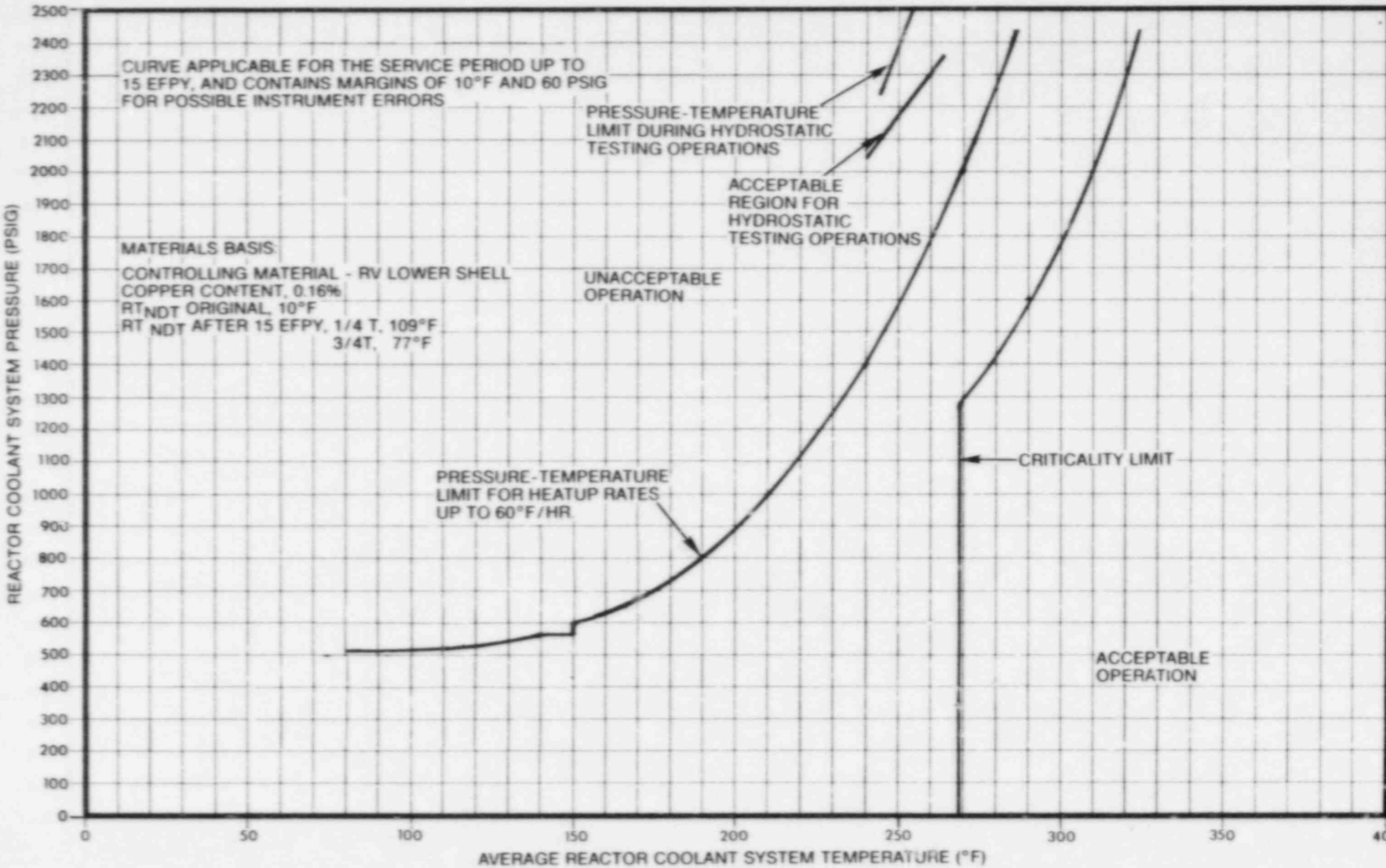

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS 60°F/HOUR HEATUP RATE—CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

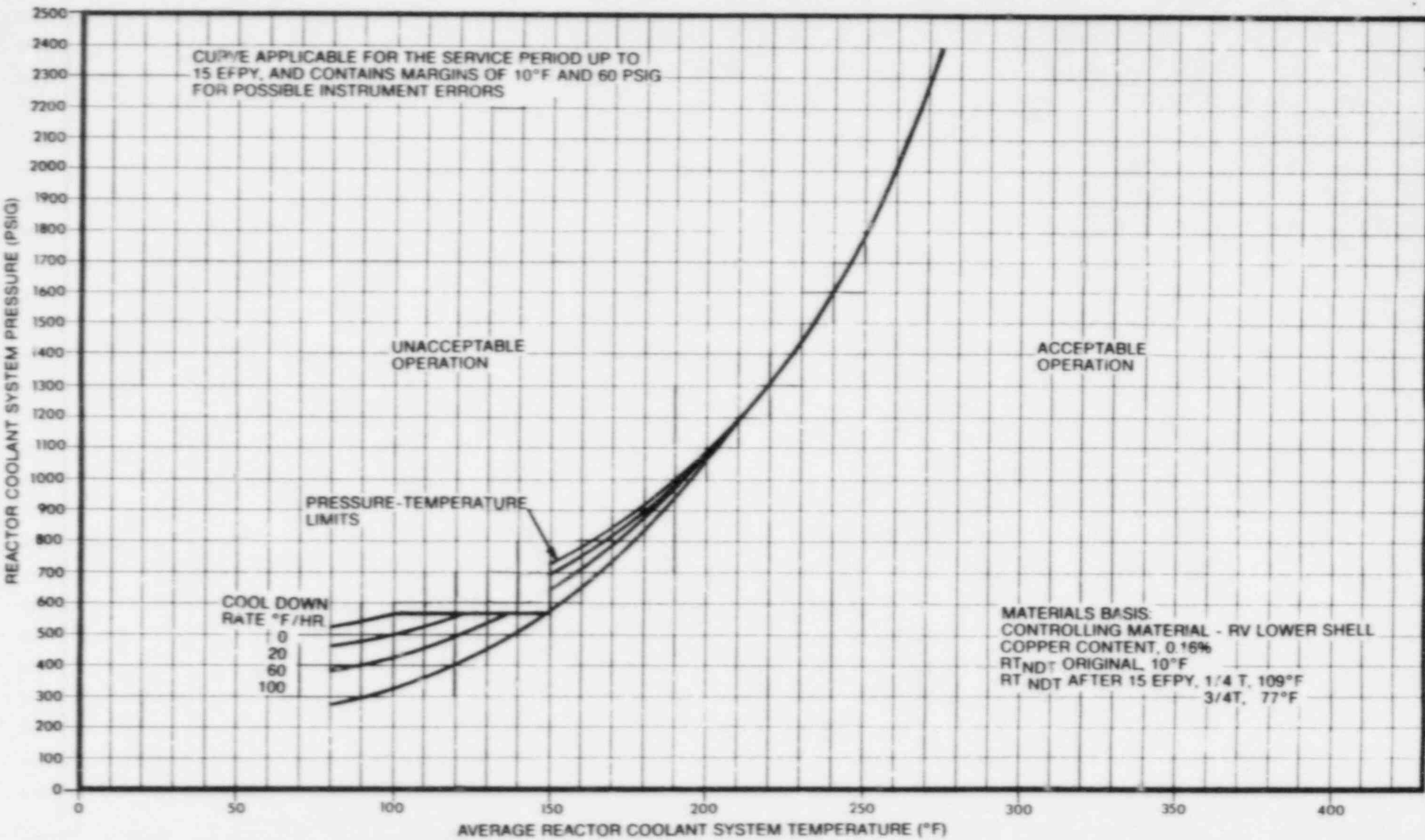

FIGURE 3.4-3
REACTOR COOLANT SYSTEM PRESSURE — TEMPERATURE LIMITS VERSUS COOLDOWN RATES

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1. U	Replacement of first core region
2. X	4.2 EFPY[a]
3. V	8.4 EFPY[b]
4. Y	16.8 EFPY[c]
5. W	Standby
6. Z	Standby

[a] Approximately EOL 1/4T fluence.

[b] Approximates EOL surface fluence (peak location).

[c] Approximates twice the EOL surface fluence (peak location).

REACTOR COOLANT SYSTEM

BASES

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (ie, no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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 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 upon the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY.

The knee in the heatup and cooldown rate pressure temperature limit curves is based on the May 31, 1983 revision to Section IV.A.2 of 10 CFR 50, Appendix G. This revision requires that when the core is not critical, and pressure exceeds 20 percent of the preservice system hydrostatic test pressure (620 psi), that the temperature of the closure flange regions must exceed the reference temperature of the material in those regions by at least 120°F for normal operation. The reference temperature of the head flange is limiting in this case (20°F) and is taken from Technical Specification Table B 3/4.4-1.

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, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 15 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

REACTOR COOLANT SYSTEM

BASES

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E 185-82, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for in-service 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 an RCS vent opening of greater than 3.40 square inches 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 $\leq$ 290°F. Either the PORV 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 $\leq$ 50°F above the RCS cold leg temperature or (2) the start of one safety injection pump, two centrifugal charging pumps, and one positive displacement pump and their simultaneous injection into a water solid RCS. The RCS vent opening is based on the flow area of the open PORV.

An initial pressurizer steam volume of $\geq$ 200 ft^3 will provide sufficient surge volume to preclude the RCS from becoming water-solid during the transient where a reactor coolant pump is started in an idle RCS with one or more steam generator secondary side temperatures > 50°F above the RCS cold leg temperatures.

The current heatup and cooldown curves are applicable for the first 15 EPY. The value of 290°F used for the overpressure mitigation system is similarly dependent on the irradiation of the reactor vessel and is applicable only for the first 15 EFPY.