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50.90

NLS2005090
October 12, 2005

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

Subject: License Amendment Request to Revise Technical Specification Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 to Replace Cycle Restriction with Effective Full Power Year Limitation
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:

1. Letter NLS2004005 from Randall K. Edington, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated January 29, 2004, "License Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3," Cooper Nuclear Station Docket 50-298, DPR-46."

2. Letter NLS2004043 from Randall K. Edington, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated April 8, 2004, "Supplement to License Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3," Cooper Nuclear Station Docket 50-298, DPR-46."

3. Letter from Michelle C. Honcharik, U.S. Nuclear Regulatory Commission, to Randall K. Edington, Nebraska Public Power District, dated July 14, 2004, "Cooper Nuclear Station – Issuance of Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 (TAC No. MC1940)."

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90 to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). This submittal is based on References 1 and 2 and resolves a Nuclear Regulatory Commission (NRC) concern regarding the validity of the Pressure/Temperature (P/T) Limit Curves as vessel fluence approached 32 Effective Full Power Years (EFPY). In License Amendment 204, Reference 3, the NRC authorized use of the current TS P/T curves through Cycle 23, pending recalculation of the vessel fluence using Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," compatible methodology and consequent revalidation of the curves.

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The reactor pressure vessel fluence has been recalculated. A fluence evaluation was performed by TransWare Enterprises Inc. using the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology. The revised fluence causes a slight increase in the Adjusted Reference Temperature (ART) for the limiting material. In order to adjust for this increase the P/T curves should be limited to 30 EFPY. Therefore, NPPD requests NRC approval to remove the Cycle 23 restriction from the current P/T curves and replace it with a limitation of 30 EFPY.

The proposed amendment would revise TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," curves 3.4.9-1, "Pressure/Temperature Limits for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown," 3.4.9-2, "Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests, and 3.4.9-3, "Pressure/Temperature Limits for Criticality," to remove the cycle operating restriction and replace it with a limitation of 30 EFPY.

NPPD requests approval of the proposed TS revisions by September 1, 2006, to support startup from Refueling Outage 23, projected for October 2006. Once approved, the amendment will be implemented within 60 days.

Attachment 1 provides a description of the TS changes, the basis for the amendment, the no significant hazards consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the environmental impact evaluation pursuant to 10 CFR 51.22. Attachment 2 provides marked up pages of the proposed changes to the current CNS TS, and Attachment 3 provides the revised TS pages in final typed format. CNS Calculation Number NEDC 05-019, "Reactor Pressure Vessel Fluence Evaluation," is enclosed for your information.

The proposed TS changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 211, issued March 22, 2005, have been incorporated into this request. NPPD has concluded that the proposed change does not involve a significant hazards consideration and that it satisfies the categorical exclusion criterion of 10 CFR 51.22(c)(9).

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

If you have any questions concerning this matter, please contact Paul Fleming, Licensing Manager, at (402) 825-2774.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 10/12/05
(date)

Sincerely,

Randall K. Edington

Randall K. Edington
Vice President – Nuclear and
Chief Nuclear Officer

/cb

Attachments
Enclosure

cc: U.S. Nuclear Regulatory Commission
Regional Office - Region IV

Senior Project Manager
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector
USNRC - CNS

Nebraska Health and Human Services
Department of Regulation and Licensure

NPG Distribution

CNS Records

# License Amendment Request to Revise Technical Specification Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 to Replace Cycle Restriction with Effective Full Power Year Limitation

## Cooper Nuclear Station, NRC Docket 50-298, DPR-46

**Revised Pages**
Technical Specification Pages
3.4-23
3.4-24
3.4-25

## 1.0 Description

This letter is a request to amend Operating License (OL) DPR-46 for Cooper Nuclear Station (CNS).

The proposed change would revise the OL to remove the one cycle restriction on Technical Specification (TS) Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 and replace it with a limitation of 30 Effective Full Power Years (EFPY). Nebraska Public Power District (NPPD) is requesting approval of the proposed TS revision by September 1, 2006, which is projected for October 2006.

## 2.0 Proposed Change

This proposed change will revise TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits," curves to remove the cycle restriction from TS Figures 3.4.9-1, "Pressure/Temperature Limits for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown," 3.4.9-2, "Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests," and 3.4.9-3, "Pressure/Temperature Limits for Criticality," and replace it with a limitation of 30 EFPY. The proposed amendment includes a full set of updated P/T curves for pressure test, core not critical, and core critical conditions.

1. On TS Figure 3.4.9-1, "Pressure/Temperature Limits for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown, 32 EFPY," in the title change "32" to "30" and at the bottom change "End of Cycle 23" to "30 EFPY."

2. On TS Figure 3.4.9-2, "Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests," in the title change "32" to "30" and at the bottom change "End of Cycle 23" to "30 EFPY."

3. On TS Figure 3.4.9-3 "Pressure/Temperature Limits for Criticality," in the title change "32" to "30" and at the bottom change "End of Cycle 23" to "30 EFPY."

## 3.0 Background

The current Heatup/Cooldown curves were developed for 32 EFPY in accordance with the requirements of 10 CFR 50 Appendix G. The methods in the 1995 Edition, 1996 Addenda of American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G, were used with ASME Section XI Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." Regulatory Guide (RG) 1.147, Revision 14, identifies ASME Section XI Code Case N-640 as a Nuclear Regulatory Commission (NRC) acceptable ASME Section XI code case. The P/T curves for CNS were developed by Structural Integrity Associates. In License Amendment 204 (Reference 10), the NRC authorized use of the current P/T curves through Cycle 23, pending recalculation of the vessel fluence using a RG 1.190 compatible methodology, and consequent revalidation of the curves. CNS calculation NEDC 05-019, "Reactor Pressure Vessel Fluence Evaluation," satisfies this commitment and allows us to remove the Cycle 23 restriction from the P/T curves.

The reactor vessel is a vertical cylindrical pressure vessel with hemispherical heads of welded construction (Reference 16). The reactor vessel is designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. The vessel is designed, fabricated, inspected, tested, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition and January 1966 Addenda), its interpretations, and applicable requirements for Class A Vessels as defined therein, (Reference 4).

Amendment 201 (Reference 12) to the CNS Operating License, issued October 31, 2003, implemented the Boiling Water Reactor Vessel and Internals Reactor Pressure Vessel Integrated Surveillance Program (ISP). As part of the ISP, CNS vessel surveillance capsules are evaluated using fluence calculations that conform with RG 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence. Additionally, CNS committed in Amendment 201 to recalculate fluences for previously pulled surveillance capsules in conformance with RG 1.190.

The Radiation Analysis Modeling Application (RAMA) Fluence Methodology (Reference 5) is a system of software components that include a transport code, parts model builder code, state-point model builder code, fluence calculator, and nuclear data library. The RAMA transport code couples a three-dimensional deterministic transport solver with an arbitrary geometry modeling capability to provide a flexible and accurate tool for determining fluxes in any light water reactor design. The model builder codes use reactor design inputs and operating data to generate geometry and material inputs for the transport solver. The fluence calculator uses isotopic activation and decay information with reactor operating history to provide an accurate estimate of component fluence. The nuclear data library contains nuclear cross section data and response functions that are used in the transport and fluence calculations. The nuclear data library is based upon the ENDF/B-VI data file and the BUGLE-96 nuclear data library.

## 4.0 Technical Analysis

Appendix G to 10 CFR Part 50, which is invoked by 10 CFR 50.60, specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including reactor pressure vessels (RPV), during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. In order to support evaluations that demonstrate compliance with these requirements will be maintained, information regarding irradiated RPV material properties and the neutron fluence level of the RPV is necessary.

The Updated Safety Analysis Report (Reference 17) describes the RPV Material Surveillance Test Program, which is used to periodically revalidate and update the P/T curves.

The RAMA Methodology for neutron fluence has been benchmarked using experimental and numerical problems specified in RG 1.190. The results of the benchmark cases are documented in the EPRI report entitled "RAMA Fluence Methodology - Benchmark Manual Evaluation of R G 1.190 Benchmark Problems."

The reactor vessel neutron fluence was calculated using the RAMA methodology. Projected neutron fluence values are presented for two points in time: The end of cycle 21 and the current end of design life, 32 EFPY. It was assumed in projecting the 32 EFPY fluence that cycle 21 was an equilibrium cycle and representative of future operating cycles. Thus, the incremental fluence determined for cycle 21 was assumed to be constant which allows the fluence between cycle 21 and 32 EFPY to be estimated using linear interpolation for the purpose of evaluating the P/T curves. The fluence at 32 EFPY calculated by RAMA is 1.67E18 neutron per centimeter squared ($n/cm^2$) (NEDC 05-019, Table 7-3). The fluence at 32 EFPY used in the development of the current P/T curves is 1.57E18 $n/cm^2$. The revised fluence causes a slight increase in the Adjusted Reference Temperature (ART) for the limiting material (was 127.6°F).

The calculation meets applicable industry standards and provides a reasonable estimate of the Reactor Vessel beltline region neutron fluence. The calculation uses a methodology approved by the NRC (Reference 15.)

The maximum beltline fluence calculated in NEDC 05-019 is slightly higher than the fluence used in the generation of the current P/T curves. Therefore, the P/T curves should be limited to 30 EFPY: [(1.57E18 $n/cm^2$)/(1.67E18 $n/cm^2$)* 32 EFPY= 30.1 EFPY].

The "End of Cycle 23" limitation on the existing P/T curves in the TS may be removed and replaced with a limit of 30 EFPY. 30 EFPY is conservative because it is less than the 30.1 EFPY in the calculation.

## 5.0 Regulatory Analysis

### 5.1 No Significant Hazards Consideration

10 CFR 50.91(a)(l) requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazard posed by issuance of an amendment. Nebraska Public Power District (NPPD) has evaluated this proposed amendment with respect to the criteria given in 10 CFR 50.92(c).

This proposed change will revise Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits," curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3, from 32 Effective Full Power Years (EFPY) to 30 EFPY and by deleting a restriction that the curves are valid through the end of cycle 23. The proposed change includes a full set of updated P/T curves for pressure test, core not critical, and core critical conditions. The three regions of the reactor pressure vessel that are evaluated are the beltline region, the bottom head region, and the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

The following evaluation supports a finding of "no significant hazards consideration" associated with this proposed change.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed revisions to the Cooper Nuclear Station (CNS) P/T curves are based on the recommendations in Regulatory Guide (RG) 1.99, Revision 2, and are, therefore, in accordance with the latest Nuclear Regulatory Commission (NRC) guidance. The fluence evaluation for the P/T curves for 30 EFPY was performed using the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology. The curves generated from this method provide guidance to ensure that the P/T limits will not be exceeded during any phase of reactor operation. Accordingly, the proposed revision to the CNS P/T curves is based on an NRC accepted means of ensuring protection against brittle reactor vessel fracture, and compliance with 10 CFR 50 Appendix G. The curves are the same as approved in Amendment Number 204, CNS is only requesting to remove the one cycle limitation and limit their use to 30 EFPY based on the shift in the Adjusted Reference Temperature (ART) using the new fluence values. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9 P/T curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 does not significantly increase the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change updates existing P/T operating limits to correspond to the current NRC guidance. The proposed TS change extends the use of the current, NRC-approved P/T curves beyond the end of Cycle 23 to 30 EFPY. The proposed change does not involve a physical change to the plant, add any new equipment or any new mode of operation. These TS changes demonstrate compliance with the brittle fracture requirements of 10 CFR 50 Appendix G and, therefore, do not create the possibility for a new or different kind of accident from any accident previously evaluated.

   Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9 P/T curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   Response: No

   The proposed change revises the existing CNS P/T curves to limit their use to 30 EFPY based on fluence calculation using the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology. The curves have not been recalculated. Limiting the use of the P/T curves to 30 EFPY, based on the recalculation of the fluence per the NRC-approved (RAMA) fluence methodology does not affect a margin of safety. These changes do not affect any system used to mitigate accidents or transients.

   Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9 P/T curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 does not involve a significant reduction in the margin of safety.

In conclusion, NPPD has determined that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

### 5.2 Regulatory Requirements and Guidance

10 CFR 50.60, "Acceptance Criteria for Fracture Prevention of Lightwater Nuclear Power Reactors for Normal Operation," requires that the pressure and temperature limits as well as the associated vessel surveillance program be consistent with 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

10 CFR 50 Appendix G and Appendix H also describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P/T curves. 10 CFR 50 Appendix G specifies the fracture toughness and testing requirements for reactor vessel material in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G. 10 CFR 50, Appendix G also requires the prediction of the effects of neutron irradiation on the vessel embrittlement by calculating the ART and Charpy upper shelf energy. Generic Letter 88-1 1, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," requires that the methods in Regulatory Guide 1 .99, Revision 2, be used to predict the effect of neutron irradiation on the reactor vessel material. Appendix H of 10 CFR 50 requires the establishment of a surveillance program to periodically withdraw surveillance capsules from the reactor vessel.

The heatup and cooldown process for CNS is controlled by TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits" curves, which are developed based on fracture mechanics analysis. These limits are developed according to the NRC-approved Radiation Analysis Modeling Application (RAMA) fluence methodology.

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

## 6.0 Environmental Consideration

10 CFR 51.22(b) allows that an environmental assessment (EA) or an environmental impact statement (EIS) is not required for any action included in the list of categorical exclusions in 10 CFR 51.22(c). 10 CFR 51.22(c)(9) identifies an amendment to an operating license which changes a requirement with respect to installation or use of a facility component located within the restricted area, or which changes an inspection or a surveillance requirement, as a categorical exclusion if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amount of any effluents that may be released off-site, or (3) result in an increase in individual or cumulative occupational radiation exposure.

NPPD has reviewed the proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the proposed license changes. The basis for this determination is as follows:

1. The proposed license amendment does not involve significant hazards as described previously in the No Significant Hazards Consideration Evaluation.

2. This proposed change does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site. The proposed license amendment does not introduce any new equipment, nor does it require any existing equipment or systems to perform a different type of function than they are presently designed to perform. NPPD has concluded that there will not be a significant increase in the types or amounts of any effluents that may be released off-site and these changes do not involve irreversible environmental consequences beyond those already associated with normal operation.

3. This change does not adversely affect plant systems or operation and, therefore, does not significantly increase individual or cumulative occupational radiation exposure beyond that already associated with normal operation.

## 7.0 References

1. 10 CFR Part 50, Appendix G, Fracture Toughness Requirements
2. 10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements
3. 10 CFR 50.60, Acceptance Criteria for Fracture Prevention of Lightwater Nuclear Power Reactors for Normal Operation
4. ASME Boiler and Pressure Vessel Code, Section III (1965 Edition and January 1966 Addenda)
5. BWRVIP-114, RAMA Fluence Methodology Theory Manual, June 11, 2003
6. CNS Calculation Number NEDC 05-019, "Reactor Pressure Vessel Fluence Evaluation"
7. EPRI Report entitled "RAMA Fluence Methodology – Benchmark Manual Evaluation of Regulatory Guide 1.190 Benchmark Problems"
8. Generic Letter 88-1 1, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"
9. Letter from Michelle C. Honcharik, U.S. Nuclear Regulatory Commission, to Randall K. Edington, Nebraska Public Power District, dated July 14, 2004, "Cooper Nuclear Station – Issuance of Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 (TAC No. MC1940)." [Amendment 204]
10. Letter NLS2004005 from Randall K. Edington, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated January 29, 2004, "License Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3," Cooper Nuclear Station Docket 50-298, DPR-46."
11. Letter NLS2004043 from Randall K. Edington, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated April 8, 2004, "Supplement to License Amendment Request to Revise Technical Specification 3.4.9 Pressure Temperature (P/T) Curves, Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3," Cooper Nuclear Station Docket 50-298, DPR-46."

12. Letter from Michelle C. Honcharik, U. S. Nuclear Regulatory Commission, to Clay C. Warren, Nebraska Public Power District, dated October 31, 2003, "Cooper Nuclear Station – Issuance of Amendment to Implement Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program (TAC NO. MB7209)" [Amendment 201]

13. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001

14. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

15. NRC Safety Evaluation of Proprietary EPRI Reports, "BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual-Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115)," "RAMA Fluence Methodology-Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," and "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)," and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)" (TAC No. MB9765).

16. USAR Section IV-2.5.1

17. USAR Section IV-2.7.2

# ATTACHMENT 2

## PROPOSED TECHNICAL SPECIFICATION REVISIONS (MARK-UP)

## COOPER NUCLEAR STATION
## NRC DOCKET 50-298, LICENSE DPR-46

Technical Specification Pages

3.4-23
3.4-24
3.4-25

30

Cooper Heatup/Cooldown, Core Not Critical Curve (Curve B), ~~32~~ EFPY

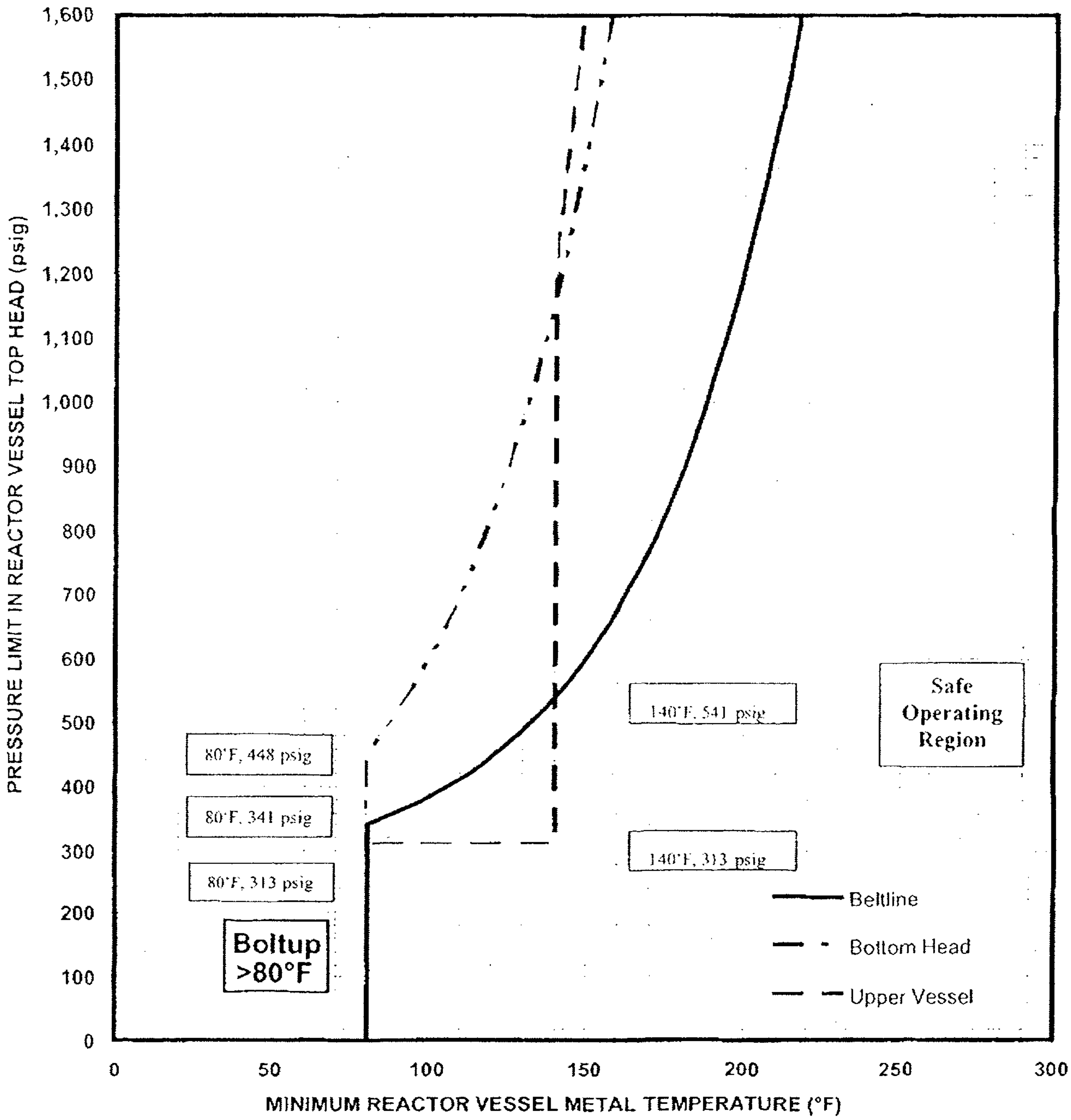


Figure 3.4.9-1 (page 1 of 1)
Pressure/Temperature Limits for
Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown
Valid Through ~~End of Cycle 23~~
30 EFPY

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Cooper Pressure Test Curve (Curve A), ~~32~~ EFPY

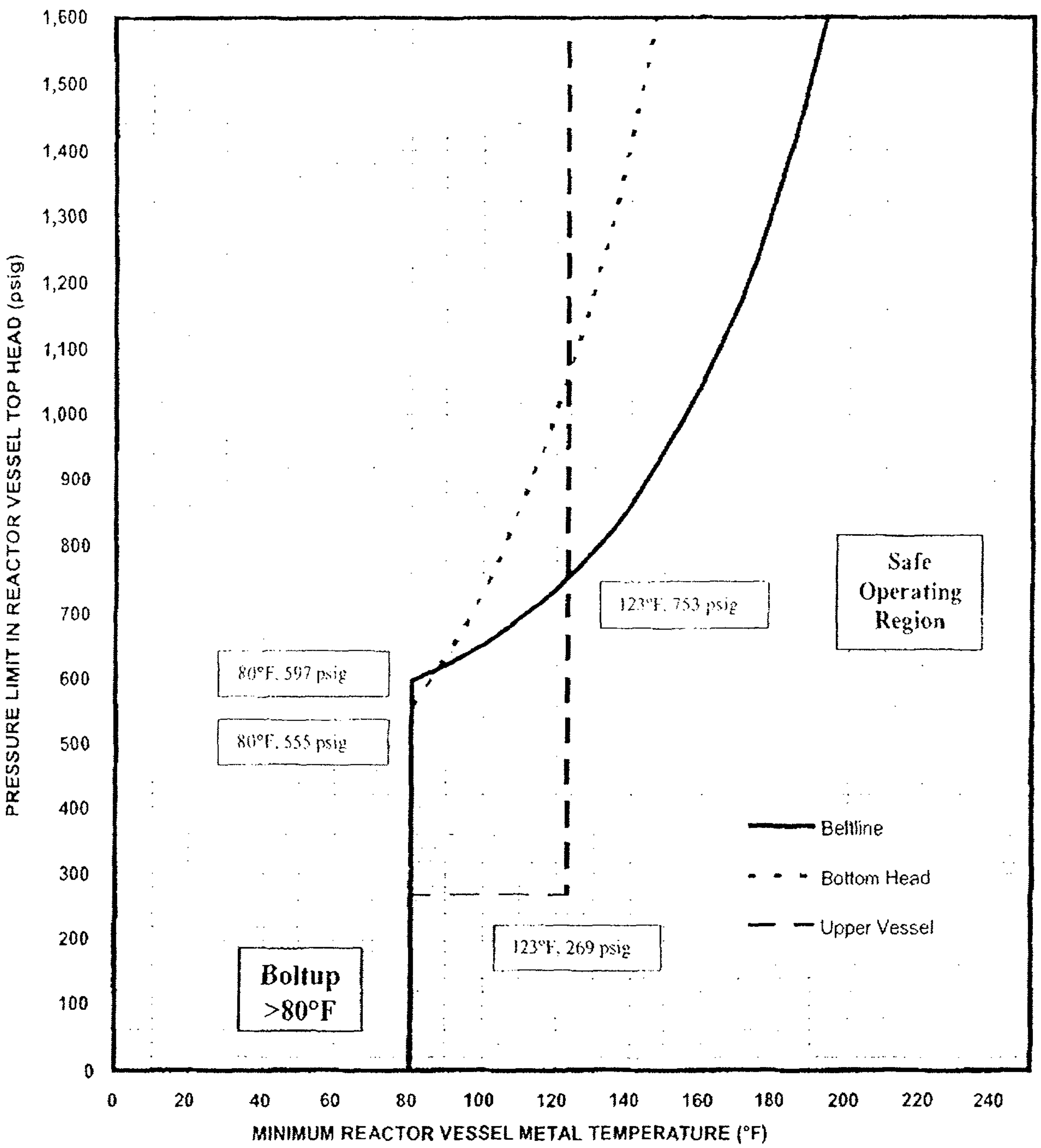


Figure 3.4.9-2 (page 1 of 1)
Pressure/Temperature Limits for
Inservice Hydrostatic and Inservice Leakage Tests
Valid Through ~~End of Cycle 23~~
30 EFPY

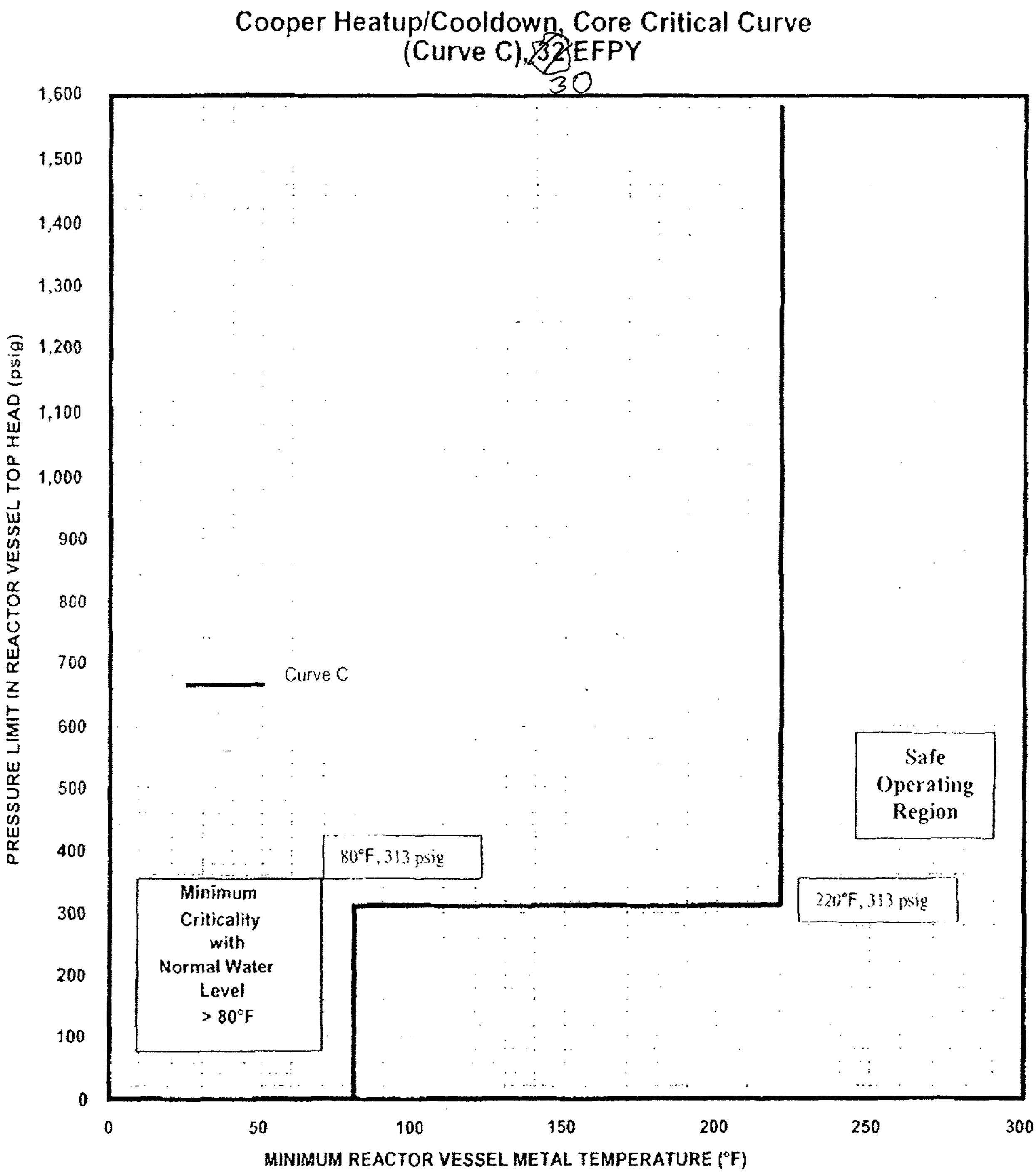


Figure 3.4.9-3 (page 1 of 1)
Pressure/Temperature Limits for Criticality
Valid Through ~~End of Cycle 23~~
30 EFPY

# ATTACHMENT 3

## PROPOSED TECHNICAL SPECIFICATION REVISIONS
## (FINAL TYPED)

## COOPER NUCLEAR STATION
## NRC DOCKET 50-298, LICENSE DPR-46

Technical Specification Pages

3.4-23
3.4-24
3.4-25

## Cooper Heatup/Cooldown, Core Not Critical Curve (Curve B), 30 EFPY

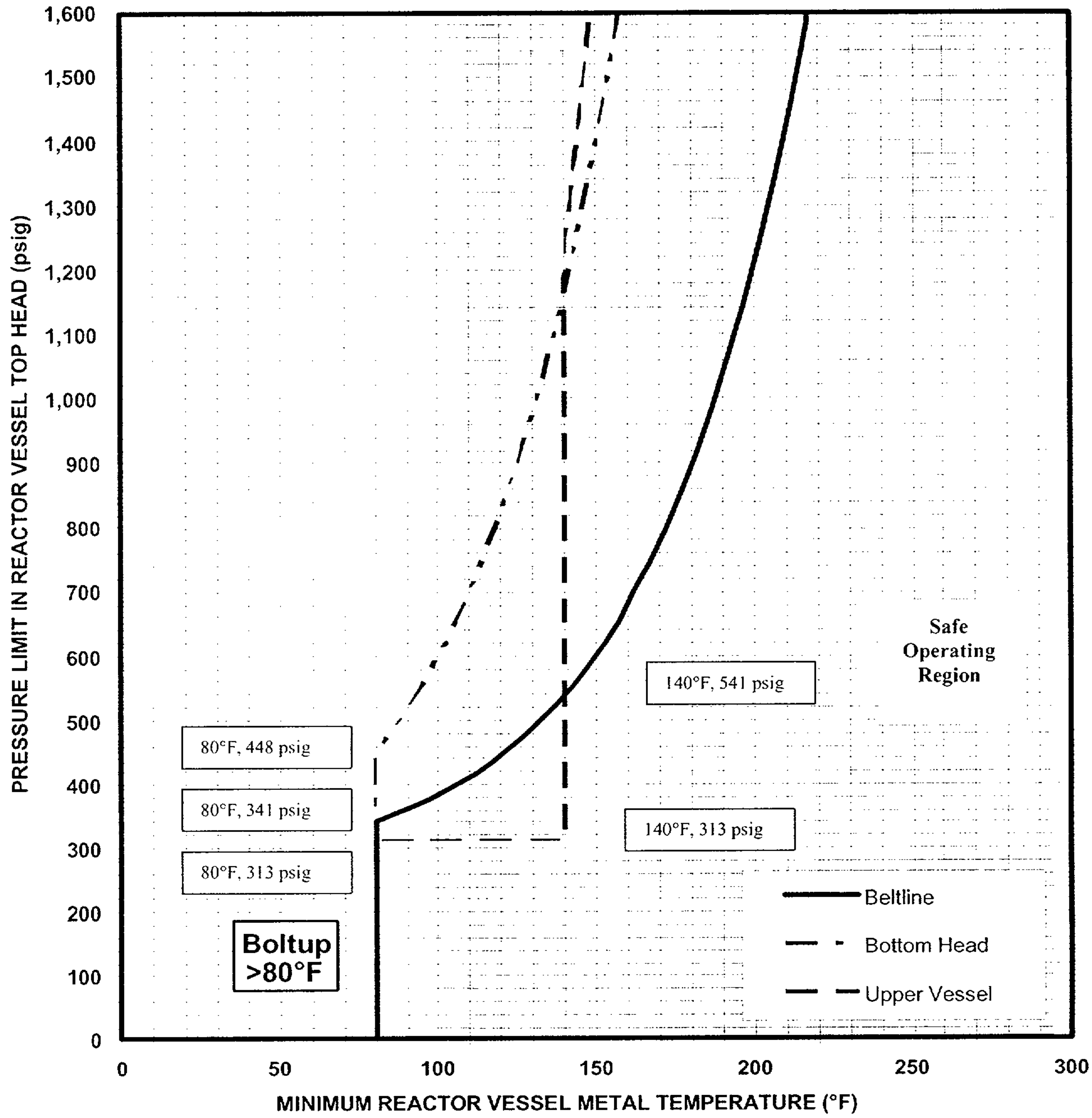


**Figure 3.4.9-1 (page 1 of 1)**
**Pressure/Temperature Limits for**
Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown

**Valid Through 30 EFPY**

## Cooper Pressure Test Curve (Curve A), 30 EFPY

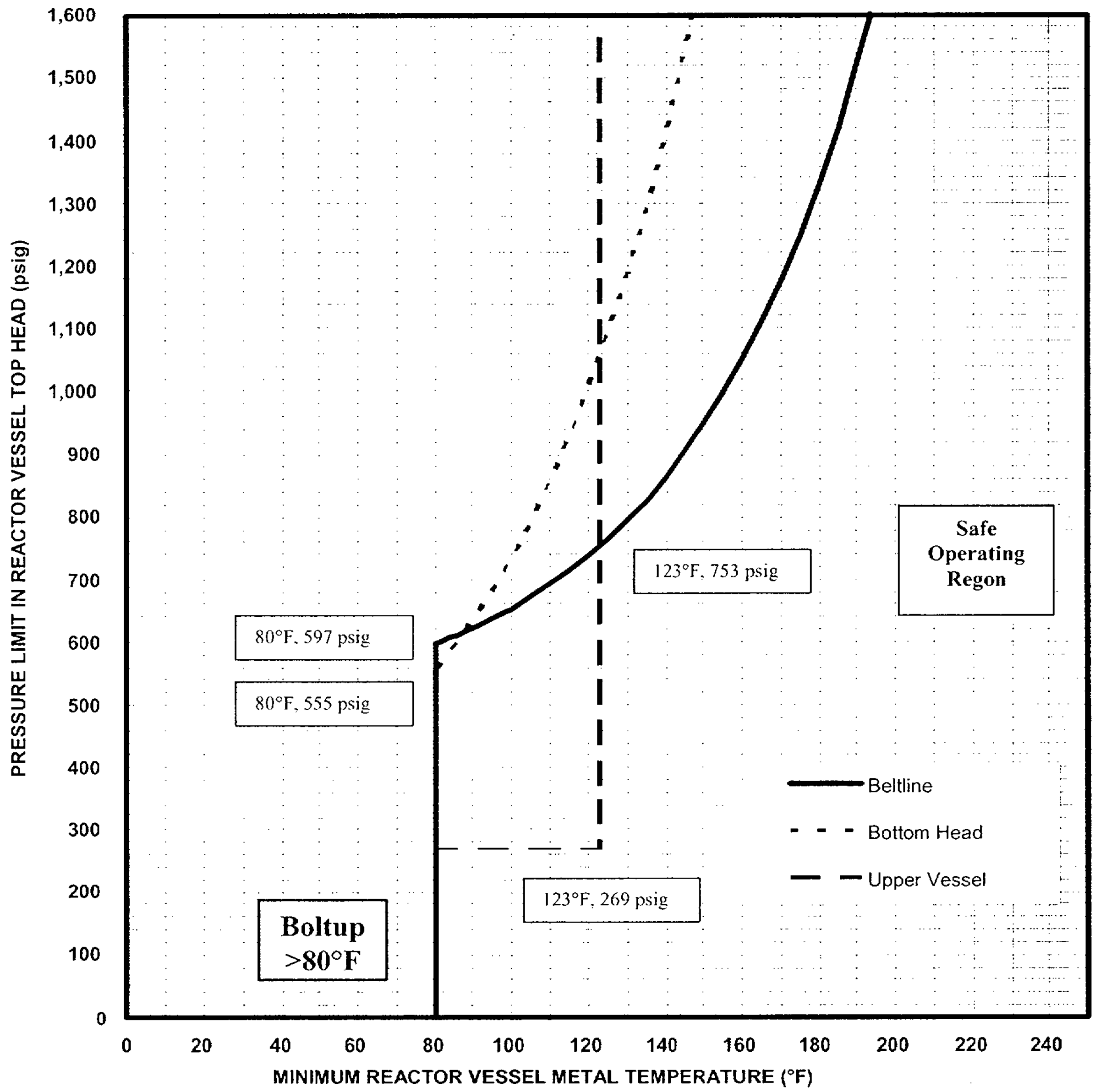


**Figure 3.4.9-2 (page 1 of 1)**
**Pressure/Temperature Limits for**
**Inservice Hydrostatic and Inservice Leakage Tests**

**Valid Through 30 EFPY**

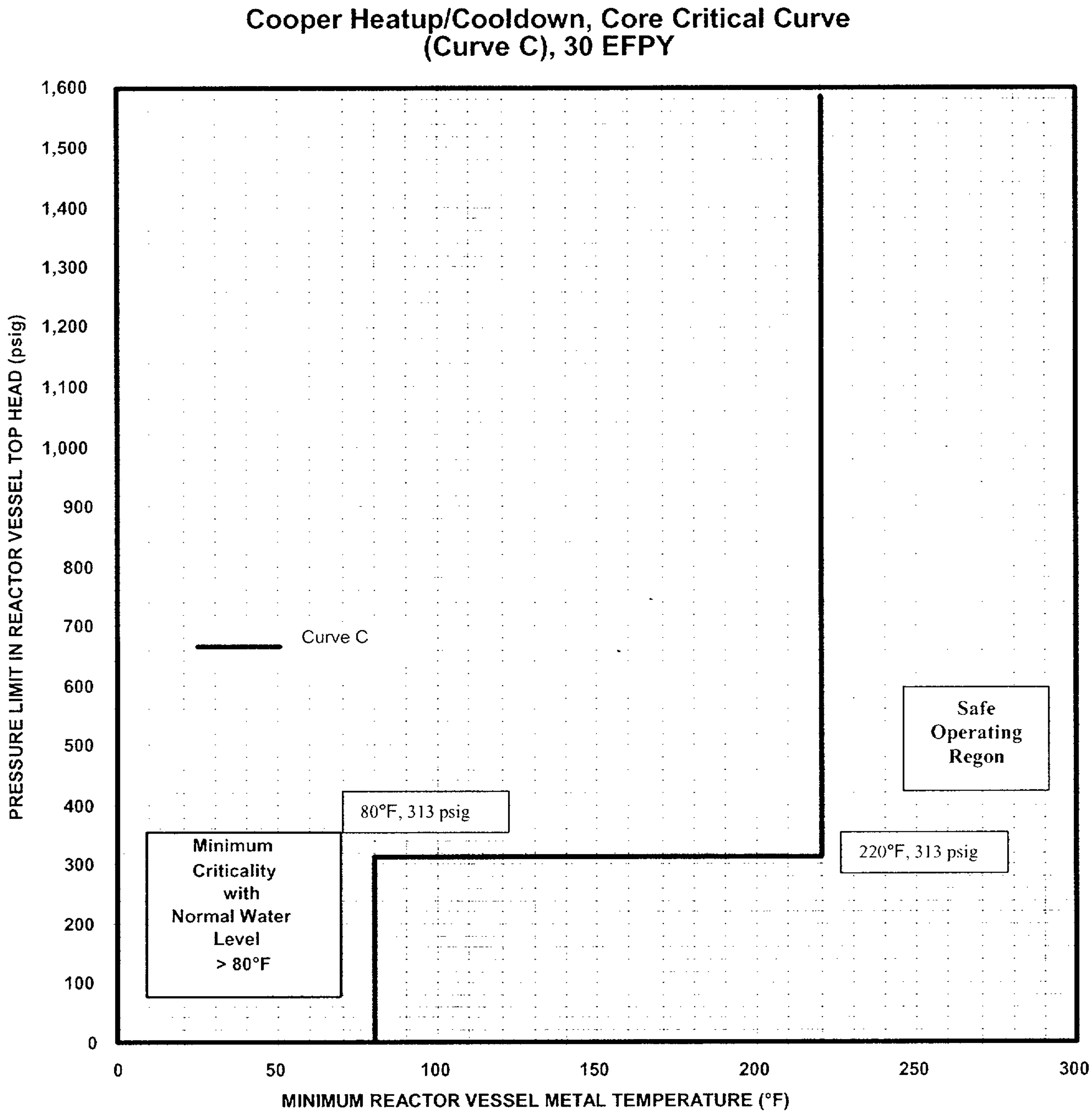


**Figure 3.4.9-3 (page 1 of 1)**
**Pressure/Temperature Limits for Criticality**

**Valid Through 30 EFPY**