Exhibit B

License Amendment Request Dated March 31, 1989
Docket No. 50-263 License No. DPR-22

Exhibit B consists of revised pages for the Monticello Nuclear Generating Plant Technical Specifications annotated to show the proposed wording changes:

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229ff (new page)

LIST OF FIGURES

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B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1, except for the reactor vessel bottom head, shall be at or above the temperatures shown on the two curves of Figure 3.6.2, where the dashed curve, "RPV Core Beltline," is increased by the core beltline temperature adjustment from Figure 3.6.1. The reactor vessel bottom head temperature shall be at or above the temperatures shown on the solid curve of Figure 3.6.2, "RPV Remote from Core Beltline," with no adjustment from Figure 3.6.1.

2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

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B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes.

   a. Reactor vessel shell adjacent to shell flange.

   b. Reactor vessel bottom head.

   c. Reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region.

2. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E 185-66. Samples shall be withdrawn at one fourth and three fourths service life. Analysis of the first sample shall include a quantitative determination of the material chemistries. (Note: Analysis of the first sample has been completed. The Figure 3.6.1 core beltline temperature adjustment curve reflects the chemistry data obtained).

3. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated value of neutron fluence at one fourth of the beltline shell thickness that is used to determine the NDTT shift from Figure 3.6.1.

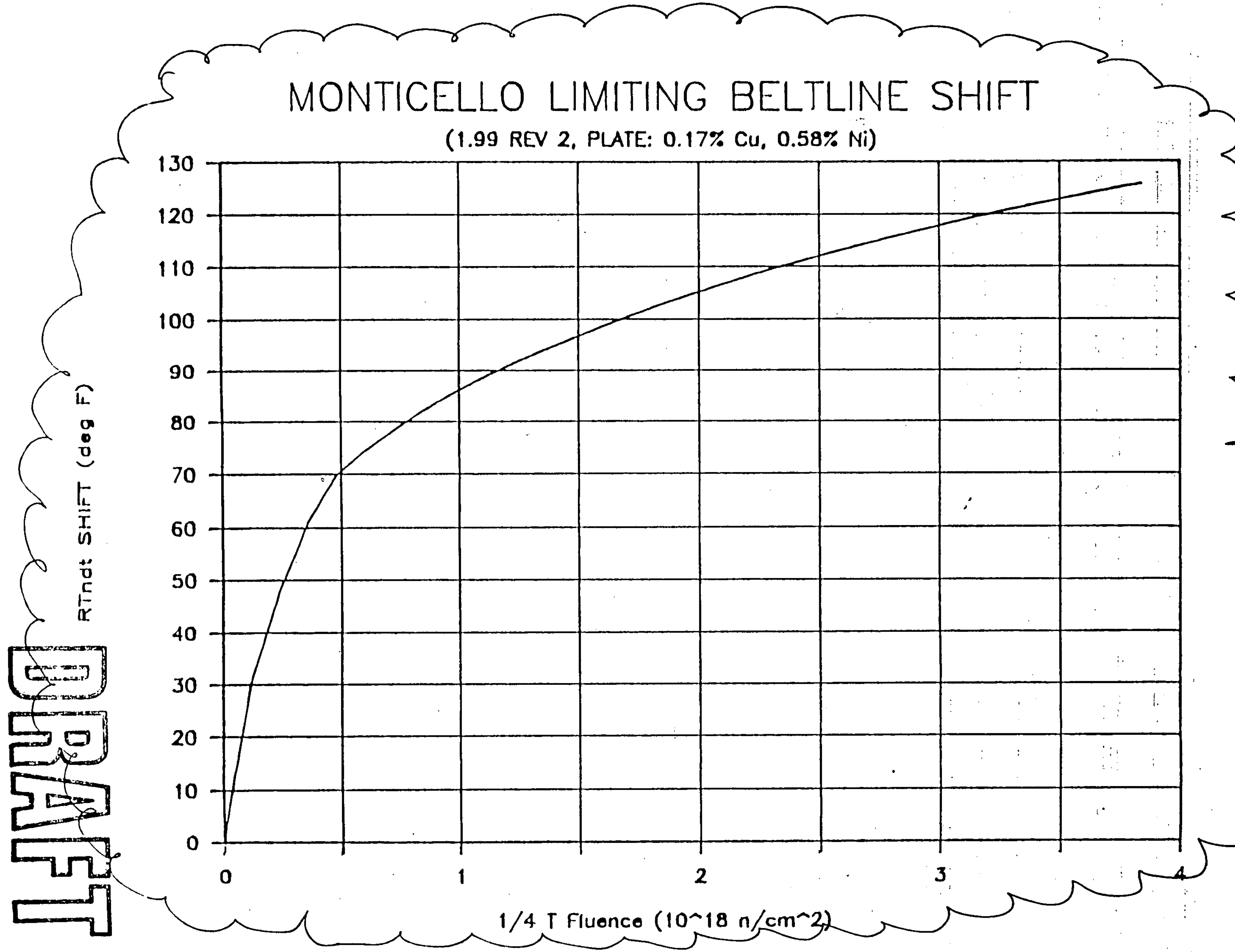

Figure 3.6.1 Core Beltline Operating Limits Curve Adjustment vs. Fluence

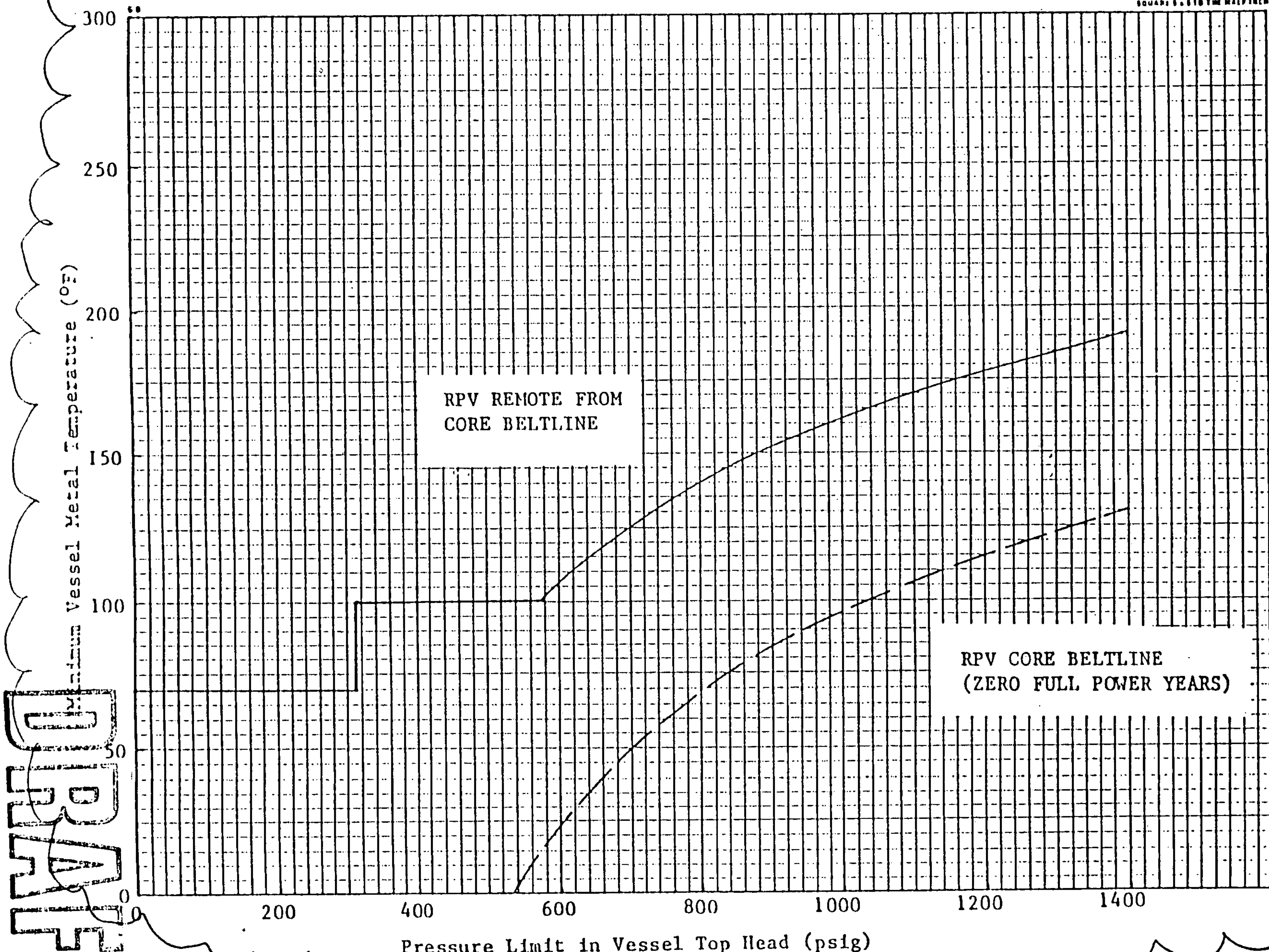

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Figure 3.6.2 Minimum Temperature vs. Pressure for Pressure Tests

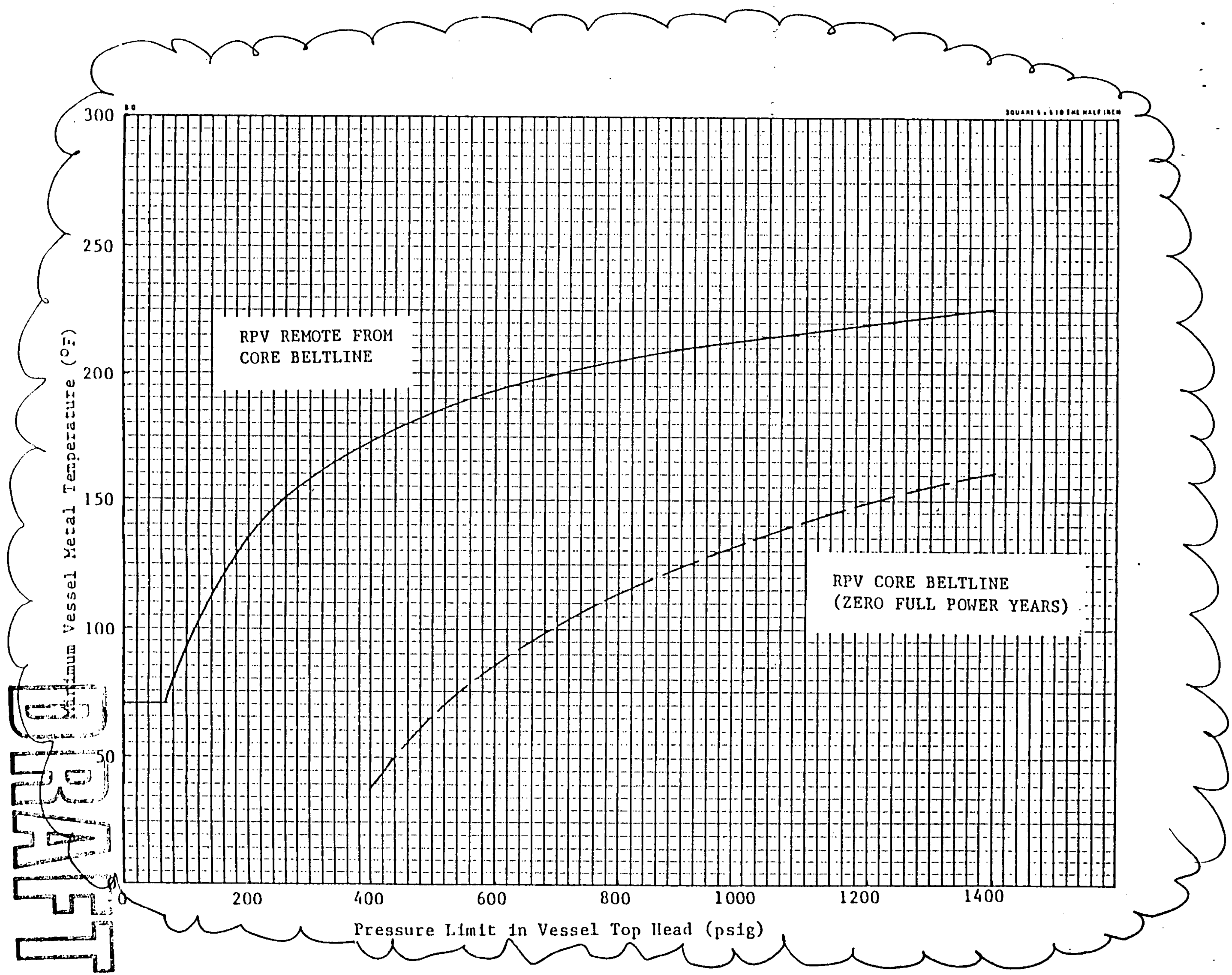

Figure 3.6.3 Minimum Temperature vs. Pressure for Mechanical Heatup or Cooldown Without the Core Critical

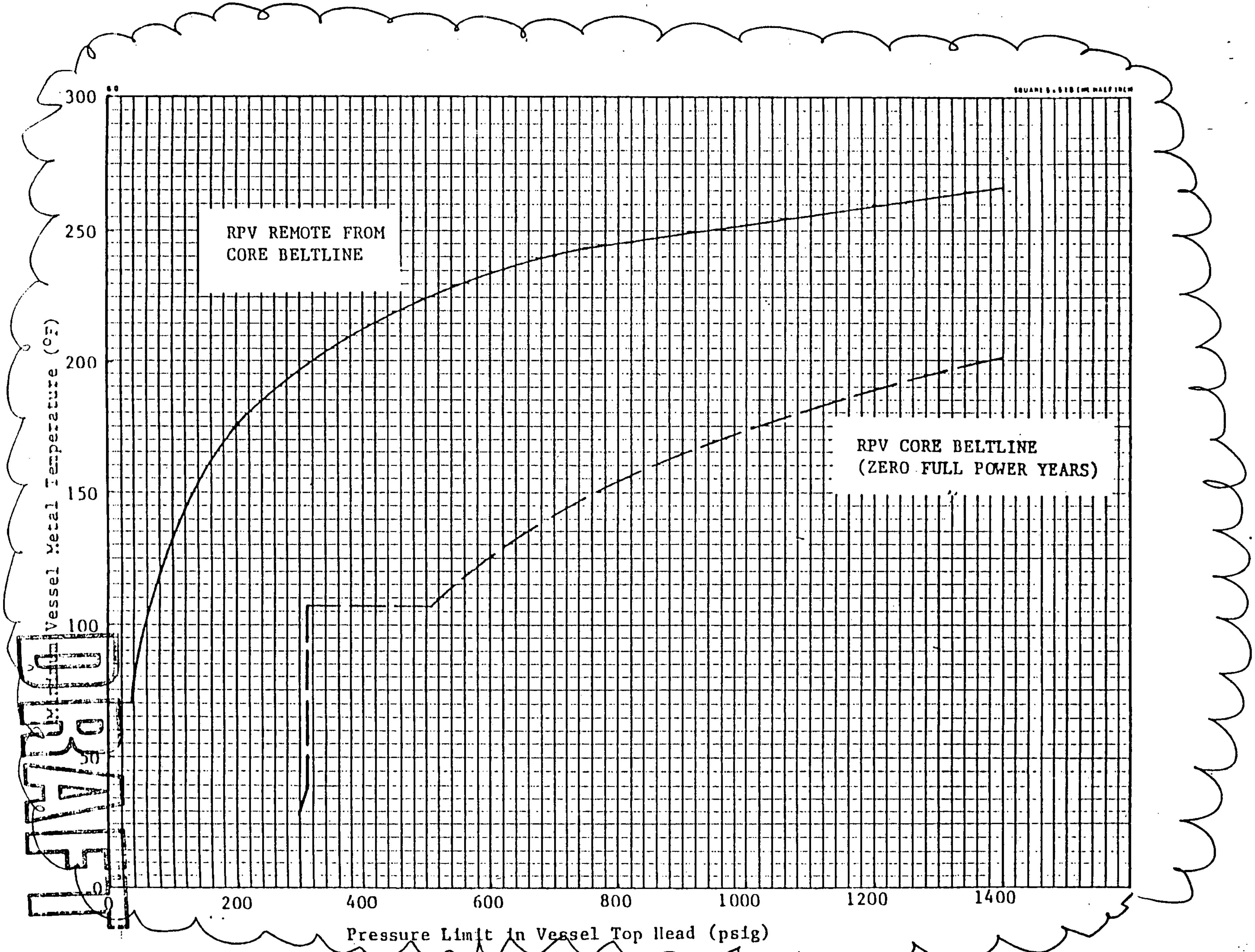

Figure 3.6.4 Minimum Temperature vs. Pressure for Core Operation

Bases 3.6 and 4.6:

A. Reactor Coolant Heatup and Cooldown

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the pump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel thermal and pressure transients.

During hydrostatic pressure testing, a coolant heatup or cooldown of 20°F in any one-hour period has a negligible effect on the reactor operating limits of Figure 3.6.2.

B. Reactor Vessel Temperature and Pressure

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown and during inservice hydrostatic testing were established using 10 CFR Part 50, Appendix G, May 1983 and Appendix G of the Summer 1976 or later Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that a large postulated surface flaw, having a depth of 0.24 inches at the flange-to-vessel junction and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions can be safety accommodated. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1966 Addenda).

A General Electric Company procedure, designed to evaluate fracture toughness requirements for older plants where information may be incomplete, was used to estimate RT_{NDT} values on an equivalent basis to the new requirements for plants which have construction permits after August 15, 1973.

Bases 3.6 and 4.6 - Continued:

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials give in regulatory Guide 1.99, Revision 2, was used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown following nuclear shutdown), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. An analysis and report will be submitted to the Commission on all such surveillance specimens removed from the reactor vessel in accordance with 10 CFR 50, Appendix H, including information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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b. When Primary Containment Integrity is required, leakage rates shall be limited to:

1. An overall integrated leakage rate of less than or equal to La, 1.2 percent by weight of the containment air per 24 hours at Pa, 42 psig.

2. A combined leakage rate of less than or equal to 0.6La for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to Pa.

3. Less than or equal to 11.5 scf per hour for any one main steam isolation valve when tested at 25 psi.

With the measured overall integrated primary containment leakage rate exceeding 0.75La, or the measured combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C testing exceeding 0.6La, or the measured leak rate exceeding 11.5 scf per hour for any one main steam isolation valve, restore leakage rates to less than or equal to these values prior to increasing reactor coolant system temperature above 212°F or, alternatively, restore measured leakage rates to within these limits within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

b. The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods, and provisions of 10 CFR Part 50:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at 40±10 month intervals* during shutdown at ≥Pa during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If any periodic Type A test fails to meet 0.75La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75La, at which time the above test schedule may be resumed.

3. All Type A test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

*The second test of the second 10-year service period may be conducted during the 1989 refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	2. Welds in austenitic stainless steel piping four inches or larger in diameter containing reactor coolant at a temperature above 200 degrees F during power operation, including reactor vessel attachments and appurtenances, not meeting the requirements of NUREG-0313, Revision 2, for IGSCC Category A weldments shall be included in an augmented inspection program meeting the requirements of NUREG-0313, Revision 2.

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Exhibit C

License Amendment Request Dated March 31, 1989
Docket No. 50-263 License No. DPR-22

Exhibit C consists of retyped pages for the Monticello Nuclear Generating Plant Technical Specifications showing the final form of the proposed wording changes:

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146
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229ff (new page)

LIST OF FIGURES

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

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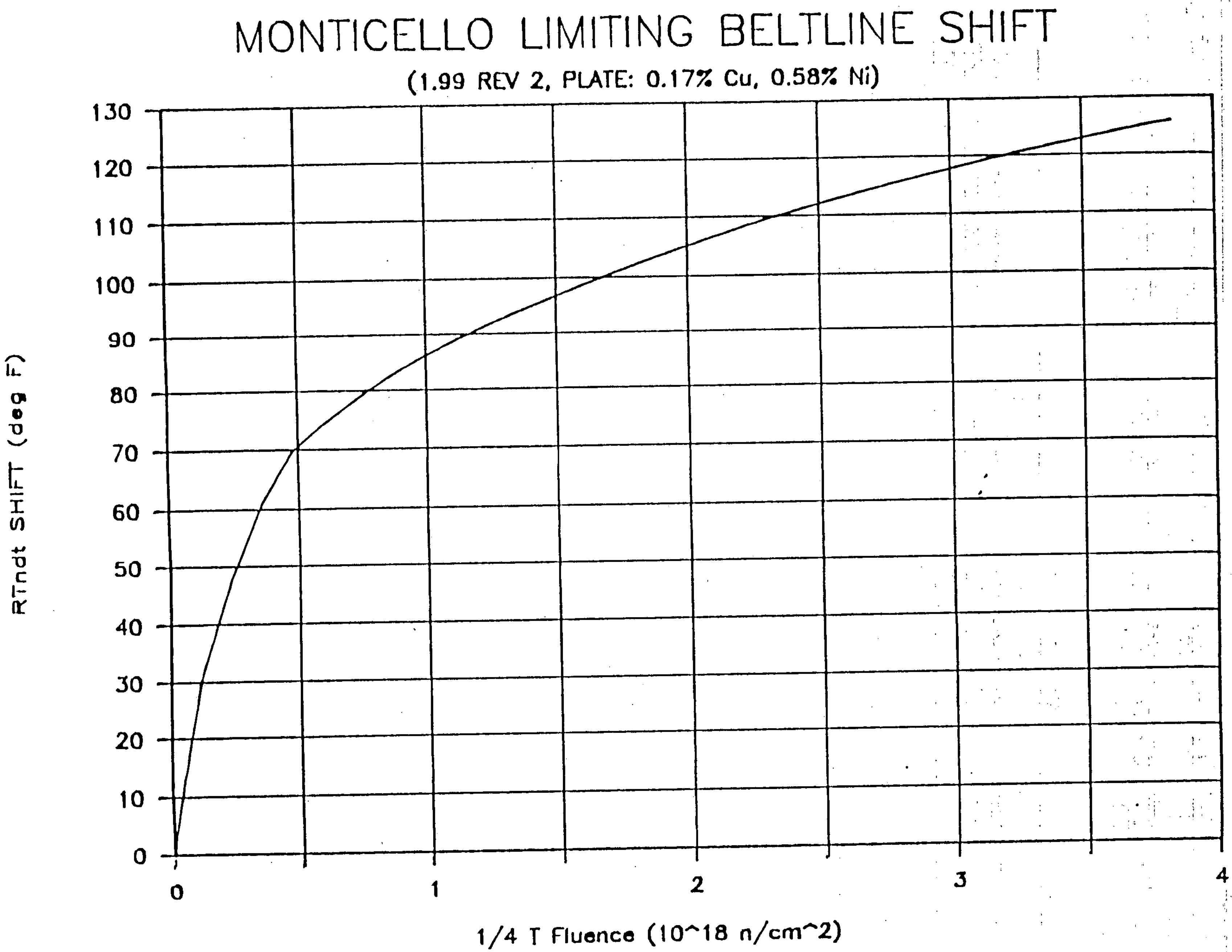

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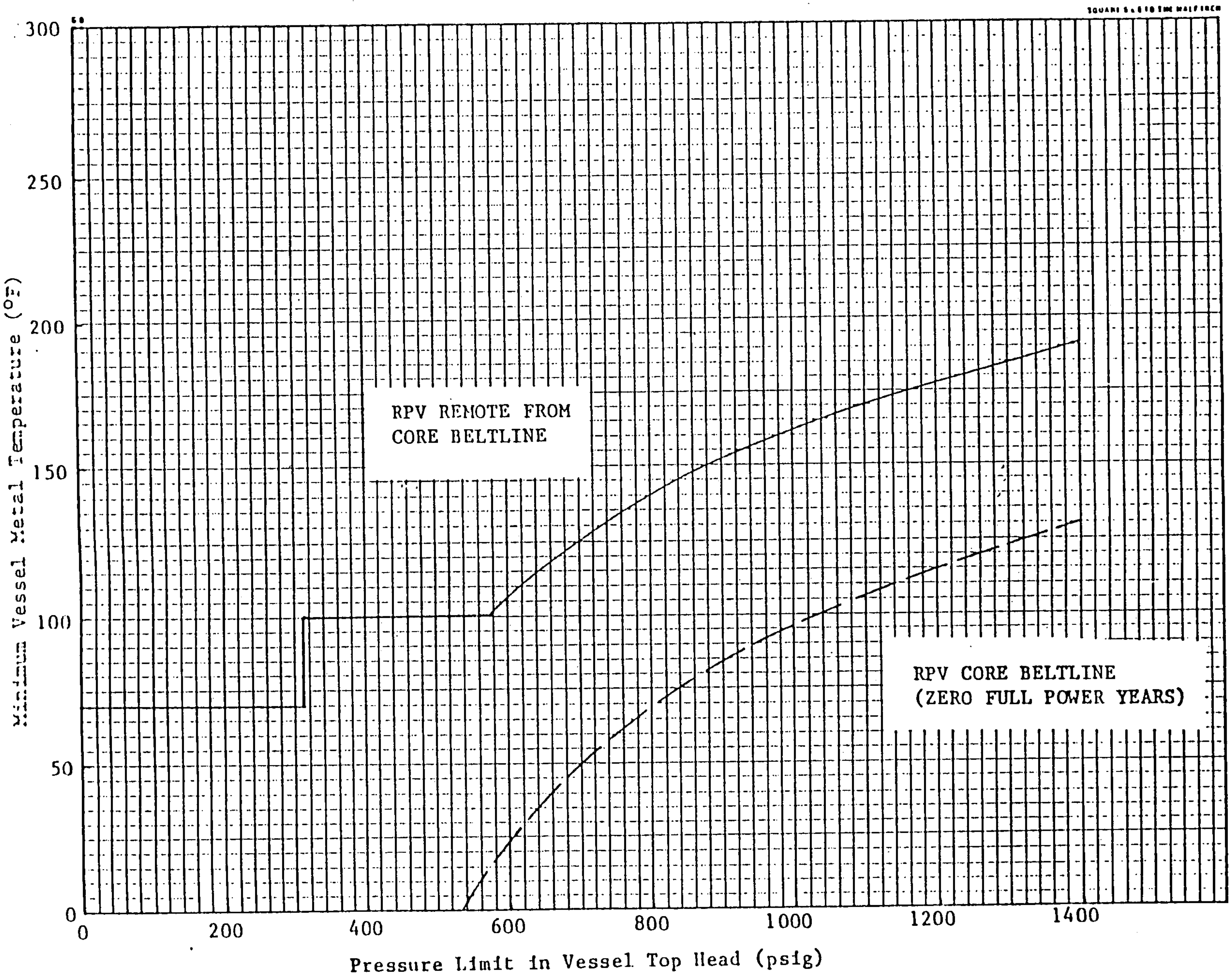


Figure 3.6.2 Minimum Temperature vs. Pressure for Pressure Tests

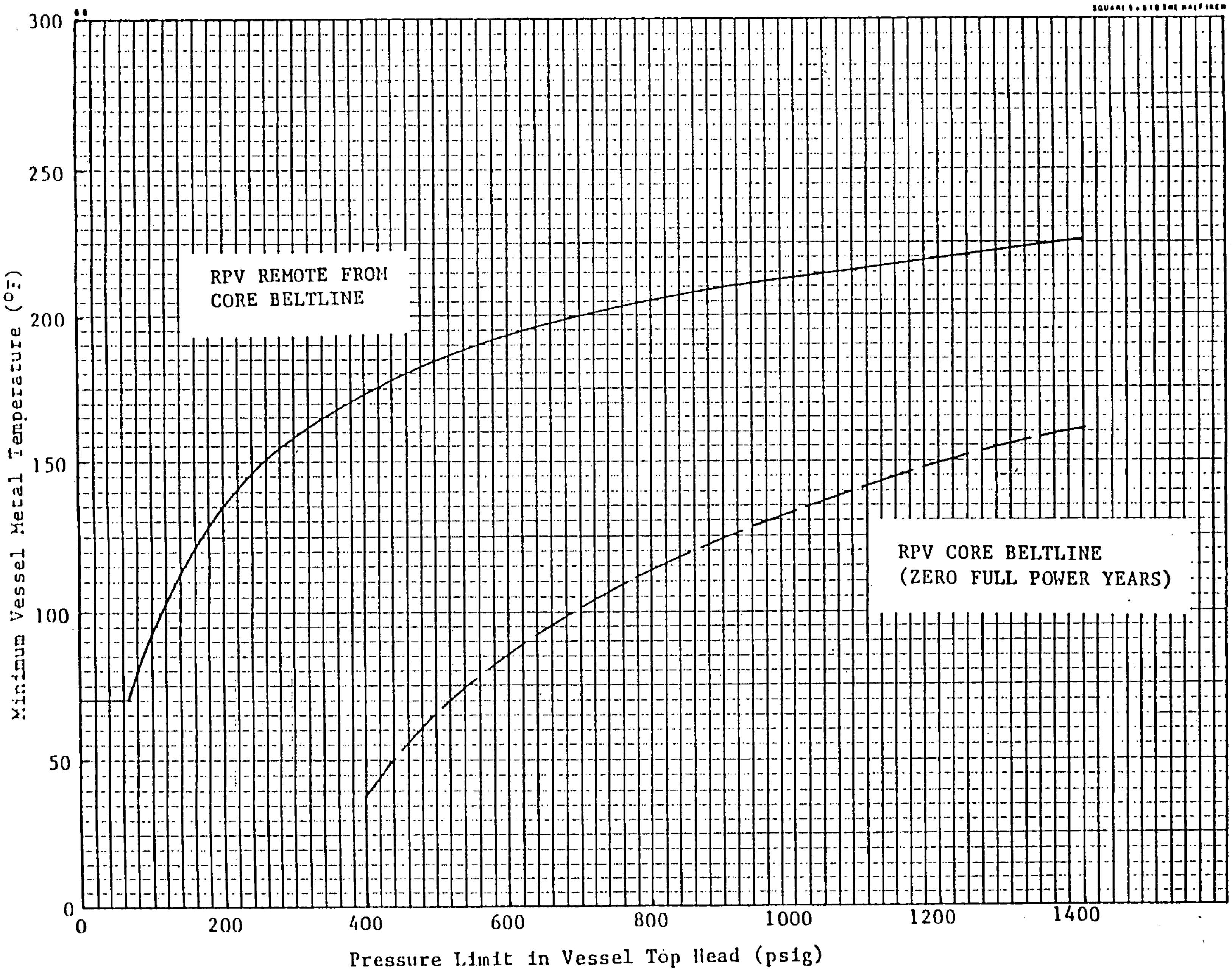
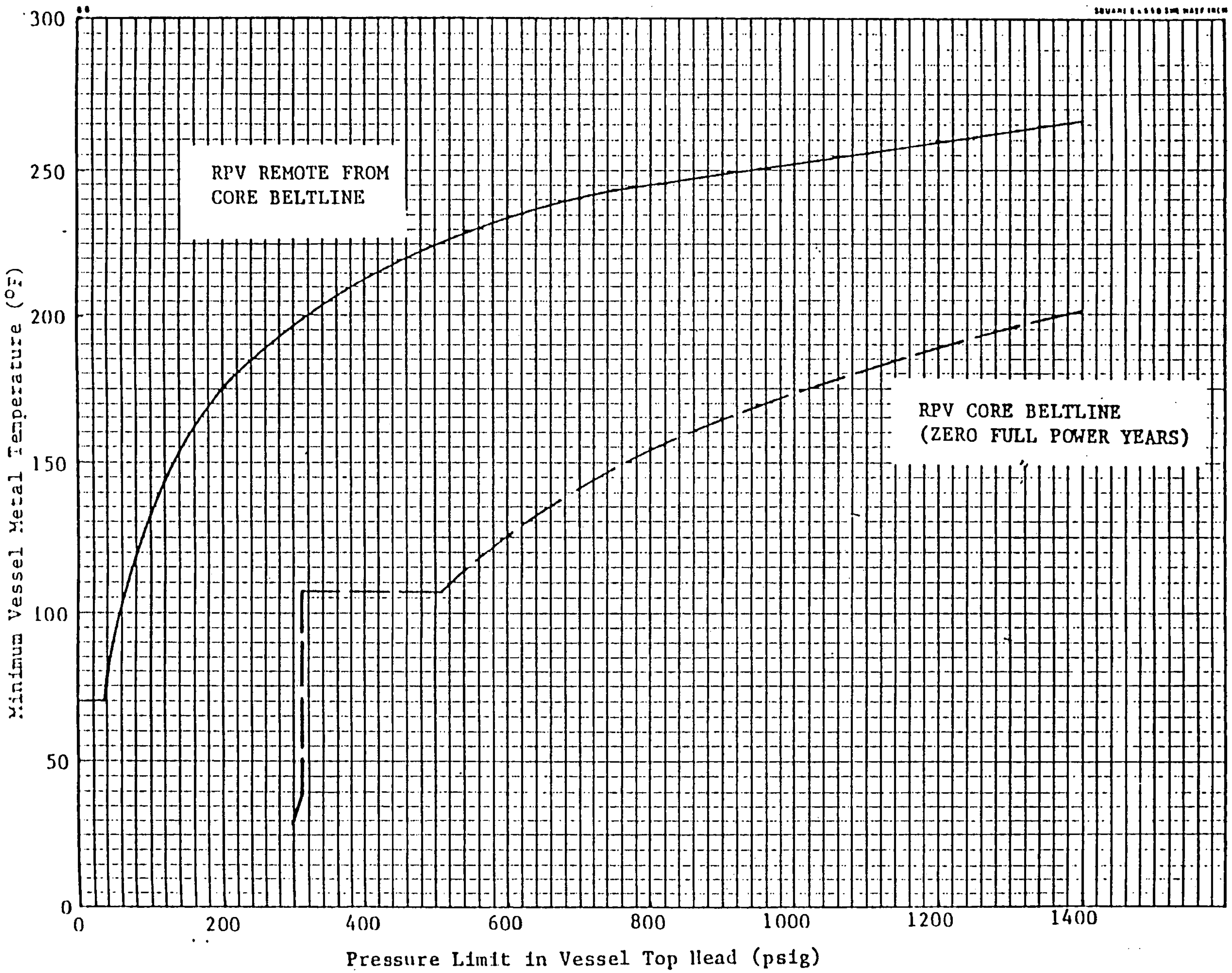


Figure 3.6.3 Minimum Temperature vs. Pressure for Mechanical Heatup or Cooldown Without the Core Critical



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4.0 SURVEILLANCE REQUIREMENTS

b. The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods, and provisions of 10 CFR Part 50:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at 40±10 month intervals* during shutdown at >Pa during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If any periodic Type A test fails to meet 0.75La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75La, at which time the above test schedule may be resumed.

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Exhibit D

License Amendment Request Dated March 31, 1989
Docket No. 50-263 License No. DPR-22

Exhibit D consists of General Electric Report SASR 88-99, Revision 1, January, 1989, "Implementation of Regulatory Guide 1.99, Revision 2, for the Monticello Nuclear Generating Plant." This report was prepared to support changes proposed in this License Amendment Request.