3.1.2 Heatup and Cooldown Rates (Contd)

(2) (Contd)

surveillance program capsule which was removed at the beginning of the Cycle 3. For purposes of determining fluence at the reactor vessel beltline until a fluence of 1.3×10^{19}nvt is realized at the inner vessel wall at the beltline region, the following basis is established: 5.9×10^{19}nvt calculated at the reactor vessel beltline for 2540 MW_t for 40 years at an 80% load factor. This conversion has resulted in a correlation of 1.989×10^{12}nvt per 1 MWd_t.

(3) The limit lines in Figures 3-1, 3-2 and 3-3 are based on the requirements of Reference 9, Paragraphs IV.A.2 and IV.A.3. These lines reflect a preservice hydrostatic test pressure of 2400 psig and a vessel flange material reference temperature of 60°F.

Basis

All components in the primary coolant system are designed to withstand the effects of cyclic loads due to primary system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and start-up and shutdown operation. During unit start-up and shutdown, the rates of temperature and pressure changes are limited. A maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.(2)

The reactor vessel plate and material opposite the core has been purchased to a specified Charpy V-Notch test result of 30 ft-lb or greater at an NDTT of + 10°F or less. The vessel weld has the highest RT_{NDT} of plate, weld and HAZ materials at the fluence to which the Figures 3-1, 3-2 and 3-3 apply.(10) The unirradiated RT_{NDT} has been determined to be -56°F.(11) An RT_{NDT} of -56°F is used as an unirradiated value to which irradiation effects are added. In addition, the plate has been 100% volumetrically inspected by ultrasonic test using

Proposed

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other primary coolant system components, meets the appropriate design code requirements and specific component function and has a maximum NDTT of +40°F.[5]

As a result of fast neutron irradiation in the region of the core, there will be an increase in the RT with operation. The techniques used to predict the integrated fast neutron (E > 1 MeV) fluxes of the reactor vessel are described in Section 3.3.2.6 of the FSAR and also in Amendment 13, Section II, to the FSAR.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift from a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The maximum integrated fast neutron (E > 1 MeV) exposure of the reactor vessel is computed to be 5.9×10^{19} nvt for 40 years' operation at 2540 MW_t and 80% load factor. The predicted RT_{NDT} shift for the base metal has been predicted based upon surveillance data and the appropriate US NRC Regulatory Guide.[6] The actual shift in RT_{NDT} will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively be securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4-11 of the FSAR. To compensate for any increase in the RT caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

Reference 7 provides a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials in Class 1 components. This procedure is based on the principles of linear elastic fracture mechanics and involves a stress intensity factor prediction which is a lower bound of static, dynamic and crack arrest critical valves. The stress

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

intensity factor computed[7] is a function of RT_{NDT}, operating temperature, and vessel wall temperature gradients.

Pressure-temperature limit calculational procedures for the reactor coolant pressure boundary are defined in Reference 8 based upon Reference 7. The limit lines of Figures 3-1 through 3-3 consider a 54 psi pressure allowance to account for the fact that pressure is measured in the pressurizer rather than at the vessel beltline. In addition, for calculational purposes, 5°F and 30 psi were taken as measurement error allowances for temperature and pressure, respectively. By Reference 7, reactor vessel wall locations at 1/4 and 3/4 thickness are limiting. It is at these locations that the crack propagation associated with the hypothetical flaw must be arrested. At these locations, fluence attenuation and thermal gradients have been evaluated. During cooldown, the 1/4 thickness location is always more limiting in that the RT_{NDT} is higher than that at the 3/4 thickness location and thermal gradient stresses are tensile there. During heatup, either the 1/4 thickness or 3/4 thickness location may be limiting depending upon heatup rate.

Figures 3-1 through 3-3 define stress limitations only from a fracture mechanic's point of view.

Other considerations may be more restrictive with respect to pressure-temperature limits. For normal operation, other inherent plant characteristics may limit the heatup and cooldown rates which can be achieved. Pump parameters and pressurizer heating capacity tends to restrict both normal heatup and cooldown rates to less than 60°F per hour.

The revised pressure-temperature limits are applicable to reactor vessel inner wall fluences of up to 1.3×10^{19}nvt. The application of appropriate fluence attenuation factors (Reference 10) at the 1/4 and 3/4 thickness locations results in RT_{NDT} shifts of 223°F and 170°F, respectively, for the limiting weld material. The criticality condition

3.1.2 Heatup and Cooldown Rates (Contd)

Basis (Contd)

which defines a temperature below which the core cannot be made critical (strictly based upon fracture mechanics' considerations) is 352°F. The most limiting wall location is at 1/4 thickness. The minimum criticality temperature, 352°F is the minimum permissible temperature for the inservice system hydrostatic pressure test. That temperature is calculated based upon 2310 psig inservice hydrostatic test pressure.

The restriction of heatup and cooldown rates to 100°F/h and the maintenance of a pressure-temperature relationship to the right of the heatup, cooldown and inservice test curves of Figures 3-1, 3-2 and 3-3, respectively, ensures that the requirements of References 6, 7, 8 and 9 are met. The core operational limit applies only when the reactor is critical.

The criticality temperature is determined per Reference 8 and the core operational curves adhere to the requirements of Reference 9. The inservice test curves incorporate allowances for the thermal gradients associated with the heatup curve used to attain inservice test pressure. These curves differ from heatup curves only with respect to margin for primary membrane stress.(7) For heatup rates less than 60°F/h, the hypothetical 0°F/h (isothermal heatup) at the 1/4 T location is controlling and heatup curves converge. Cooldown curves cross for various cooldown rates, thus a composite curve is drawn. Due to the shifts in RT_{NDT}, NDTT requirements associated with nonreactor vessel materials are, for all practical purposes, no longer limiting.

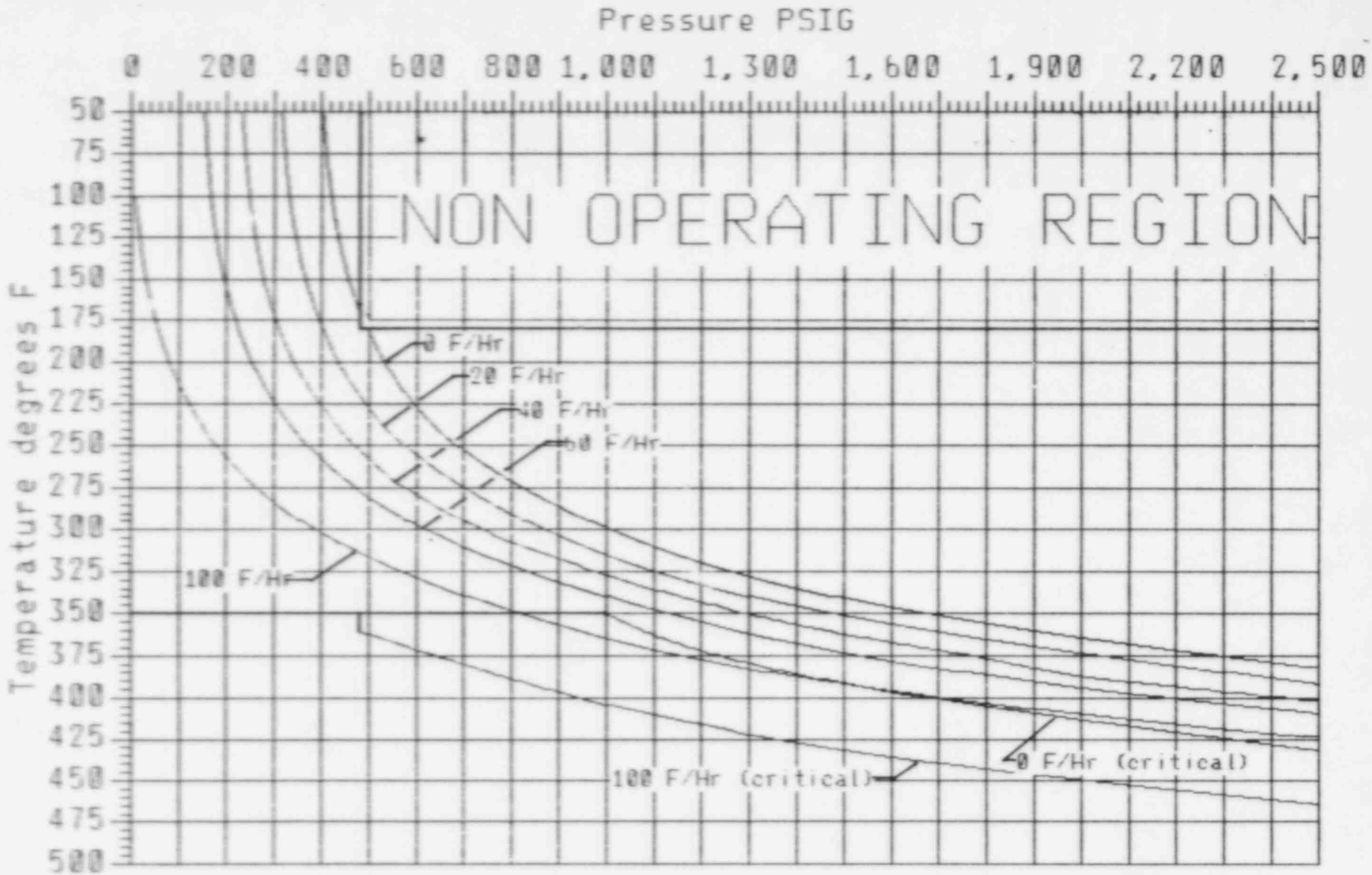
References

(1) FSAR, Section 4.2.2.

(2) ASME Boiler and Pressure Vessel Code, Section III, A-2000.

(3) Battelle Columbus Laboratories Report, "Palisades Pressure Vessel Irradiation Capsule Program: Unirradiated Mechanical Properties," August 25, 1977.

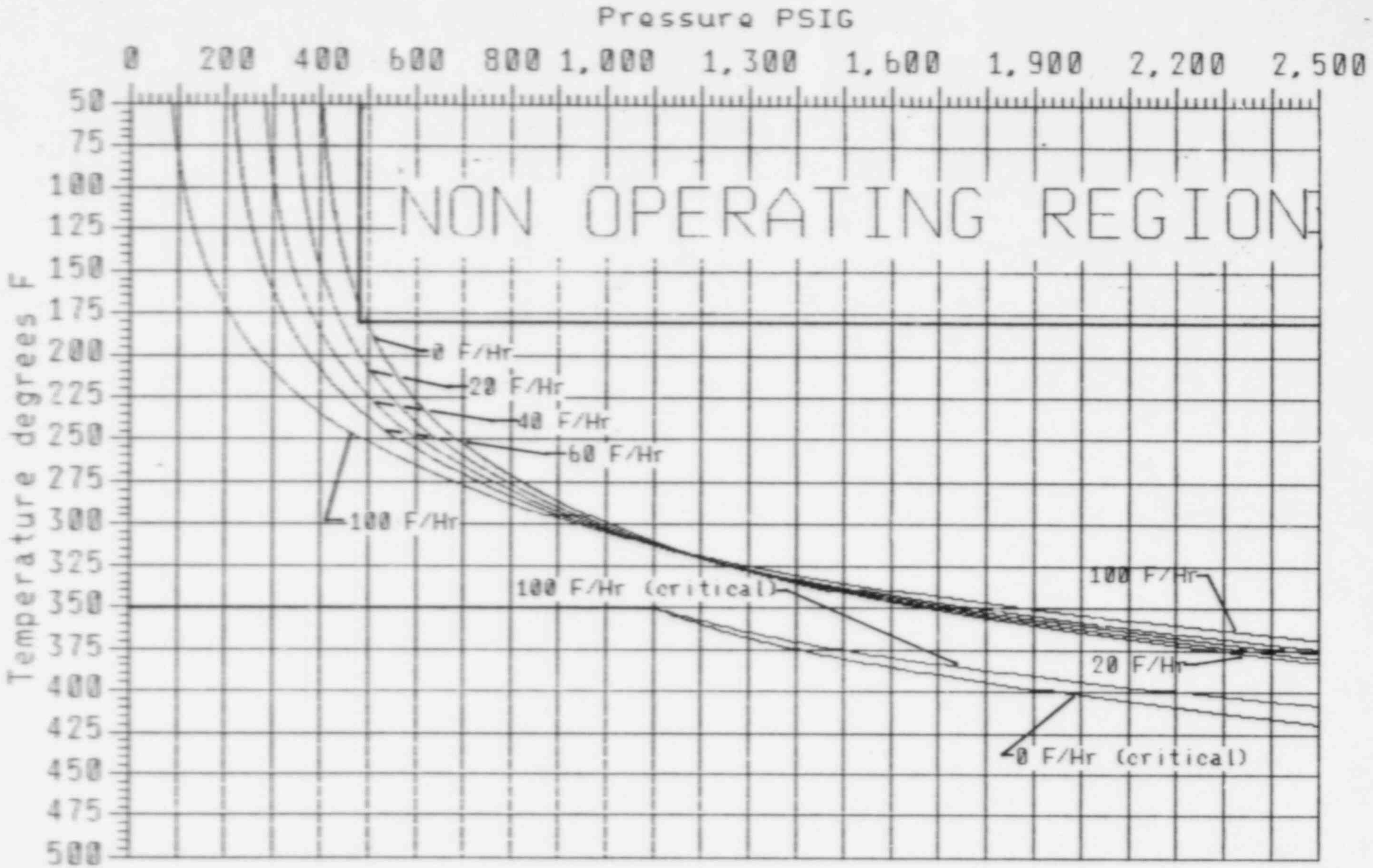
(4) Battelle Columbus Laboratories Report, "Palisades Nuclear Plant Reactor Vessel Surveillance Program: Capsule A-240," March 13, 1979, submitted to the NRC by Consumers Power Company letter dated July 2, 1979.



PALISADES PLANT   PRESSURE - TEMPERATURE LIMITS   DATE:
TECH. SPEC.   FOR HEATUP - TO 1.3 X 10^{19} nvt   AMEND. NO.

FIGURE 3-1

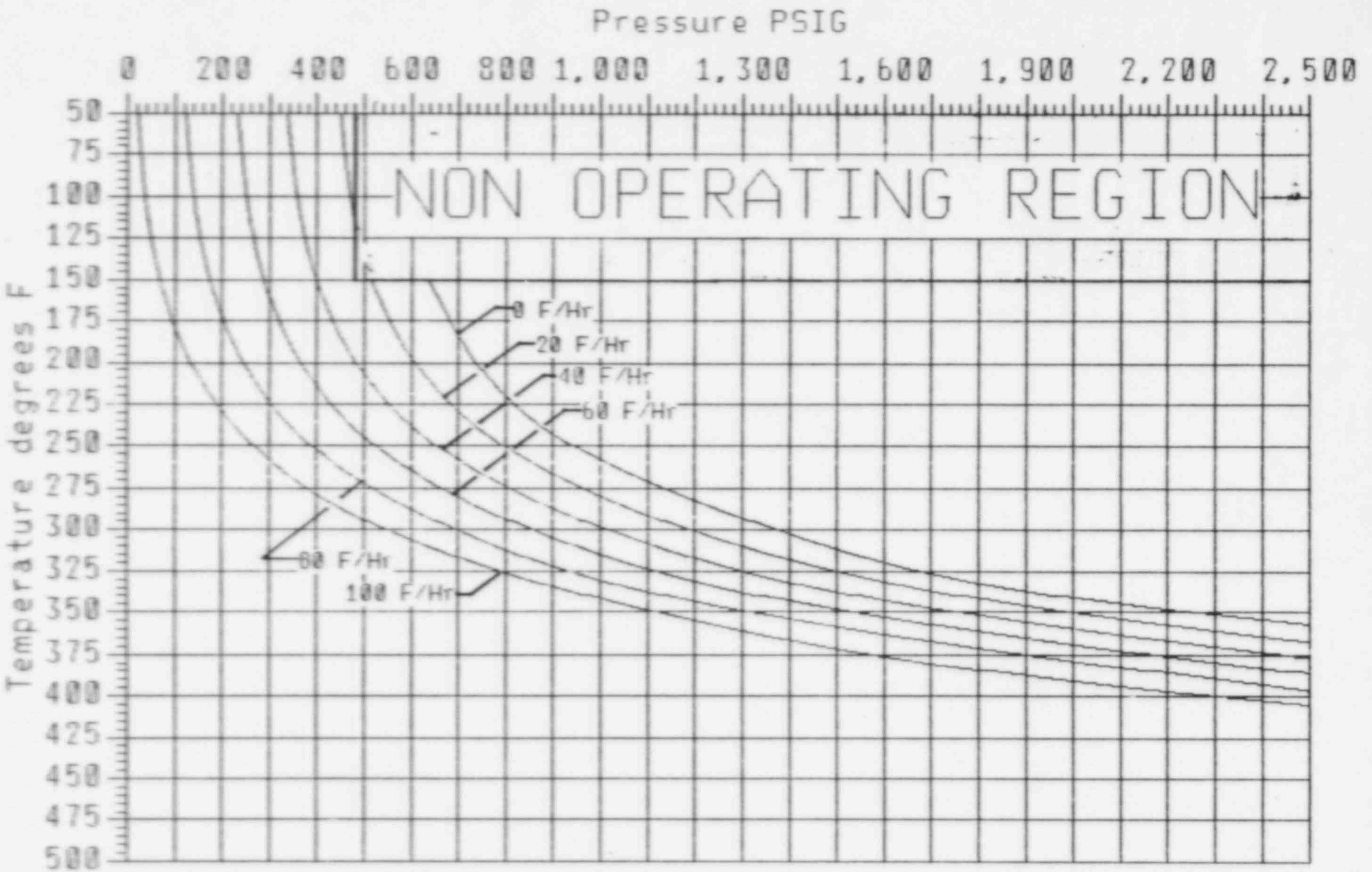
Proposed



PALISADES PLANT TECH. SPEC. — PRESSURE - TEMPERATURE LIMITS FOR COOLDOWN - TO 1.3 X 10^{19} nvt — DATE: AMEND. NO.

FIGURE 3-2

Proposed



PALISADES PLANT TECH. SPEC.

PRESSURE - TEMPERATURE LIMITS FOR HYDRO TEST - TO 1.3 X 10^{19} nvt

DATE:
AMEND. NO.

FIGURE 3-3

3.1.2 Heatup and Cooldown Rates (Contd)

References (Contd)

(5) FSAR, Section 4.2.4.

(6) US Nuclear Regulatory Commission, Regulator Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," July, 1975.

(7) ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure," 1974 Edition.

(8) US Atomic Energy Commission Standard Review Plan, Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits."

(9) 10 CFR Part 50, Appendix C, "Fracture Toughness Requirements," May 31, 1983.

(10) US Nuclear Regulatory Commission, Regulatory Guide 1.99 Draft Revision 2, April, 1984.

(11) Combustion Engineering Report CEN-189, December, 1981.

3.1.3 Minimum Conditions for Criticality

a) Except during low-power physics test, the reactor shall not be made critical if the primary coolant temperature is below 525°F.

b) In no case shall the reactor be made critical if the primary coolant temperature is below 352°F.

c) When the primary coolant temperature is below the minimum temperature specified in "a" above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

d) No more than one control rod at a time shall be exercised or withdrawn until after a steam bubble and normal water level are established in the pressurizer.

e) Primary coolant boron concentration shall not be reduced until after a steam bubble and normal water level are established in the pressurizer.

Basis

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly negative at operating temperatures with all control rods withdrawn.[1] However, the uncertainty of the calculation is such that it is possible that a slightly positive coefficient could exist.