UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee) dated October 28, 1987, and modified by letter dated February 22, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 26, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

132
133
146
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3.6 Primary System Boundary

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. During operation where the core is critical or during heatup by non-nuclear means or cooldown following shutdown, the reactor vessel metal and fluid temperatures shall be at or above the temperatures shown on the limiting curves of Figures 3.6.1.a or 3.6.1.b.

3. The reactor vessel metal temperatures for the bottom head region and beltline region shall be at or above the temperatures shown on the limiting curves of Figure 3.6.2 during inservice hydrostatic or leak testing. The Adjusted Reference Temperature (ART) for the beltline region must be determined from the appropriate beltline curve (8, 10, or 12 EFPY) depending on the current accumulated number of effective full power years (EFPY). The ART curve for the bottom head is valid to 12 EFPY.

4.6 Primary System Boundary

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 50°F.

a. Bottom head drain.

b. Recirculation loops A and B.

2. Reactor vessel temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

3. Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-73 to the degree possible.

3.6.A (cont'd.)

4.6.A (cont'd.)

The schedule for withdrawal of the remaining two capsules is based on ASTM E185-82 and is as follows:

Second Capsule: 15 EFPY
Third Capsule: 32 EFPY

4. The Reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 80°F.

4. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

5. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

6. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

Thermal and Pressurization Limitations

The requirements for the reactor vessel have been identified by evaluating the need for its integrity over the full spectrum of plant conditions and events.

This is accomplished through the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the reactor vessel. The limits expressed in the technical specification for the applicable operating states are taken from the actual Nuclear Safety Operational Requirements for the reactor vessel as given in Subsection IV-2.8 of the Updated Safety Analysis Report.

The components of the nuclear system pressure boundary are constructed so that its initial maximum nil-ductility transition temperature (RT NDT) is not greater than 40°F, as cited in Subsection IV-2.5 of the Updated Safety Analysis Report. The heatup-cooldown and hydrostatic test minimum pressurization temperatures were calculated to comply with the recommendations of Appendix G of Section III, ASME Boiler and Pressure Vessel Code, 1972 Summer Addendum.

The temperature versus pressure limits when critical which are presented in Figure 3.6.1.b assure compliance with Appendix G of 10CFR50.

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the O-rings of the head and vessel flange. The reactor vessel head flange and head are constructed so that their initial maximum NDTT is 20°F, as cited in Paragraph IV-2.5 of the Updated Safety Analysis Report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 80°F (20°F + 60°F). The total integrated neutron flux in the head flange region will be less than that at the core mid-plane level by a factor of 10^{-3} or 10^{-4}, therefore, the maximum calculated fluence in the head flange region will be far below 1×10^{17} nvt. With such a low total integrated neutron flux in the head flange region, there will be no detectable or significant NDTT shift, and the minimum stud tightening temperature remains at 80°F.

The reactor vessel is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for a pressure of 1250 psig. The pressure limit of 1035 psig represents the maximum expected operating pressure in the steam dome when the station is operating at design thermal power. Observation of this limit assures that the operator remains within the envelope of conditions considered by Chapter 14 of the Updated Safety Analysis Report.

Stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. The specific conditions analyzed included a maximum of 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F. The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80.

As described in the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The first surveillance capsule was removed at 6.8 EFPY of operation and base metal, weld metal and HAZ specimens were tested. In addition, flux wires were tested to experimentally determine the integrated neutron flux (fluence) at the surveillance capsule location. The test results are presented in General Electric Report MDE-103-0986. Measured shifts in RT_{NDT} of the base metal and weld metal were compared to predicted values per Regulatory Guide 1.99, Revision 1. The measured values were higher than predicted, so the 1.99 methods were modified to reflect the surveillance data. The test results for the flux wires were used with analytically determined lead factors to determine the peak end-of-life (EOL) fluence at the 1/4 T Vessel wall depth. The value corresponding to 40 years operation (32 EFPY) is 1.5×10^{18} n/cm^2.

The adjusted reference temperature (ART) of a beltline material is defined as the initial RT_{NDT} plus the RT_{NDT} shift due to irradiation. The curves of Figures 3.6.1.a and 3.6.1.b reflect a beltline ART of 110°F, making them valid for operation up to 12 EFPY. Figure 3.6.2, the pressure test curve, includes curves with ART values for 8, 10 and 12 EFPY to provide more flexibility in pressure testing. Figure 3.6.2 also has a separate curve for the bottom head region. The bottom head curve does not shift with increased operation. Therefore, the bottom head temperature can be monitored against lower temperature requirements than the beltline during pressure testing.

B. Coolant Chemistry

Materials in the primary system are primarily Type-304 stainless steel and Ziracloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams[1], where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

[1] W. L. Williams, Corrosion 13, 1957, p. 539t.

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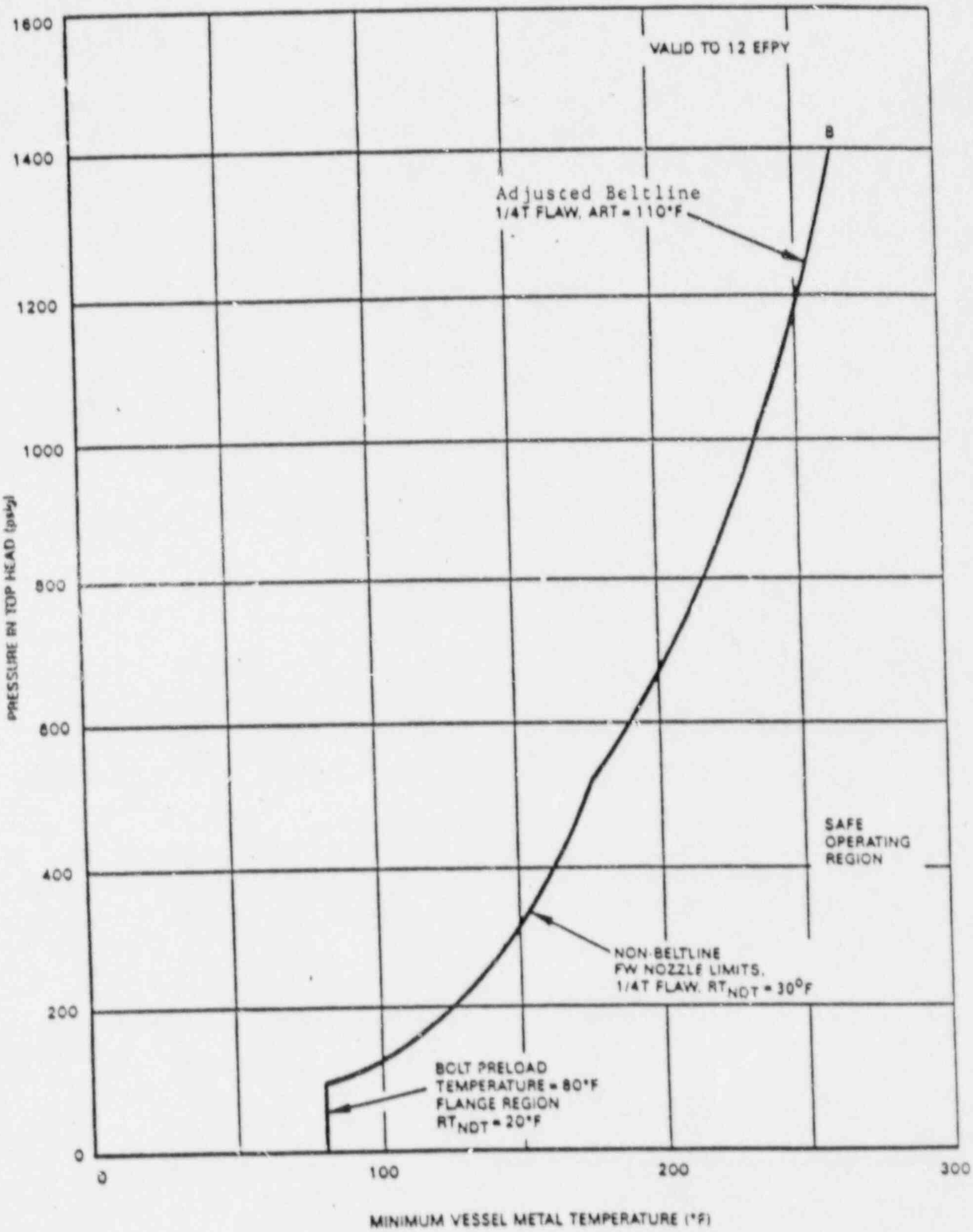

Figure 3.6.1.a Minimum Temperature for Non-Nuclear Heatup or Core Cooldown Following Nuclear Shutdown

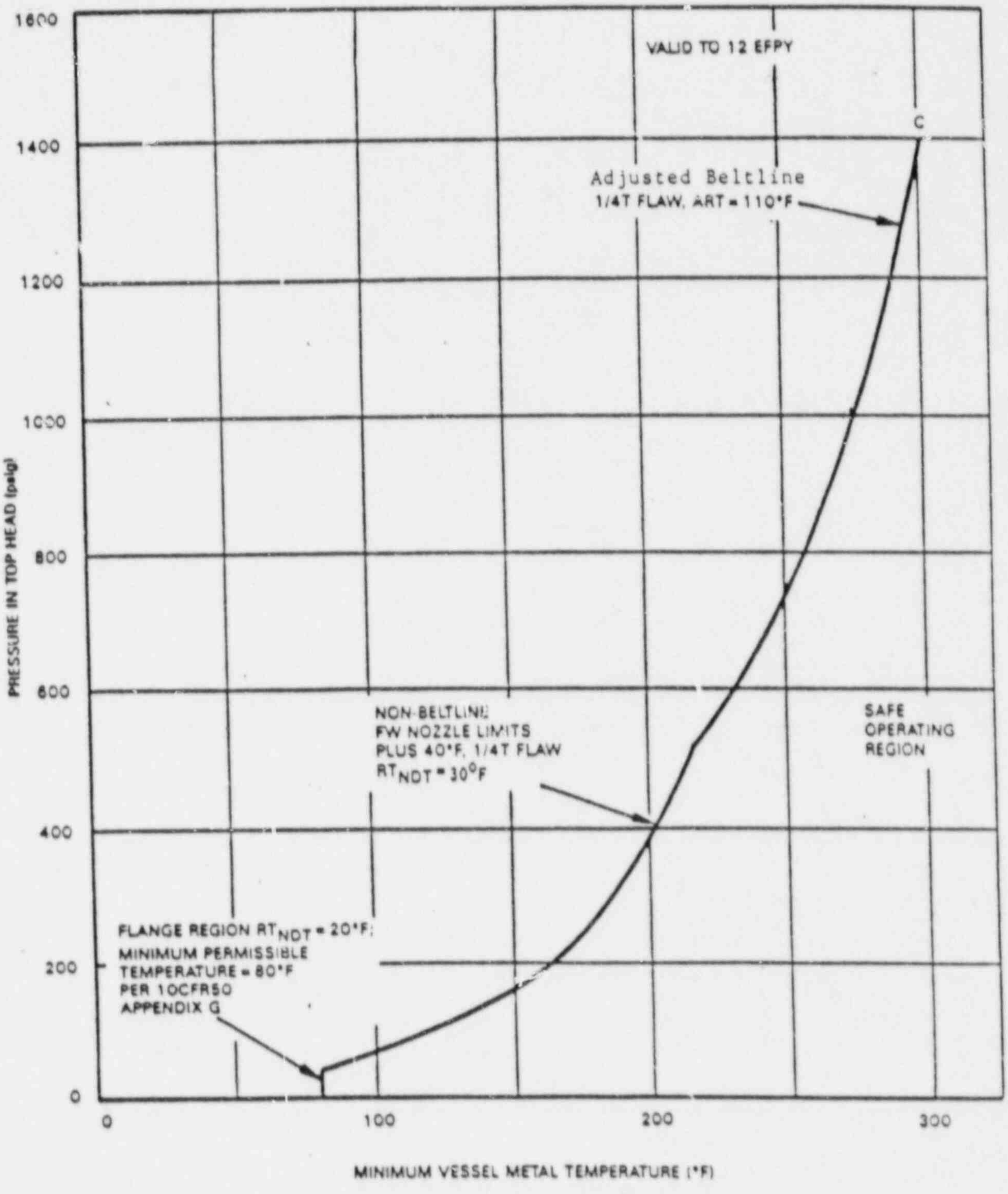

Figure 3.6.1.b Minimum Temperature for Core Operation (Criticality) - Includes 40°F Margin Required by 10CFR50 Appendix G

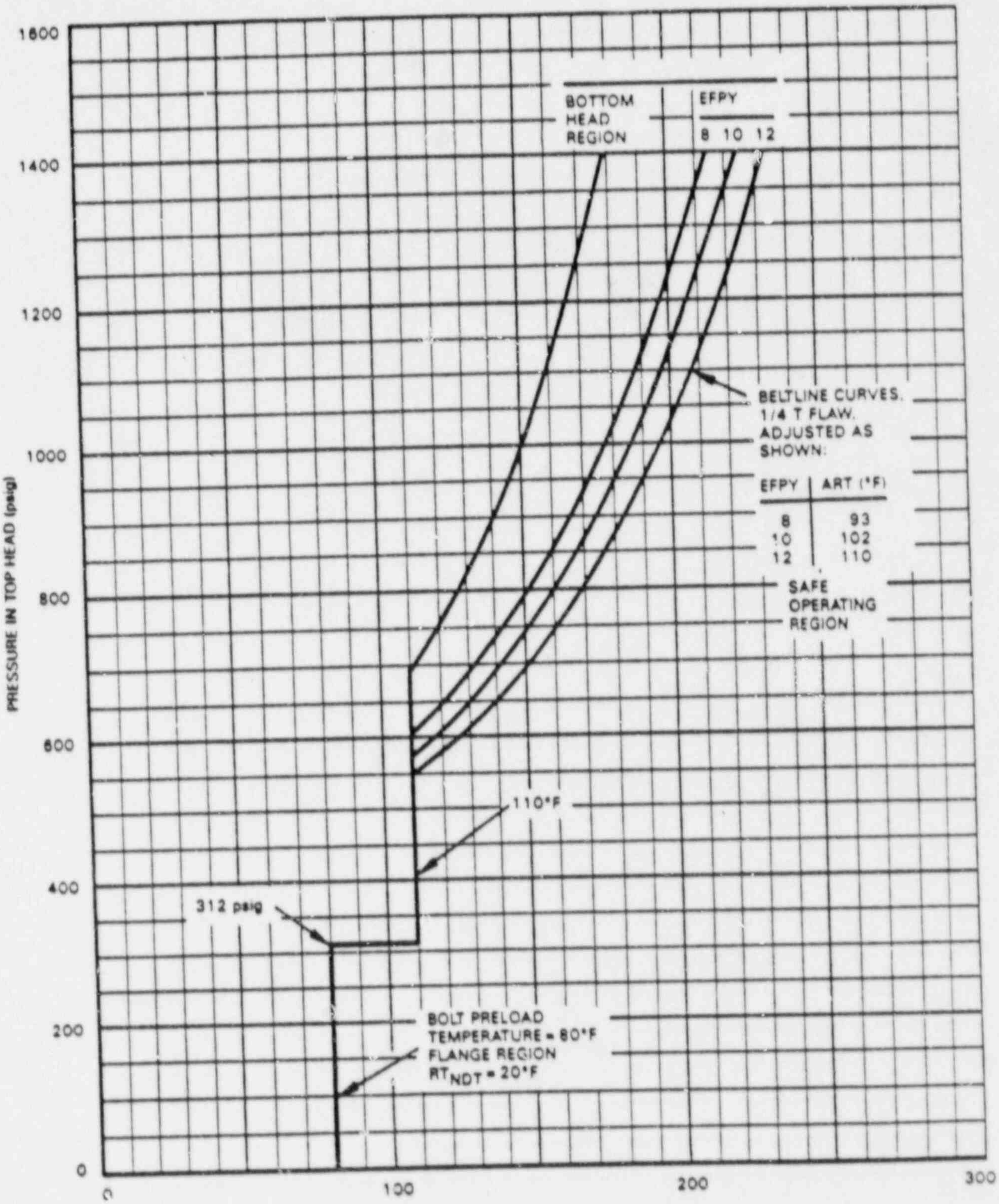

Figure 3.6.2 Minimum Temperature for Pressure Tests Such as Required by Section XI