°F per hour.

    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-4.

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-4, and

    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-4.

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-5

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.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 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-4.

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-4, and

    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-4.

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-5.

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-5, 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. 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

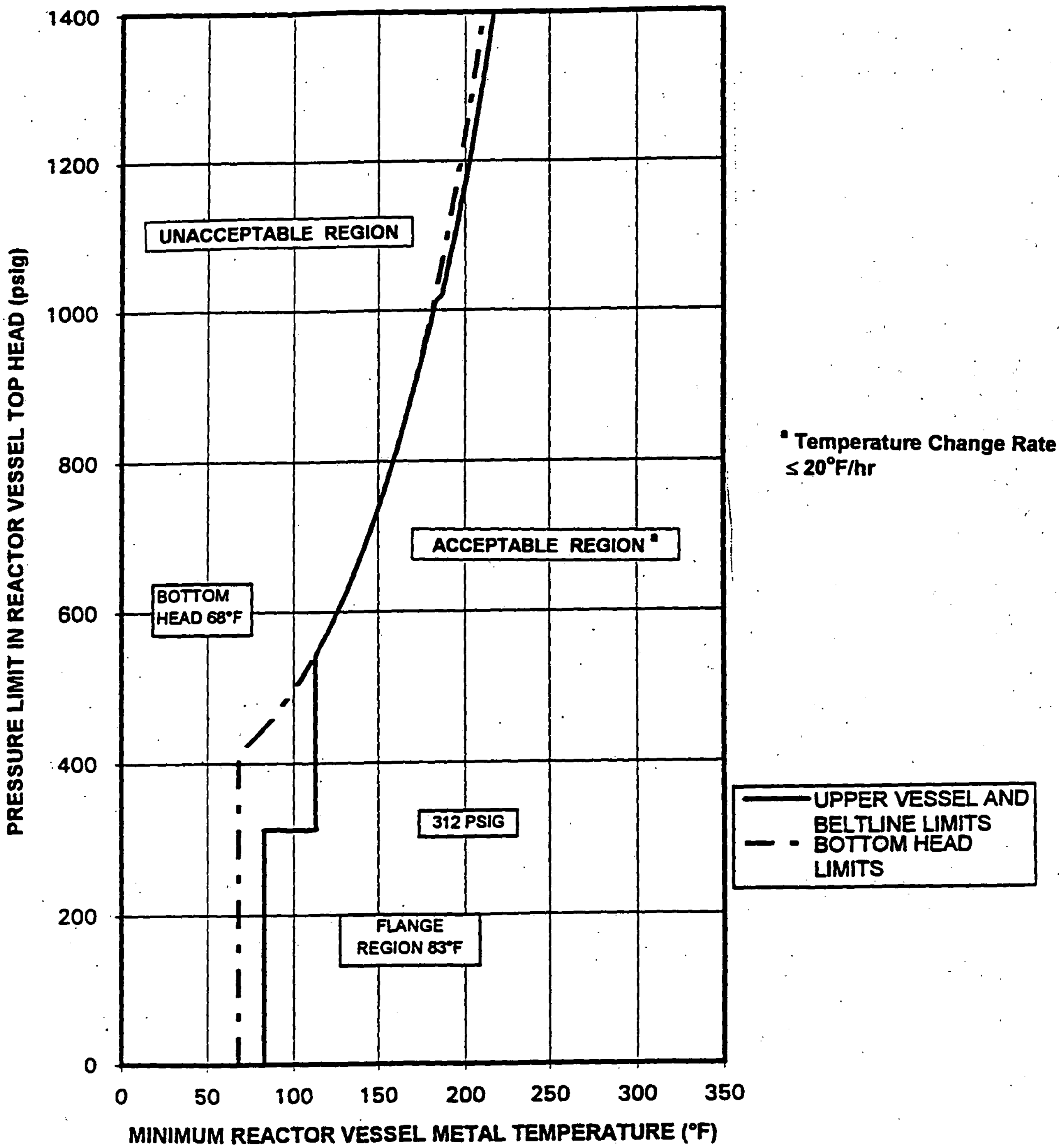


FIGURE 3.6.K-2

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

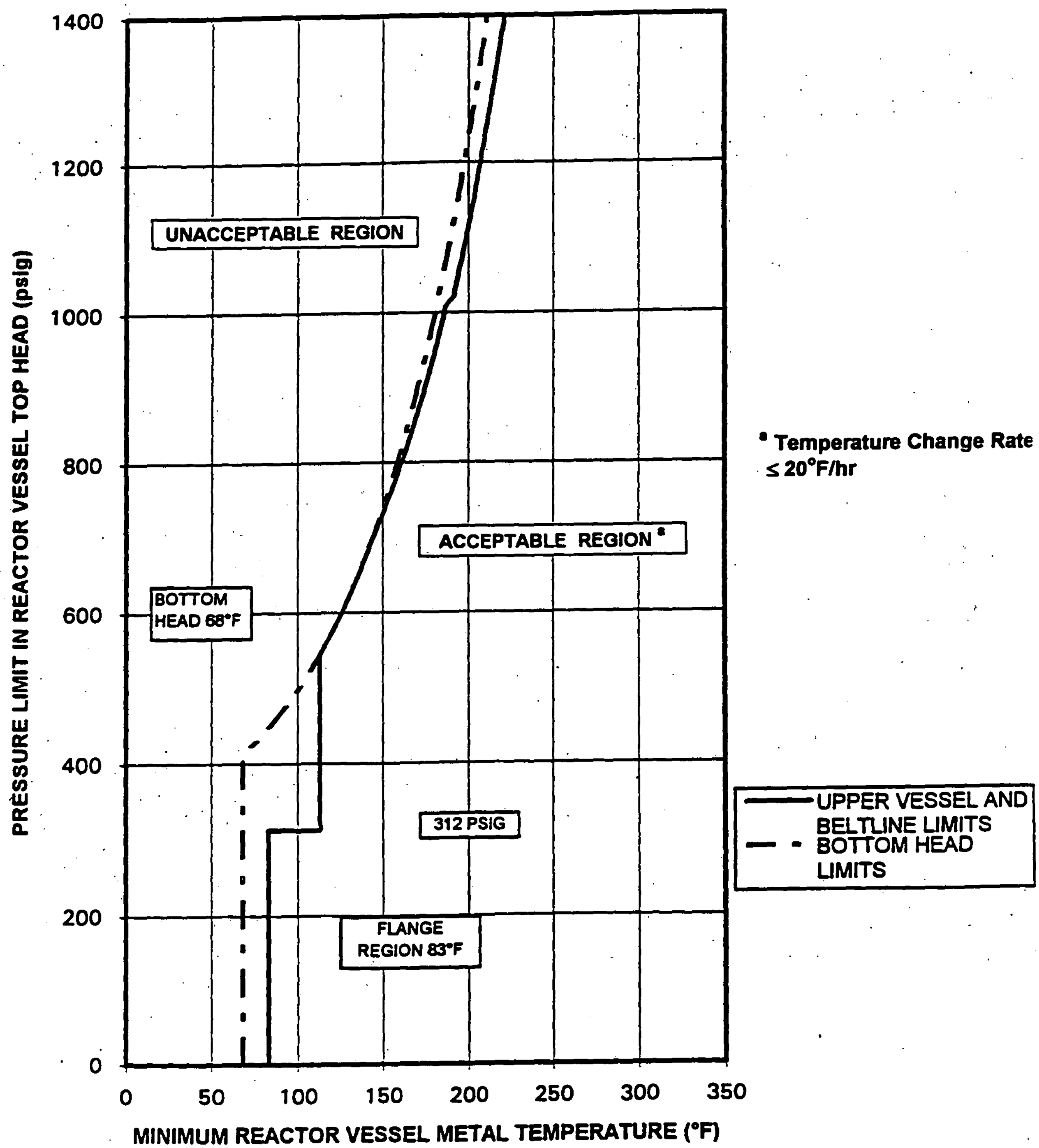
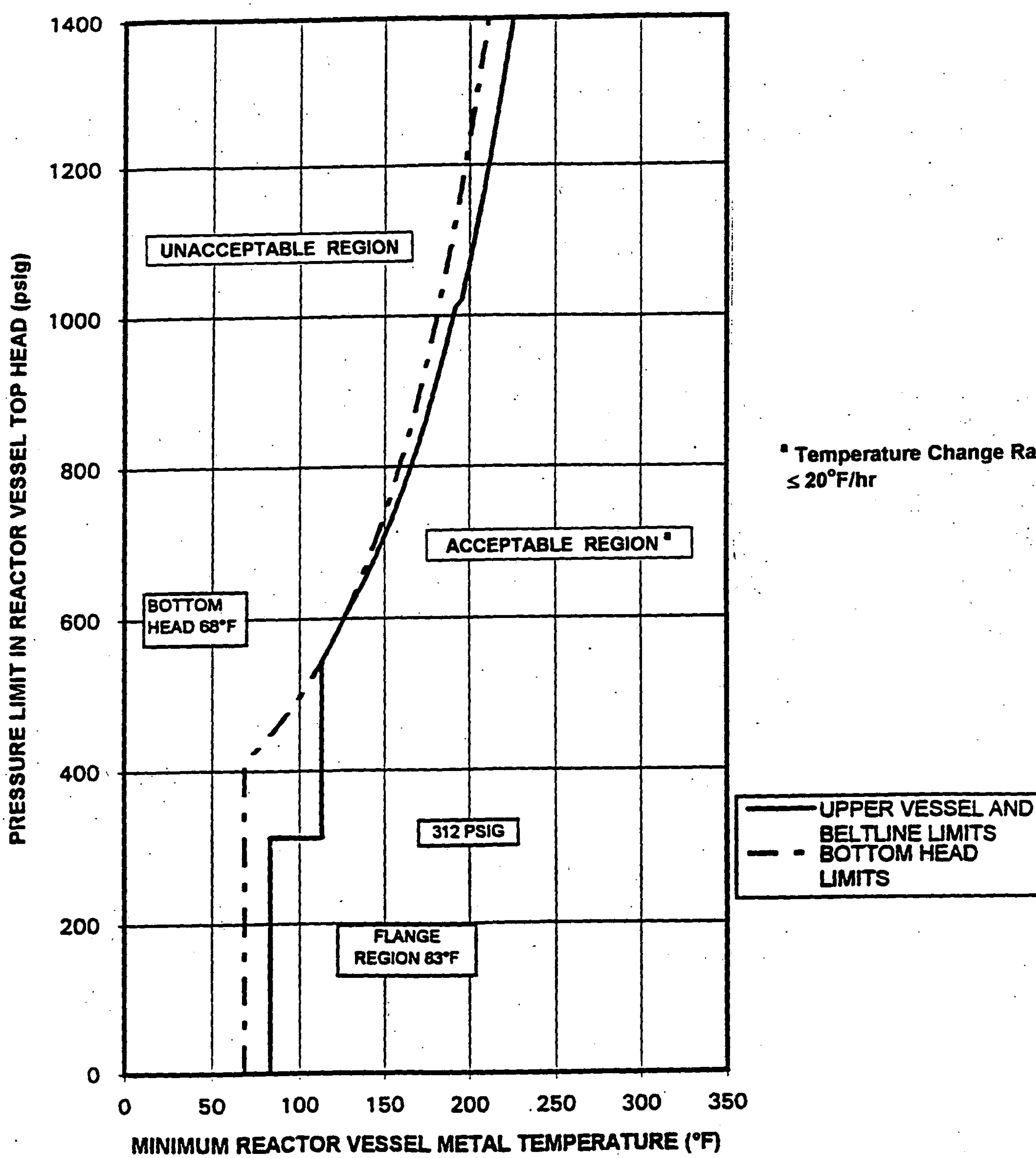


FIGURE 3.6.K-3

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



* Temperature Change Rate ≤ 20°F/hr

FIGURE 3.6.K-4

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

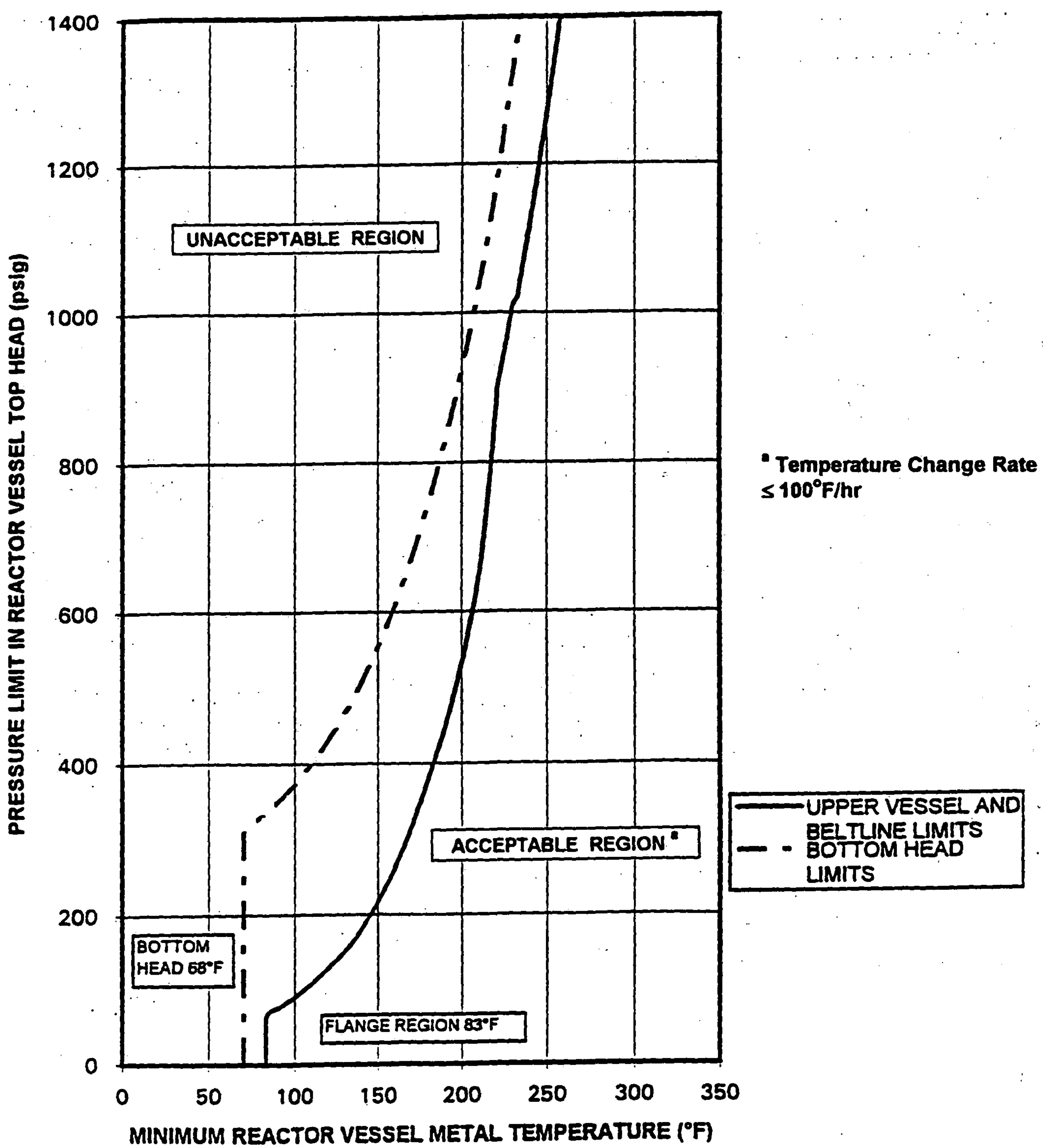
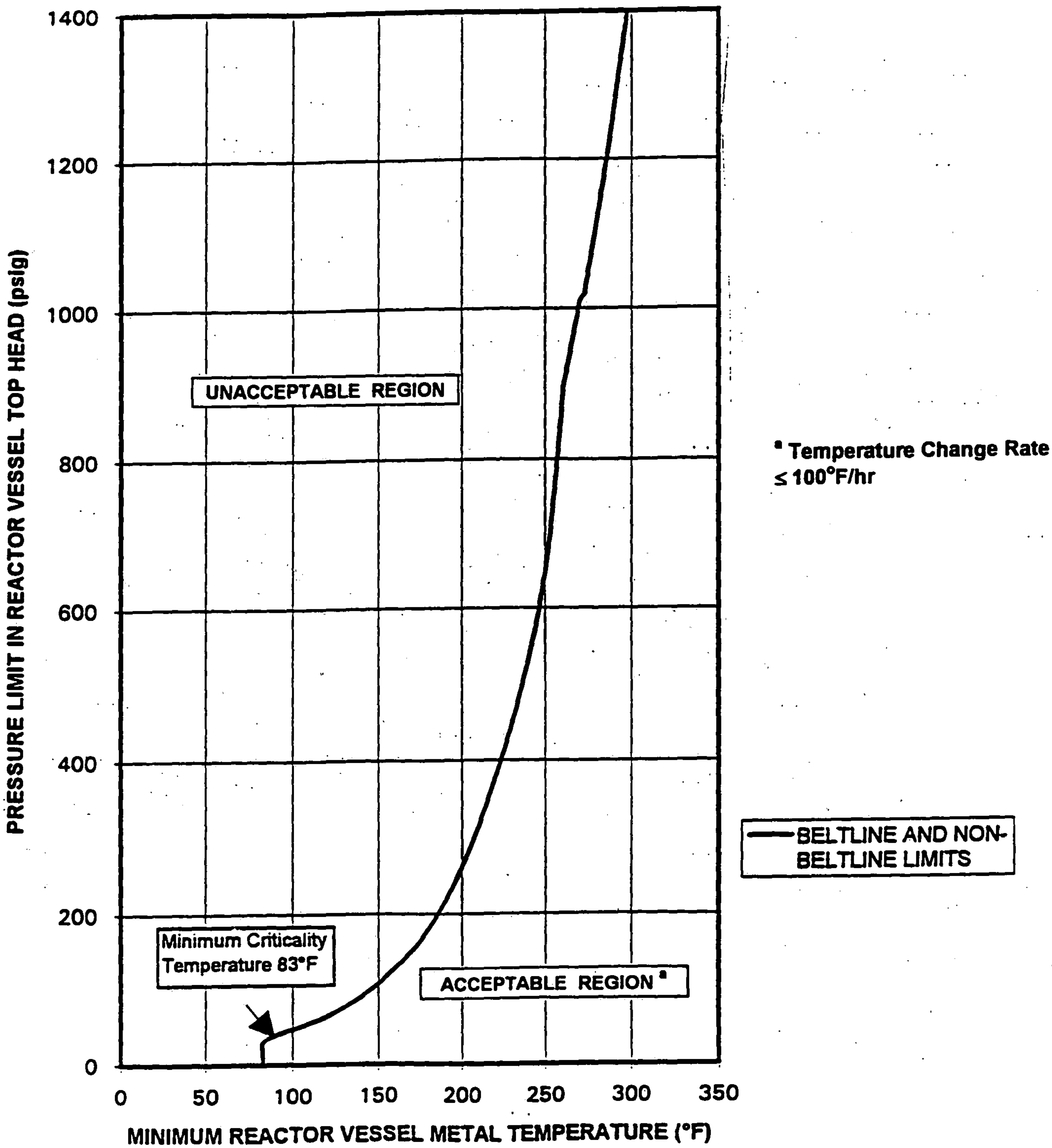


FIGURE 3.6.K-5

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



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3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1.1 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The 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

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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 3.3.6.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.

Figures 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.

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

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.

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Figure 3.6.K-5 - Core Critical Operation

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.

3/4.6.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 valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

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

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as

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required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Residual Heat Removal - HOT SHUTDOWN

3/4.6.P Residual Heat Removal - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

A single shutdown cooling mode subsystem provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two subsystems be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associtated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Therefore, to meet the Limiting Condition for Operation, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems (the ability to take credit for a common heat exchanger and discharge piping only applies to the SDC mode of RHR).