NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

April 22, 2002

10 CFR Part 50
Section 50.90 and
Section 50.12

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request
Revised Pressure and Temperature Curves

Attached is a request for changes to the Technical Specifications (TS), Appendix A of Operating License DPR-22, for the Monticello Nuclear Generating Plant. This request is submitted pursuant to, and in accordance with, the provisions of 10 CFR Part 50, Section 50.90 and Section 50.12.

Nuclear Management Company, LLC, (NMC) proposes changes to the Monticello TS to revise the reactor vessel pressure and temperature (P/T) limit curves.

In addition to proposing changes to revise the Monticello TS P/T limit curves, NMC is requesting an exemption from the requirements of 10 CFR 50, Appendix G, to allow the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-640 as the basis for these revised curves. The proposed P/T curves were developed in accordance with the 1989 Edition ASME Section XI, Appendix G; 10 CFR 50, Appendix G; and ASME Code Case N-640. The use of this Code Case as the basis for the proposed P/T curves constitutes an alternative to the requirements of 10 CFR 50, Appendix G. 10 CFR 50.60(b) provides that the NRC may grant alternatives to the requirements in Appendix G by using the procedures for exemption specified in 10 CFR 50.12. Exemptions to use Code Case N-640 have been granted for several plants, including Quad Cities, Units 1 and 2 (February 4, 2000, ADAMS Accession No. ML003680441); Dresden, Units 2 and 3 (August 25, 2000, ADAMS Accession No. ML003745769); Hatch Units 1 and 2 (August 29, 2000, ADAMS Accession No. ML003745181); and Limerick, Unit 1 (September 7, 2000, ADAMS Accession No. ML003740024).

Exhibit A contains the Proposed Changes, Reason for Changes, a Safety Evaluation, a Determination of No Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains the Request for Exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G. Exhibit C contains current Monticello TS figures marked up to show the proposed change. Exhibit D contains the revised Monticello TS figures.

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This submittal does not contain any new NRC commitments and does not modify any prior commitments.

This application has been reviewed by the Monticello Operations Committee and the Offsite Review Committee. A copy of this submittal, along with the evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1).

Nuclear Management Company, LLC requests NRC approval of this Technical Specification change by December 31, 2002 to facilitate planning and scheduling for the next refueling outage, currently scheduled to begin on April 26, 2003. Once approved the amendment will be implemented within a period of 60 days.

If you have any questions regarding this License Amendment Request please contact Doug Neve, Licensing Manager, at (763) 295-1353.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 22 day of april, 2002

Kathryn I Kleine
Notary

KATHRYN I. KLEINE
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments: Exhibit A – Evaluation of Proposed Changes to the Monticello Technical Specifications
Exhibit B - Request for Exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G
Exhibit C - Current Monticello Technical Specifications Figures Marked up With Proposed Changes
Exhibit D - Revised Monticello Technical Specifications Figures

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J. Silberg, Esq

License Amendment Request
Revised Pressure and Temperature Curves

Evaluation of Proposed Changes to the Monticello Technical Specifications

Pursuant to 10 CFR Part 50, Section 50.90 and Section 50.91, Nuclear Management Company, LLC (NMC) hereby proposes the following changes to Appendix A, of Facility Operating License DPR-22, Technical Specifications (TS) for the Monticello Nuclear Generating Plant.

Background

This amendment request proposes new pressure and temperature (P/T) limit curves for the Monticello TS. The amendment proposes to update the reactor pressure vessel (RPV) P/T limit curves for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality. The proposed P/T curves were developed in accordance with the 1989 Edition ASME Code Section XI, Appendix G; 10 CFR 50, Appendix G; and ASME Code Case N-640.

Proposed Changes and Reason for Changes

NMC proposes to revise the Monticello TS requirements for reactor pressure vessel (RPV) P/T limit curves. Changes are proposed to Monticello TS Figures 3.6.2, 3.6.3 and 3.6.4 which show the P/T limit curves for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality. No changes to the Limiting Conditions for Operation or any Surveillance Requirements of Technical Specification 3.6.B are proposed.

These changes are needed because 10 CFR Part 50, Appendix G, requires that P/T limits be established for reactor pressure vessels during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G states that "The appropriate requirements on both the P/T limits and the minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the ASME Code, Section XI, Appendix G limits.

NMC is requesting that the NRC exempt Monticello from application of specific requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G, and substitute use of ASME Code Case N-640. This code case allows use of the K_{IC} fracture toughness curve shown in ASME Code, Section XI, Appendix A, Figure A-2200-1, which provides greater allowable fracture toughness than the corresponding K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness in the development of the P/T limit curves. The other margins involved with the ASME Code, Section XI, Appendix G, process of determining P/T limits remain the same.

Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of the P/T operating limits curve is more technically accurate than the K_{IA} curve since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel.

Safety Evaluation

The proposed P/T limit curves were developed in accordance with the 1989 Edition ASME Section XI, Appendix G; 10 CFR 50, Appendix G; and ASME Code Case N-640. Code Case N-640 permits the use of an alternate reference fracture toughness for reactor vessel materials in determining the P/T limits. Specifically, Code Case N-640 allows use of the K_{IC} fracture toughness curve shown in ASME Code, Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness in the development of the P/T limit curves. The other margins involved with the ASME Code, Section XI, Appendix G, process of determining P/T limits remain the same.

Use of Code Case N-640 is justified based upon the knowledge gained in the industry since the fracture toughness curve was created in 1974. Since that time, additional knowledge of the fracture toughness of materials and their response to applied loads has been acquired. This additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. Use of the K_{IC} fracture toughness curve in developing P/T limits provides additional operating margin for the P/T curves, thus realizing significant benefits primarily for the pressure test. Use of the revised curves will result in a reduction in the challenges to operators in maintaining a high temperature in a limited operating window and would eliminate steam vapor hazards by allowing primary containment inspections to be conducted at a lower coolant temperature.

The changes to the calculational methodology for the P/T limits based upon ASME Code Case N-640 continue to provide adequate margin in the prevention of a non-ductile type fracture of the RPV.

The values of adjusted reference temperature and upper shelf energy are expected to remain within the limits of Regulatory Guide 1.99, Revision 2 and Appendix G of 10 CFR 50 (less than 200°F and greater than 50 ft-lbs respectively).

Therefore, based on the above, NMC has concluded that the proposed revision to the Monticello TS is acceptable.

Determination of No Significant Hazards Consideration

The proposed changes to the Monticello Technical Specifications will revise the current pressure-temperature (P/T) limits curves. The proposed P/T limits rely on the methodology for determining allowable P/T limits specified in American Society of Mechanical Engineers (ASME) Code Case N-640. The revised P/T limits will allow required hydrostatic and leak tests to be performed at a significantly lower temperature. This expected to reduce challenges to plant operators associated with maintaining the reactor coolant system within a narrow temperature band during testing. The proposed amendment has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10 CFR Part 50, Section 50.91, using standards provided in Section 50.92. This analysis is provided below:

1. *The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the ASME Code and 10 CFR 50 Appendix G and H as restrictions on operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary.

The changes to the calculation methodology for the P/T limits are based upon ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, division 1," and provide adequate margin in the prevention of a non-ductile type fracture of the reactor pressure vessel (RPV). The code case was developed based upon the knowledge gained through years of industry experience. The P/T limits developed using the allowances of ASME Code Case N-640 provide more operating margin. However, experience gained in the areas of fracture toughness of materials and pre-existing undetected defects shows that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, use of the allowances of ASME Code Case N-640 in developing the P/T limits will provide adequate protection against nonductile-type fractures of the RPV.

Development of the revised Monticello P/T limits was performed using the approved methodologies of 10 CFR 50, Appendix G, and using the allowances of ASME Code Case N-640. The P/T limit curves generated using these methods ensure the P/T limits will not be exceeded during any phase of reactor operation. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed change will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events.

Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

2) *The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposed changes provide more operating margin in the P/T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with benefits being primarily realized during the pressure tests. Operation in the "new" regions of the newly developed P/T curves has been analyzed in accordance with the provisions of ASME Code, Section XI, Appendix G; 10 CFR 50 Appendix G, and ASME Code Case N-640, thus providing adequate protection against a nonductile-type fracture of the RPV.

The proposed changes do not alter any existing system relationships. The proposed changes do not result in any new or unanalyzed operation of any system or piece of equipment important to safety, and as a result, the possibility of a new type event is not created.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) *The proposed amendment will not involve a significant reduction in the margin of safety.*

As mentioned previously, the revised P/T limit curves provide more operating margin and thus, more operational flexibility than the current P/T limit curves. With the increased operational margin, a reduction in the safety margin results with respect to the existing curves. However, industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T limit curves are unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N-640 takes into account the acquired knowledge and establishes more realistic methodologies for the development of P/T limit curves.

Use of ASME Code Case N-640 to develop the revised P/T curves utilized the K_{IC} fracture toughness curve in lieu of the K_{IA} curve as the lower bound for fracture toughness. Use of the K_{IC} curve to determine lower bound fracture toughness is more technically correct than using the K_{IA} curve. P/T curves based on the K_{IC} fracture toughness limits enhance overall plant safety by expanding the P/T window in the low-temperature operating region. The benefits which occur are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment with no decrease to the margin of safety. Therefore, operational flexibility is gained and an acceptable margin of safety to RPV non-ductile type fracture is maintained.

Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

Based upon the evaluation described above and pursuant to 10 CFR Part 50, Section 50.91, Nuclear Management company, LLC (NMC) has determined that operation of the Monticello Nuclear Generating Plant in accordance with the proposed amendment request does not involve a significant hazards consideration as defined in 10 CFR Part 50, Section 50.92.

Environmental Assessment

NMC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration.

2. The changes do not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite.

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed changes is not required.

In Support of License Amendment Request
Revised Pressure and Temperature Curves

Request for Exemption from the
Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G

In accordance with 10 CFR 50.12, Nuclear Management Company, LLC (NMC) hereby requests approval of an exemption request from specific requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation" and 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." The requested exemption will permit the use of American Society of Mechanical Engineers (ASME) Code, Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b.

Justification for the Use of Code Case N-640

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

10 CFR 50.12(a) states that the Commission may grant exemptions from the requirements of the regulations of this part, which meet the following criteria:

- *The requested exemption is authorized by law:*

No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

- *The requested exemption does not present an undue risk to the public health and safety:*

New P/T limits were developed for the Monticello Nuclear Generating Plant using the methodologies in Code Case N-640, in lieu of 10 CFR Part 50, Appendix G, paragraph IV.A.2.b. This exemption is needed to allow the use of these new P/T limit curves in the Monticello Technical Specifications (TS).

Code Case N-640 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of K_{IA} fracture toughness curve) for reactor vessel materials in determining the P/T limits. The K_{IC} fracture toughness curve is shown in

ASME Code, Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve), and provides greater allowable fracture toughness than the corresponding K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1 (the K_{IA} fracture toughness curve). The other margins involved with the ASME Code, Section XI, Appendix G process of determining P/T limit curves remain unchanged.

Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of the P/T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor vessel. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel.

Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel (RPV) material fracture toughness. Since 1974, additional knowledge about the fracture toughness of vessel materials and their fracture response to applied loads has been gained. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{IC} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety.

P/T limit curves based on the K_{IC} fracture toughness limits will enhance overall plant safety by opening the pressure and temperature operating window. Since the reactor coolant system (RCS) P/T operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Code, Section XI, Appendix G procedure, continued operation of Monticello with these P/T limit curves without the relief provided by ASME Code Case N-640 would unnecessarily require the RPV to maintain a temperature exceeding 212 degrees Fahrenheit in a limited operating window during the pressure test. Consequently, steam vapor hazards would continue to be one of the safety concerns for personnel conducting inspections in primary containment.

Use of the revised curves would result in a reduction in the challenges to operators in maintaining a high temperature in a limited operating window and would eliminate steam vapor hazards by allowing inspections in primary containment to be conducted at lower coolant temperature, while continuing to provide an adequate margin of safety.

- *The requested exemption will not endanger the common defense and security:*

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

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

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

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

(a)(2)(iii) - will result in undue hardship or other cost that are significant if the regulation is enforced, and

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

10CFR50.12(a)(2)(ii): ASME Code, Section XI, Appendix G, provides procedures for determining allowable loading on the RPV and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P/T operating and test curves satisfies the underlying requirement that:

1. The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a ductile manner and the probability of a rapidly propagating fracture is minimized; And

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

The ASME Code, Section XI, Appendix G, procedure was conservatively developed based upon the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge concerning these topics has greatly expanded. This increased knowledge permits relaxation of the ASME Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety.

10 CFR 50.12(a)(2)(iii): The reactor coolant system (RCS) pressure-temperature operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Code, Section XI, Appendix G procedure. Continued operation of the Monticello Nuclear Generating Plant with these P/T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P/T operating window. This restriction requires the Operations staff to maintain a high temperature during pressure tests and also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P/T limit curves.

Implementation of the proposed P/T limit curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety.

10CFR50.12(a)(2)(v): The requested exemption provides only temporary relief from the applicable regulation and Monticello Nuclear Generating Plant has made a good faith effort to comply with the regulation. Therefore, NMC requests the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry.

Code Case N-640 Conclusion for Exemption Acceptability

Compliance with the specified requirement of 10 CFR 50.60(a) will result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the lower bound fracture toughness used by ASME Code, Section XI, Appendix G, in the determination of reactor coolant P/T limits. This proposed alternative is acceptable, because the ASME Code Case maintains the relative margin of safety commensurate with the margin of safety that existed at the time ASME Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for the Monticello Nuclear Generating Plant ensures an acceptable margin of safety and does not present an undue risk to the public health and safety.

Exhibit C

License Amendment Request for Revision to

Technical Specifications Pressure and Temperature Curves

Current Monticello Technical Specification Figures
Marked Up With Proposed Change

This Exhibit consist of current Monticello Technical Specification figures marked up with the proposed changes. The figures included in the exhibit are listed below:

Figures

3.6-2
3.6-3
3.6-4

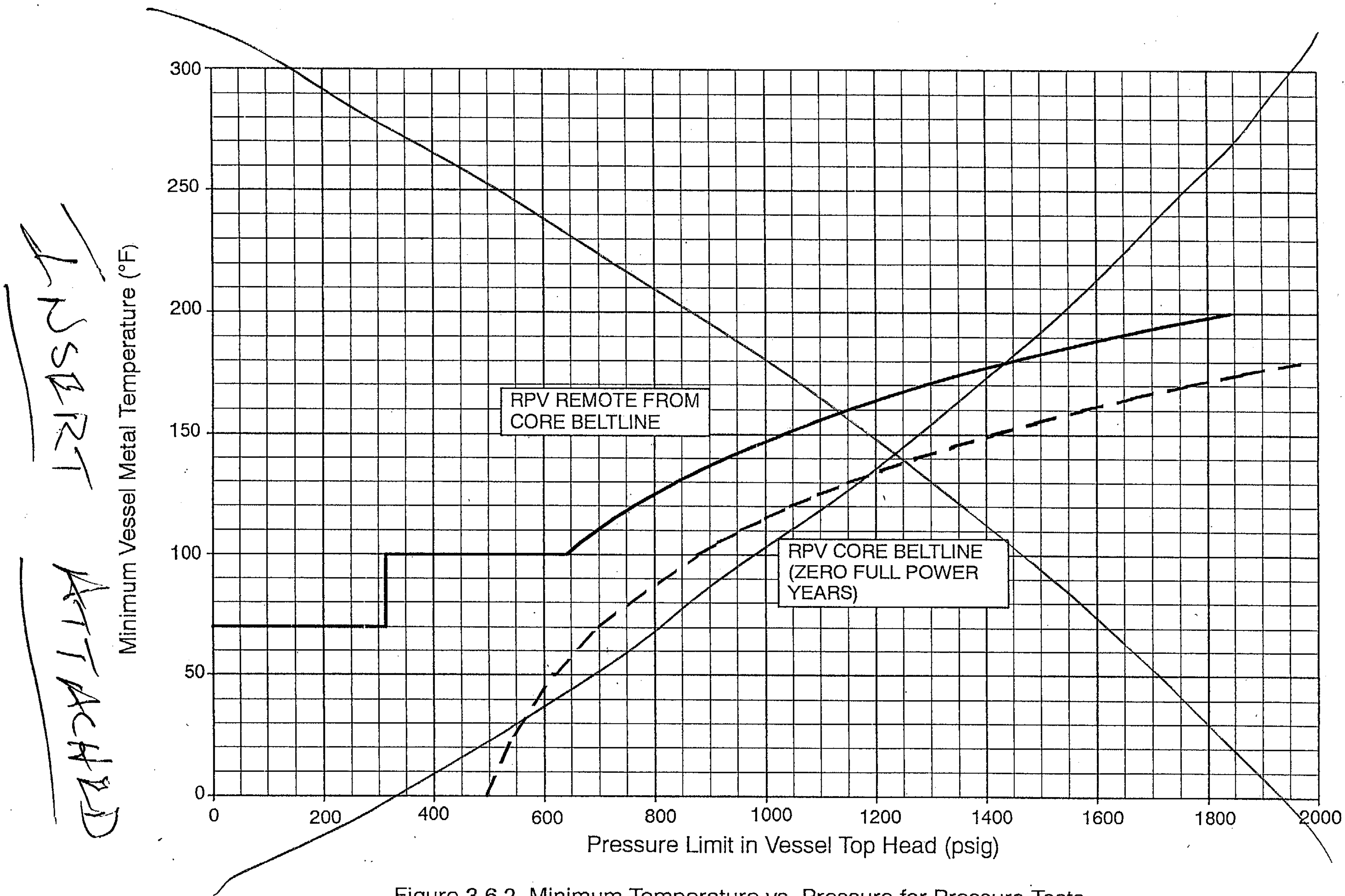

Figure 3.6.2 Minimum Temperature vs. Pressure for Pressure Tests

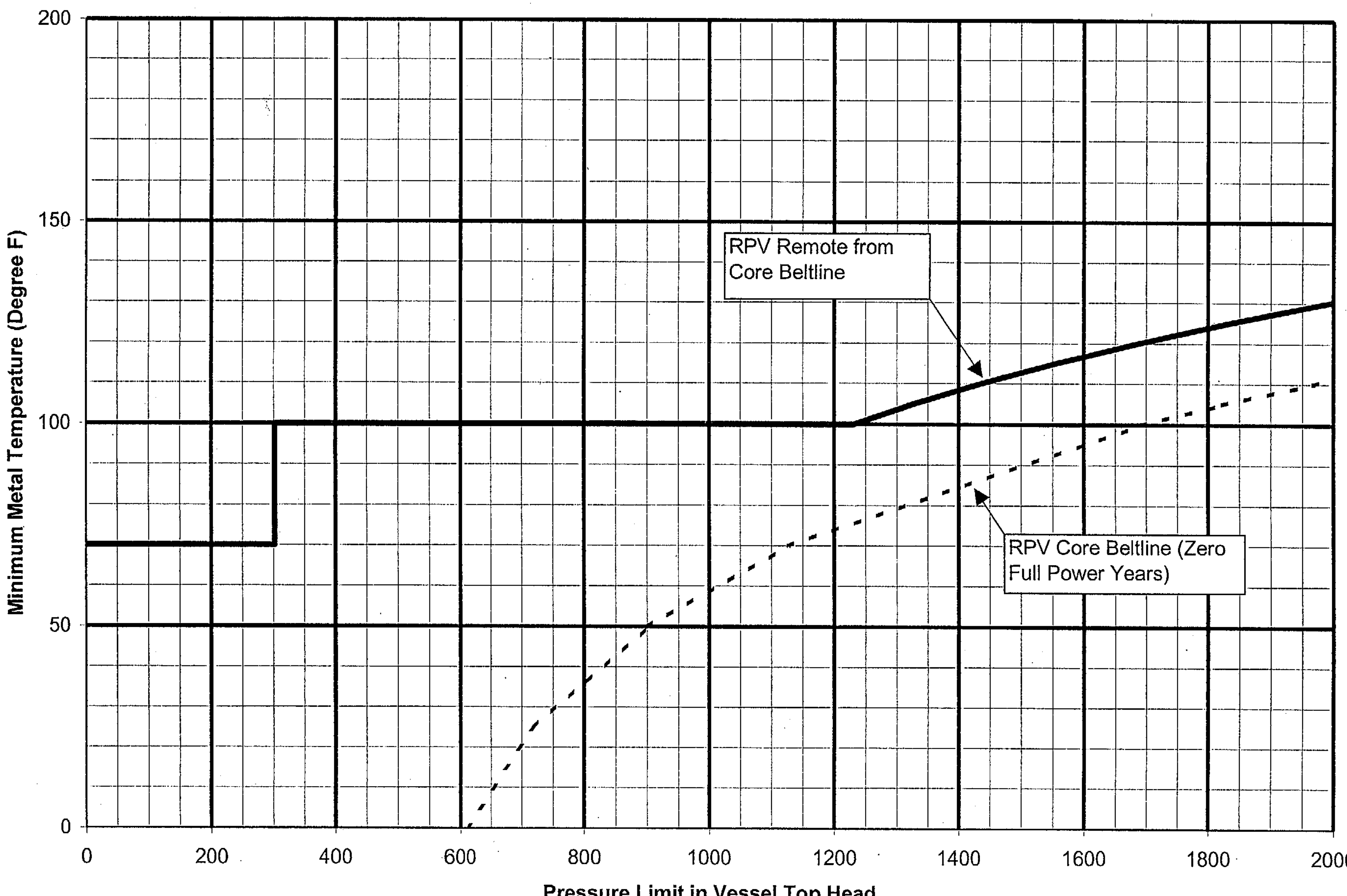

Figure 3.6.2 Minimum Temperature vs. Pressure For Pressure Tests

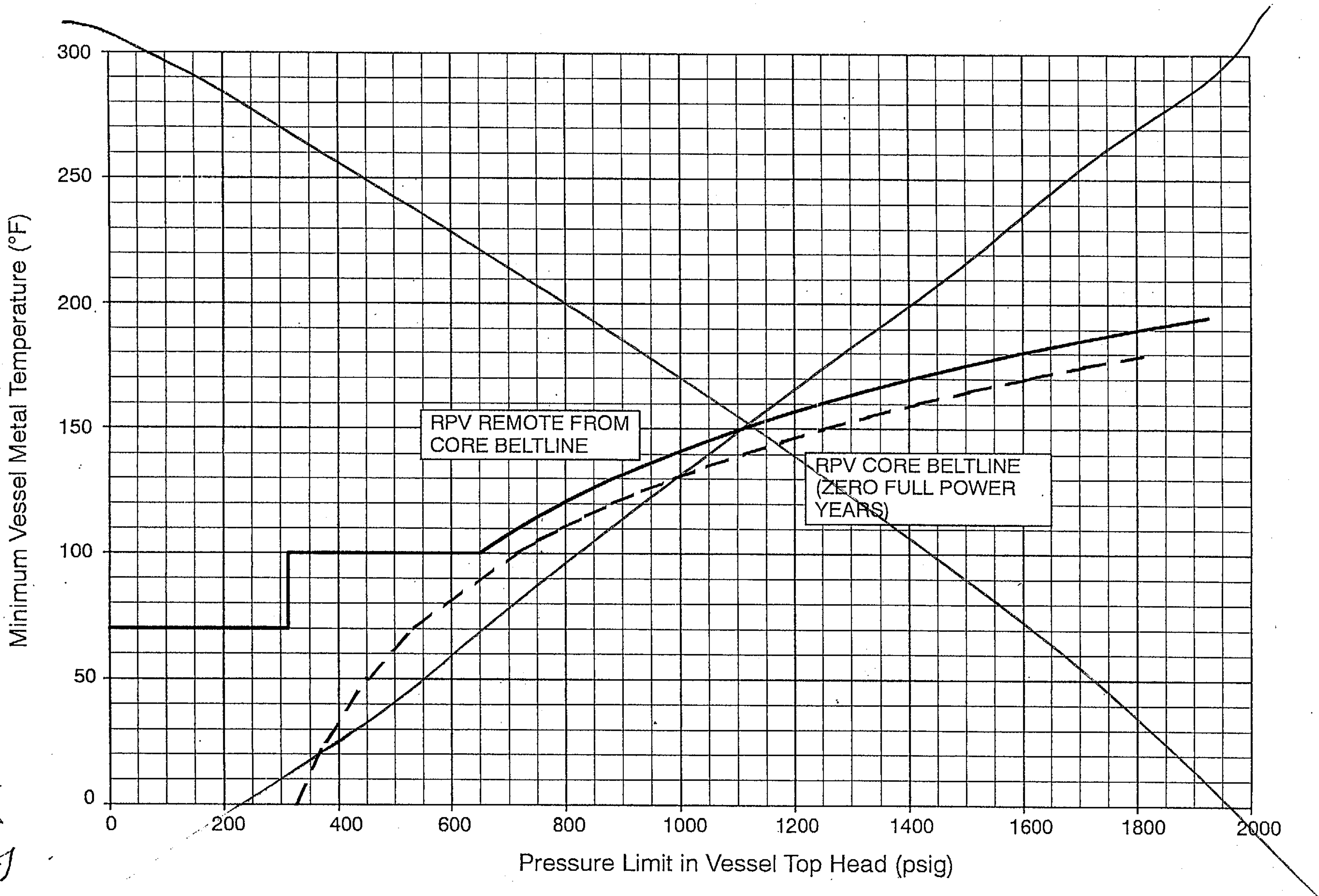

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Figure 3.6.3 Minimum Temperature vs. Pressure Mechanical Heatup or Cooldown Without the Core Critical

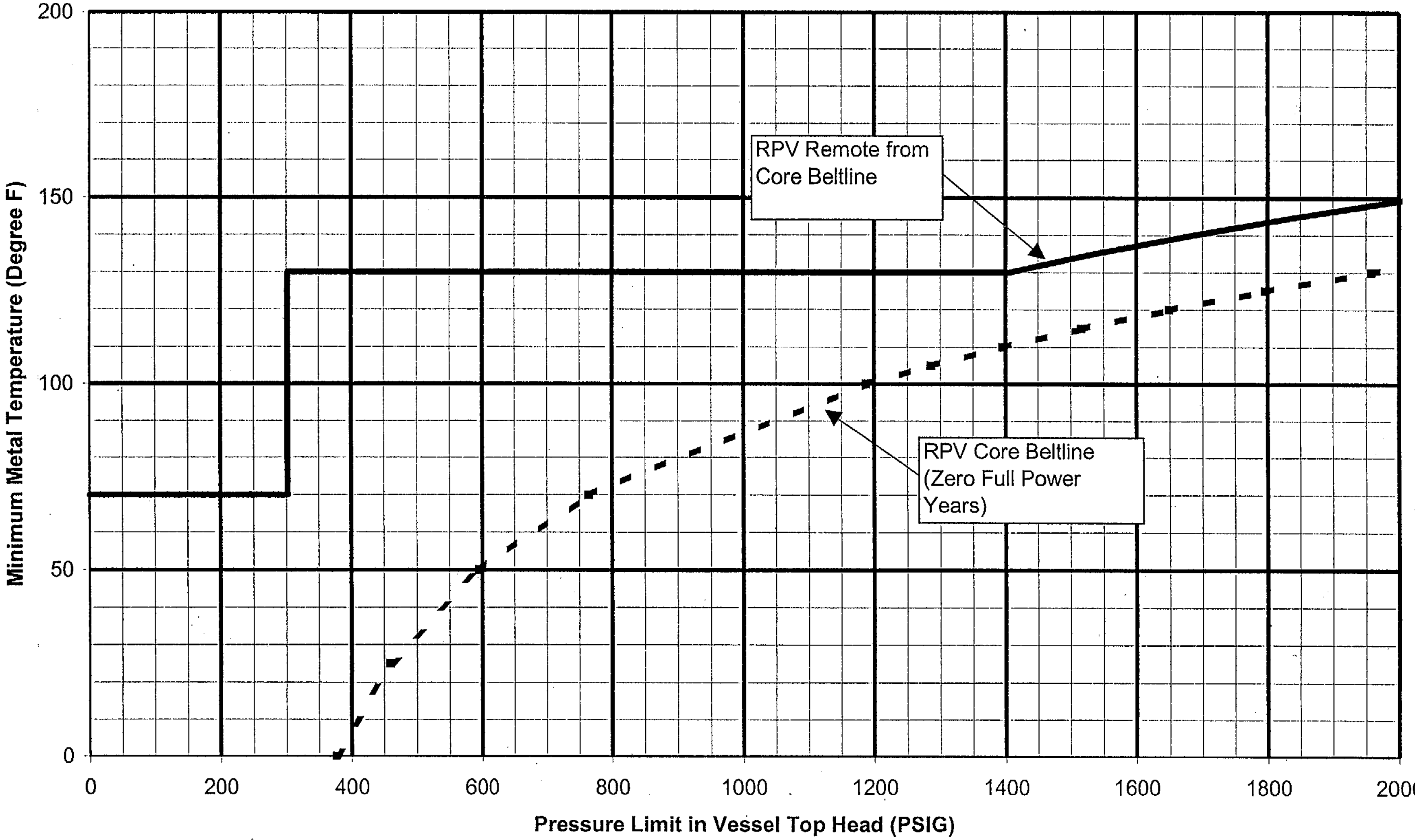

Figure 3.6.3 Minimum Temperature vs. Pressure Mechanical Heatup or Cooldown without the Core Critical

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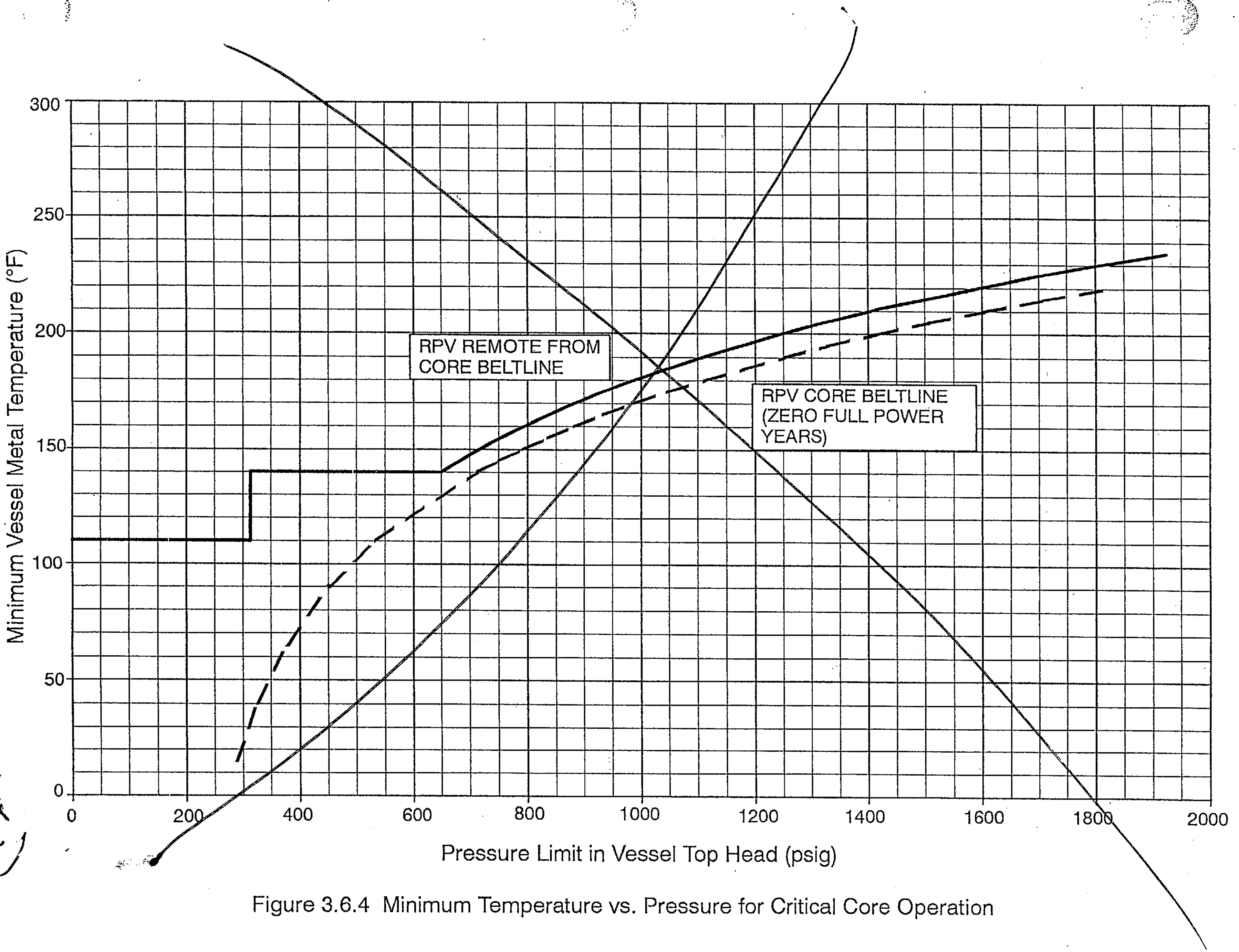

Figure 3.6.4 Minimum Temperature vs. Pressure for Critical Core Operation

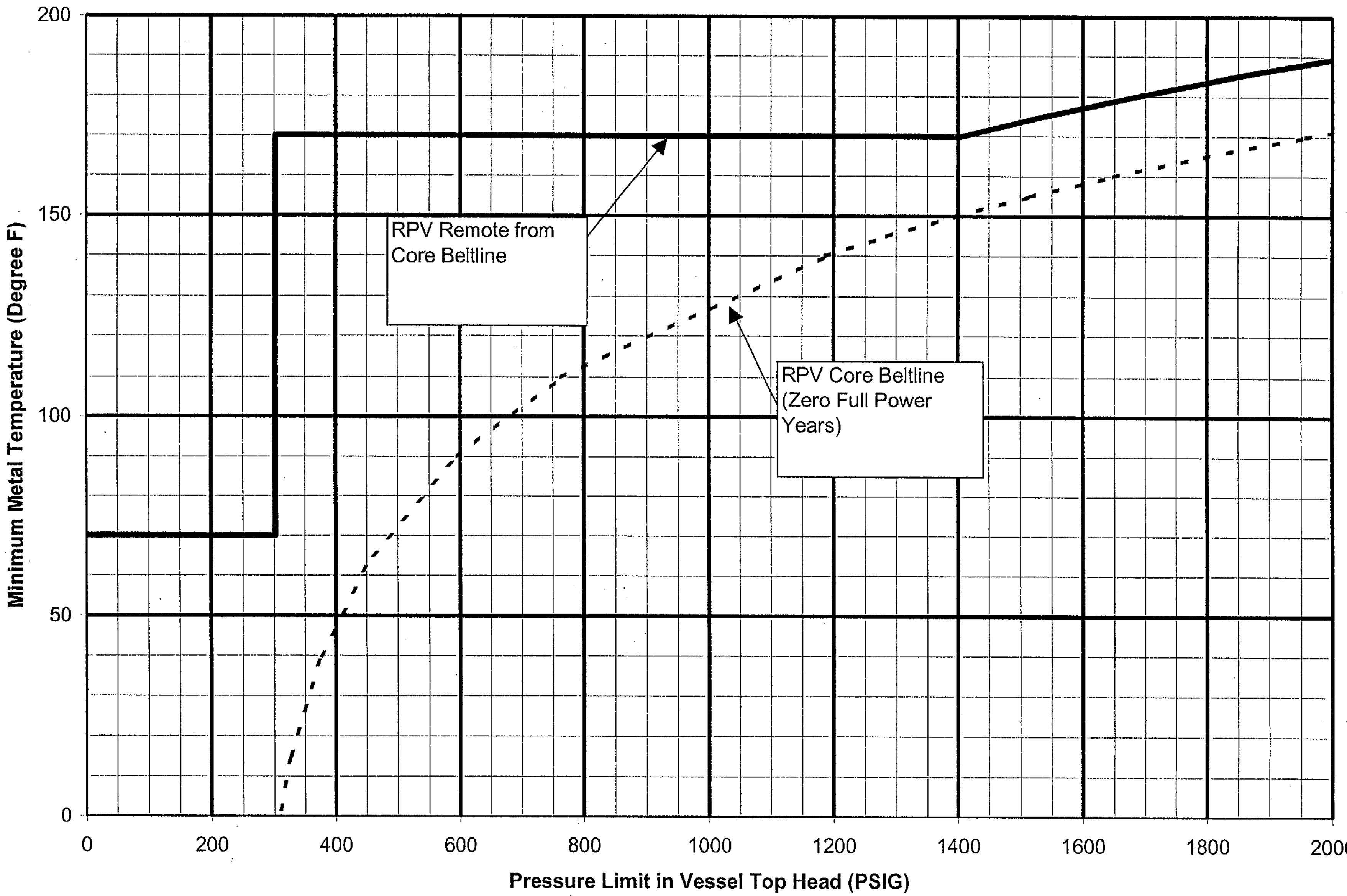

Figure 3.6.4 Minimum Temperature Vs. Pressure for Core Critical Operation

Exhibit D

License Amendment Request for Revision to

Technical Specifications Pressure and Temperature Curves

Revised Monticello Technical Specification Figures

This Exhibit consist of revised Monticello Technical Specification figures that incorporate the proposed changes. The figures included in the exhibit are listed below:

Figures

3.6-2
3.6-3
3.6-4

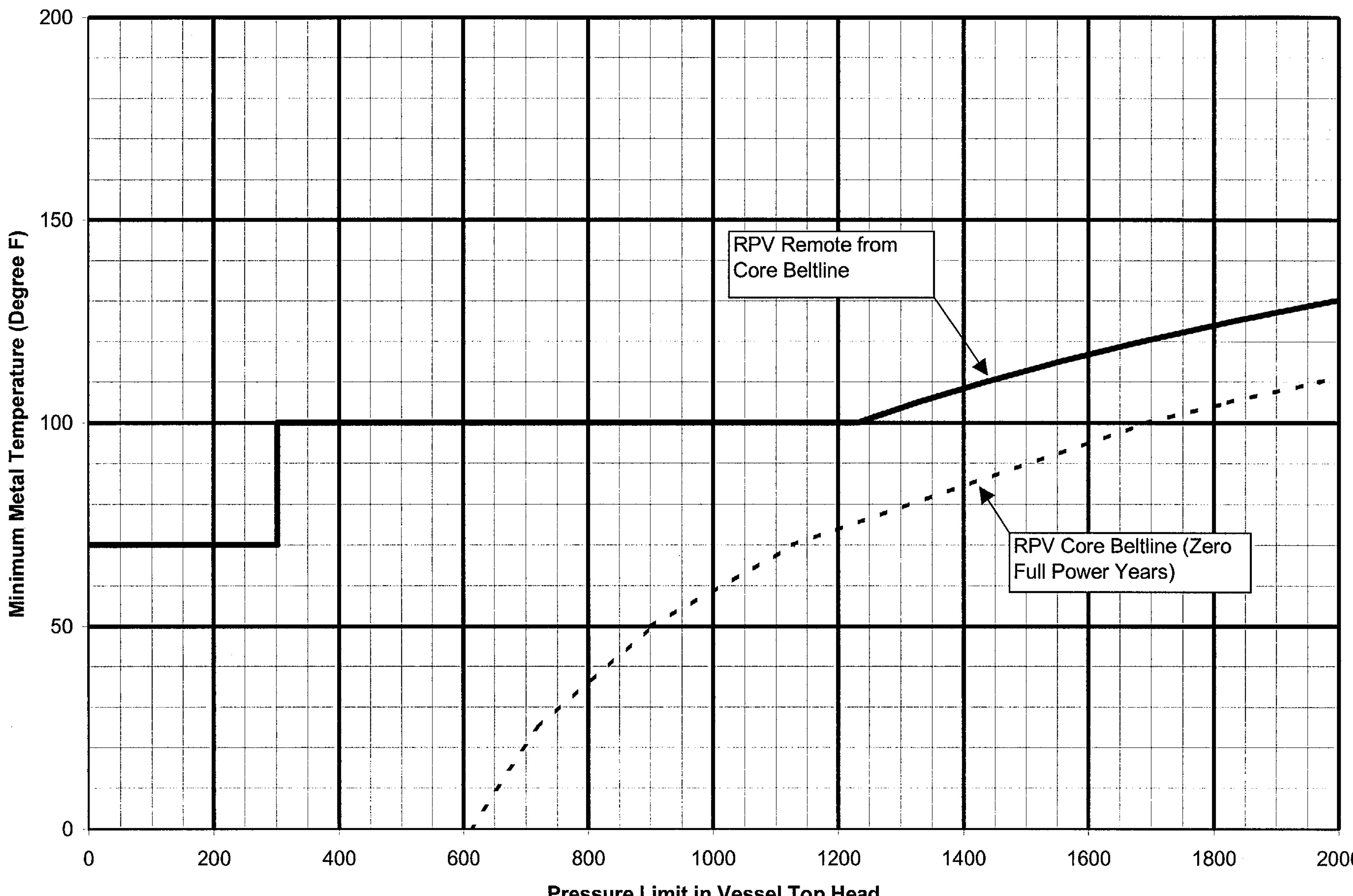

Figure 3.6.2 Minimum Temperature vs. Pressure For Pressure Tests

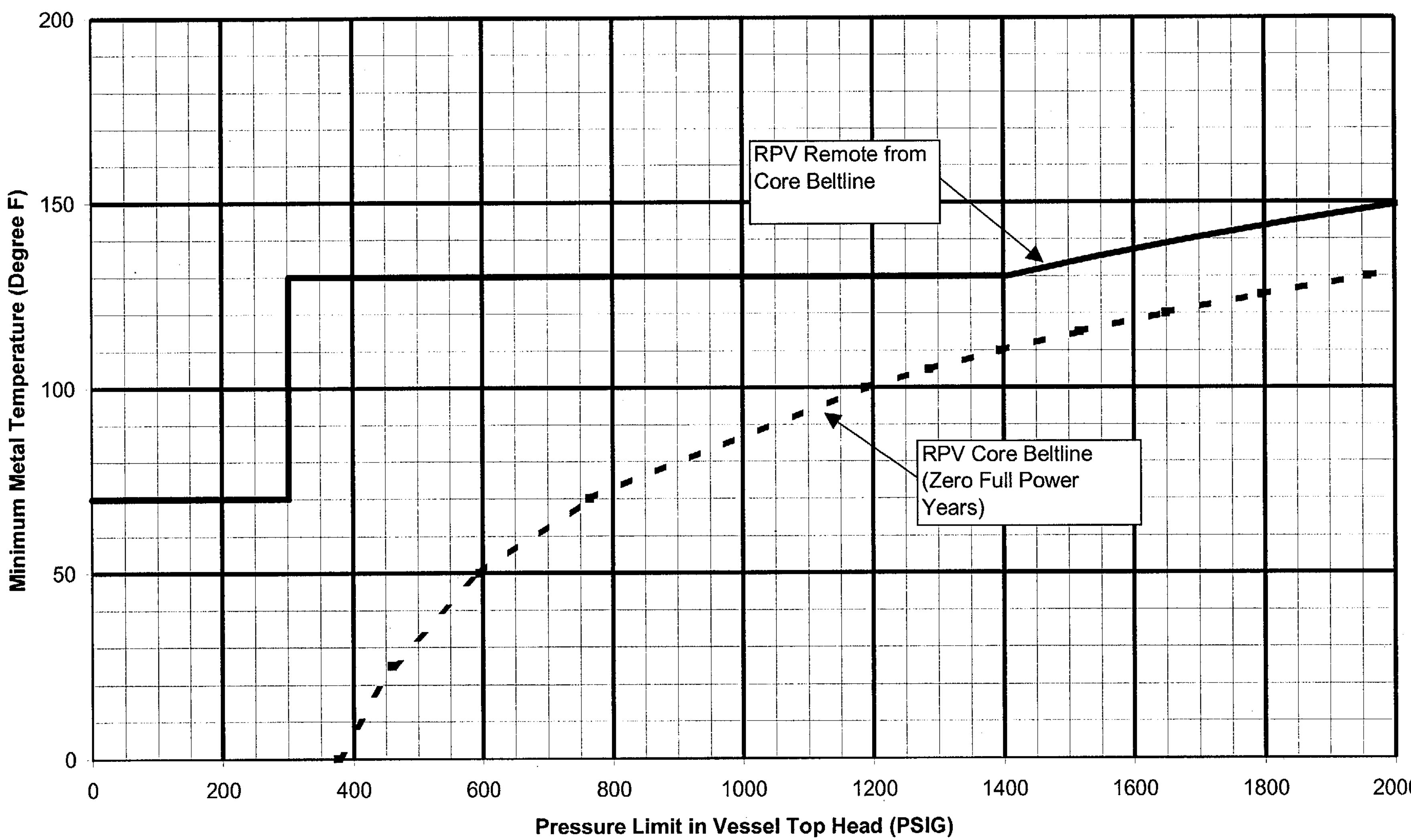

Figure 3.6.3 Minimum Temperature vs. Pressure Mechanical Heatup or Cooldown without the Core Critical

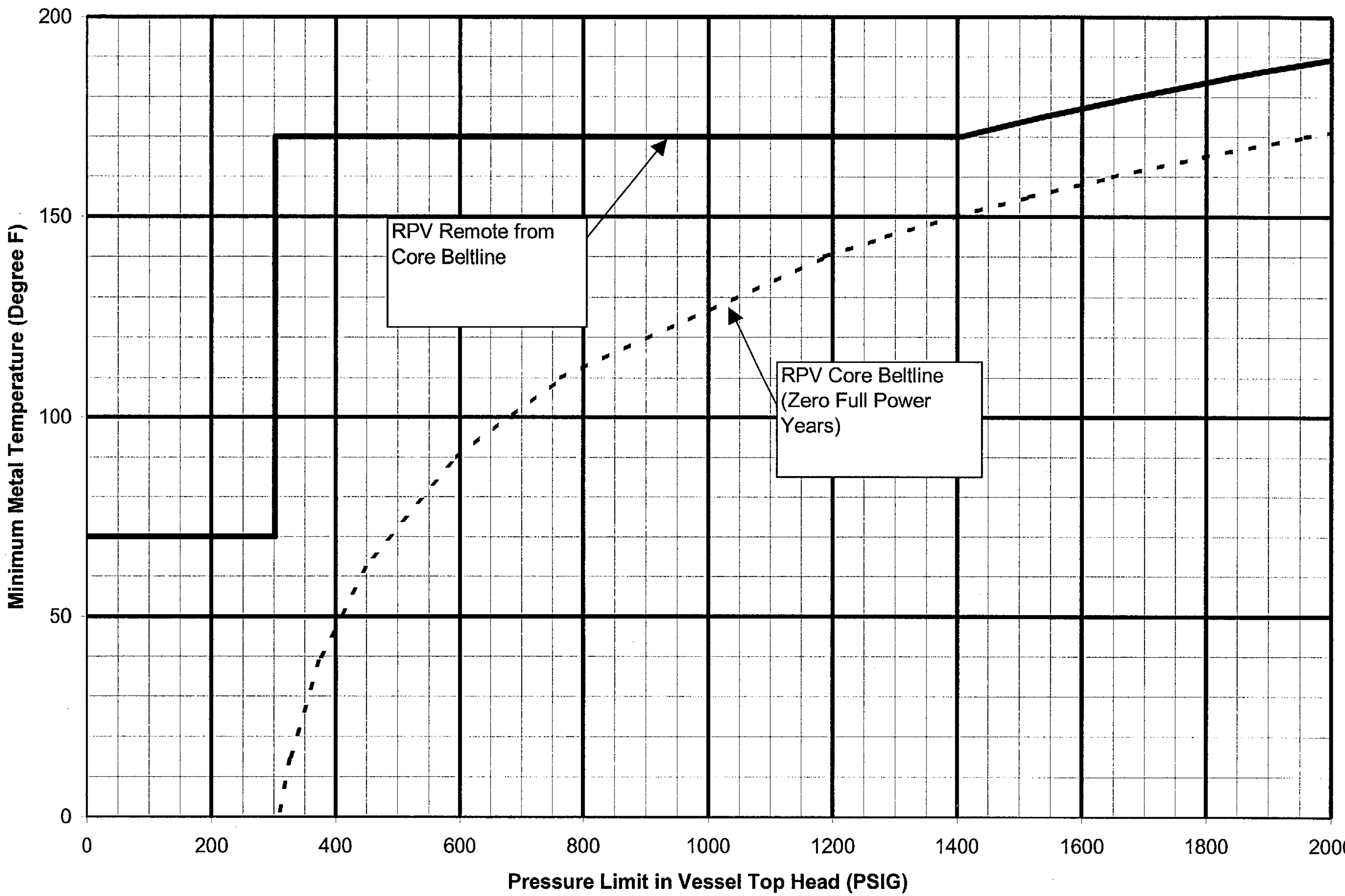

Figure 3.6.4 Minimum Temperature Vs. Pressure for Core Critical Operation