Reactor Coolant System Heatup Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F

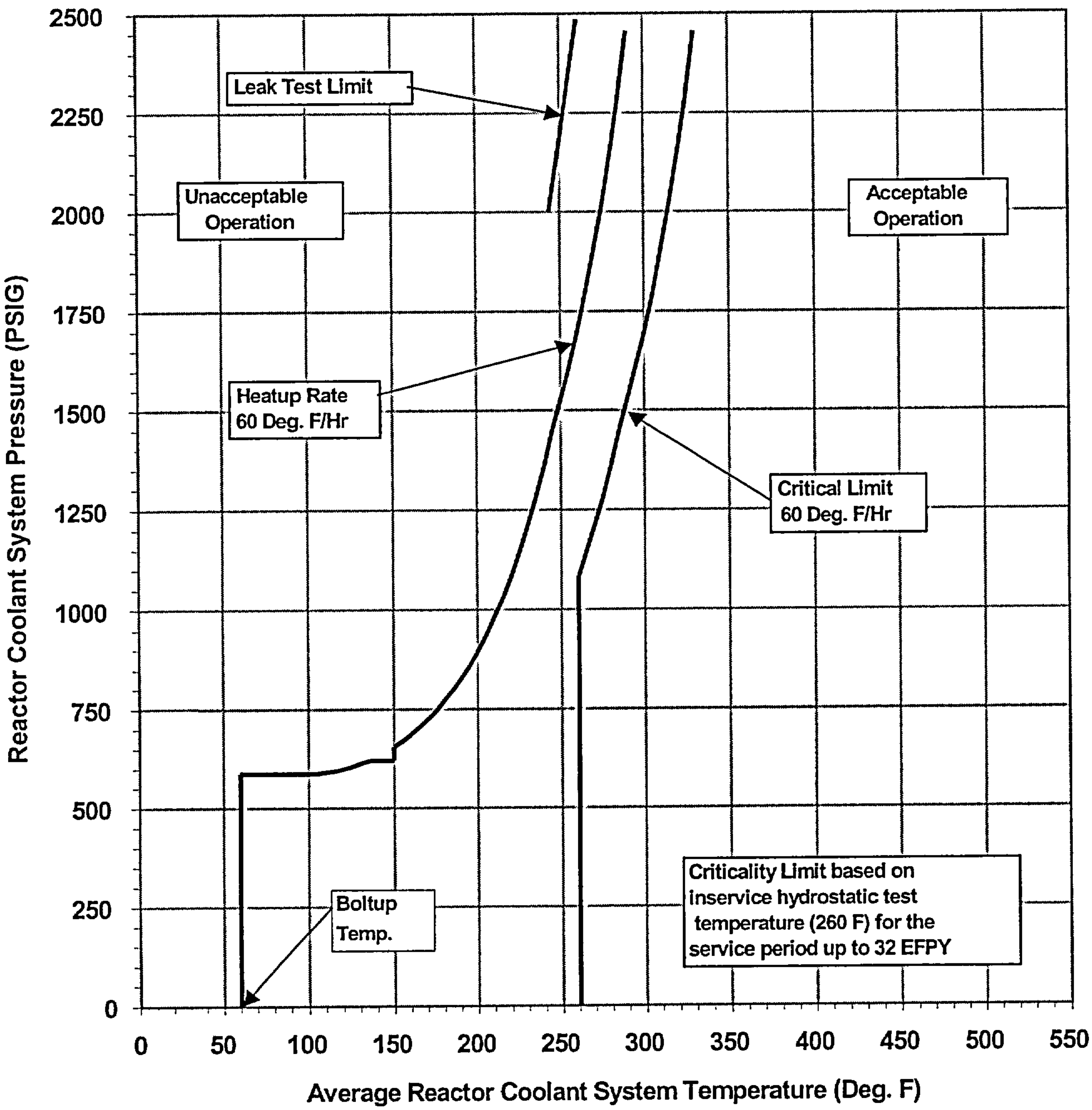

FIGURE 3.4-2
REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR 60 °F/HR RATE, CRITICALITY LIMIT, BOLTUP LIMIT, AND LEAK TEST LIMIT

Reactor Coolant System Cooldown Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F

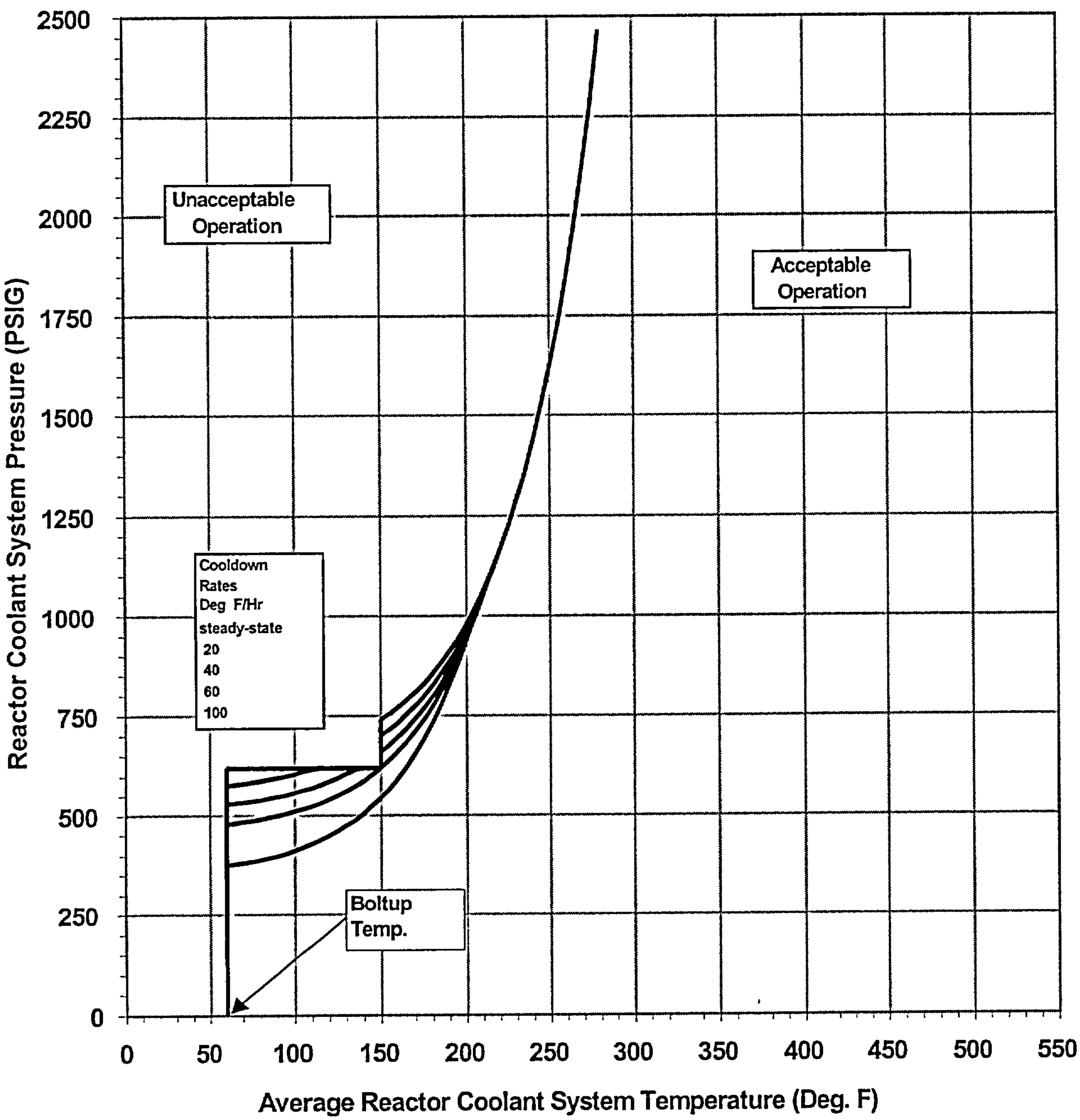

FIGURE 3.4-3
REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE, LIMITS FOR VARIOUS COOLDOWN RATES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of 32 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}: The results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron ($E > 1$ MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 32 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515, Revision 1, report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated May 2002.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.