ATTACHMENT NO. 1 TO AEP:NRC:0894E

TECHNICAL SPECIFICATION CHANGE DESCRIPTIONS

Change No. 1

Unit No. 1;	Page 3/4 4-27; Figure 3.4-2
	Page 3/4 4-28; Figure 3.4-3
	Page B 3/4 4-6; Section 3/4 4.9
	Page B 3/4 4-10; Table B 3/4.4-1
	Page B 3/4 4-11; Section 3/4 4.9

Figure 3.4-2 and Figure 3.4-3 are the heatup and cooldown curves for Unit No. 1. These curves are composite curves prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 12 EFPY. The revisions to the curves reflect the analyses done as a result of the capsule (Y) testing by Southwest Research Institute and the material data for the reactor vessel beltline region weld chemistry noted in our letter AEP:NRC:0894C dated July 3, 1985. The assumptions and limitations used in the development of the curves are noted in the revised bases. These revised curves supersede the curves submitted for Unit No. 1 in our earlier submittal No. AEP:NRC:0894 dated February 14, 1985.

The bases for Pressure/Temperature limits were re-written to reflect the current analyses and to read similarly to the Unit No. 2 bases for consistency.

The proposed change constitutes an additional limitation, restriction or control not presently included in the Technical Specifications and complies with changes in the federal regulations. We believe that the results of the change are clearly within all acceptable criteria since they are the direct result of an evaluation using methods previously accepted by the NRC. Therefore we believe the proposed changes do not involve a significant hazards consideration as defined by 10 CFR 50.92.

ATTACHMENT 2

TO

AEP:NRC:0894E

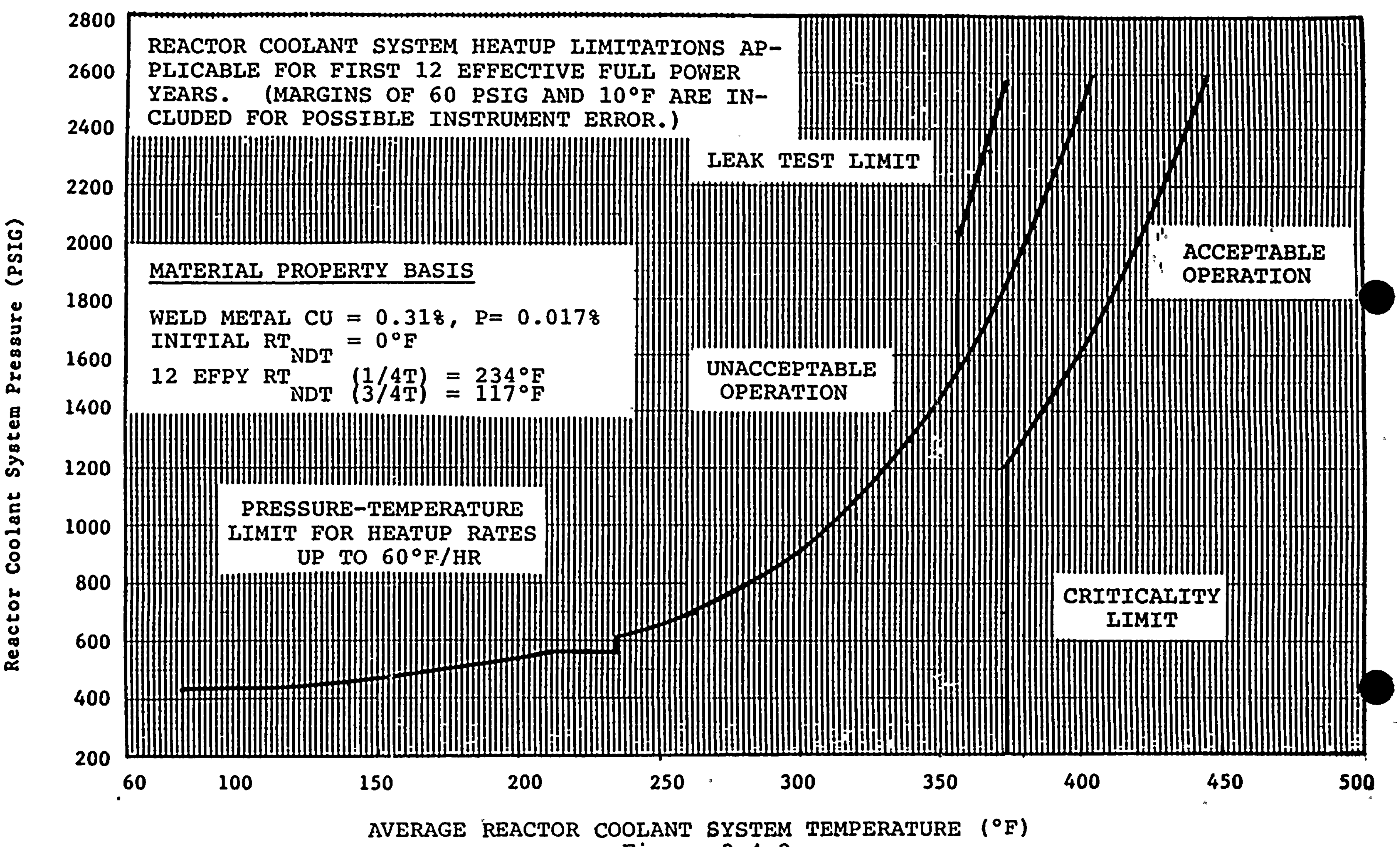

Figure 3.4-2

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

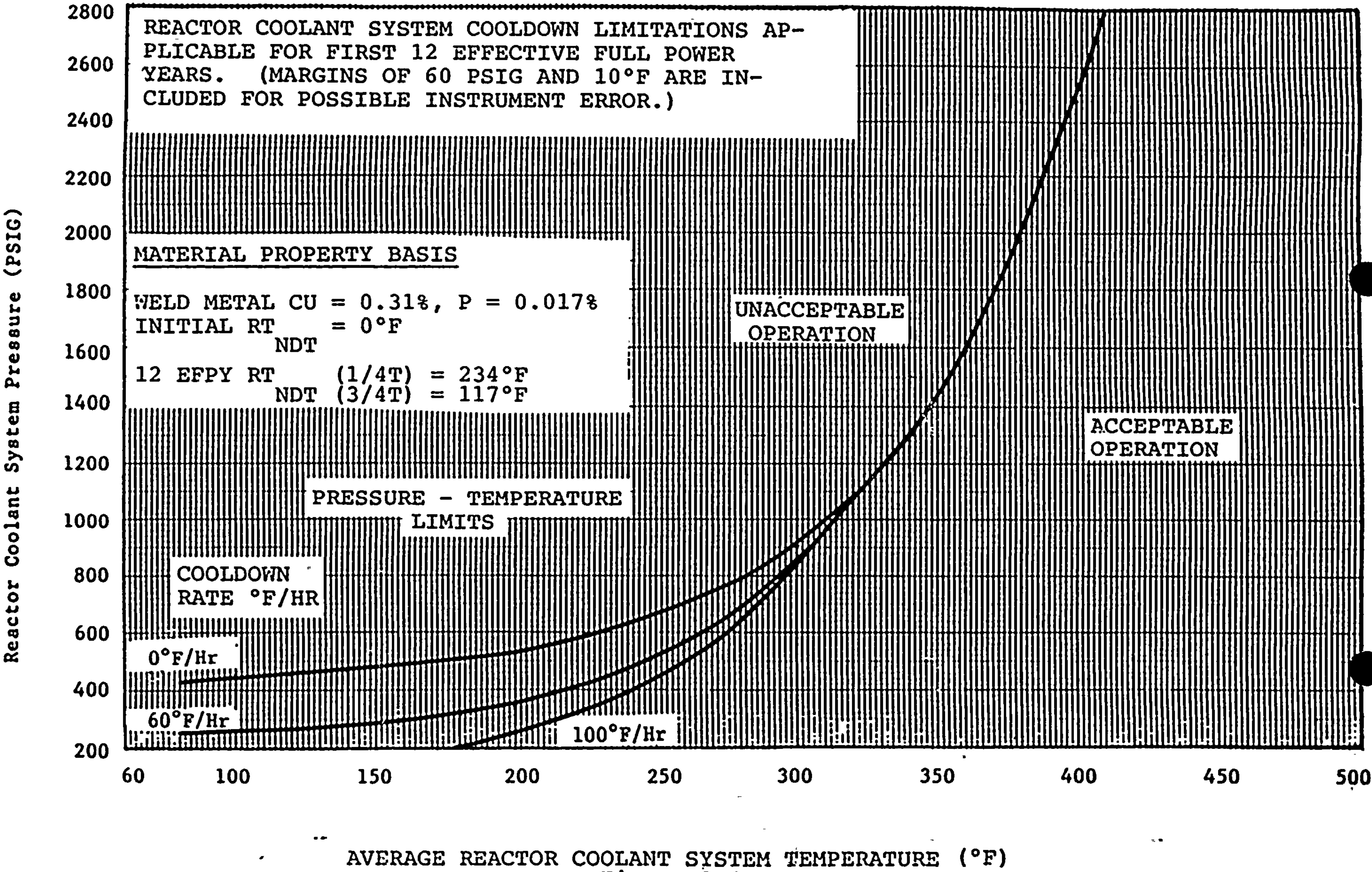

Figure 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS 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 upon the most limiting value of the predicted adjusted reference temperature at the end of 12 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, 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 of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG	TRANS	RTNDT F	MIN. UPPER SHELF FT-LB LONG	TRANS
WELD +	14GV	WELD	0.27	0.023	-70	NA	8*	-52	NA	114**
HAZ CORE	15GV	HAZ			-60	-40	NA	NA	118	NA

$ ESTIMATED (60 F OR 100 FT-LB TEMP, WHICHEVER IS LESS)
* ESTIMATED (77 FT-LB/54 MIL TEMP FOR LONGITUDINAL DATA)
** ESTIMATED (65 PERCENT OF LONGITUDINAL SHELF)
SURV PROBABLE MATERIAL FOR SURVEILLANCE PROGRAM ACCORDING TO E185

\+ FOR PURPOSES OF CALCULATING SUBSEQUENT HEAT-UP AND COOLDOWN CURVES WELD MATERIAL CHEMISTRY OF 0.31% CU AND 0.017% P SHALL BE USED.

BASES

The shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, 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 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 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, one PORV and the RHR safety valve, or an RCS vent opening of greater than or equal to 2 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 less than or equal to 188°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 or (2) the start of a charging pump and its injection into a water solid RCS.