REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3, and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period,

b. For the temperature ranges specified below, the cooldown rates should be as specified (in any one hour period):

i.	$T > 270^{\circ}F$	$\leq 100^{\circ}F/Hr$,
ii.	$270^{\circ}F \geq T > 170^{\circ}F$	$\leq 50^{\circ}F/Hr$,
iii.	$170^{\circ}F \geq T$	$\leq 10^{\circ}F/Hr$,

and

c. A maximum temperature change of less than or equal to 5°F in any one hour period during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.

ACTION:

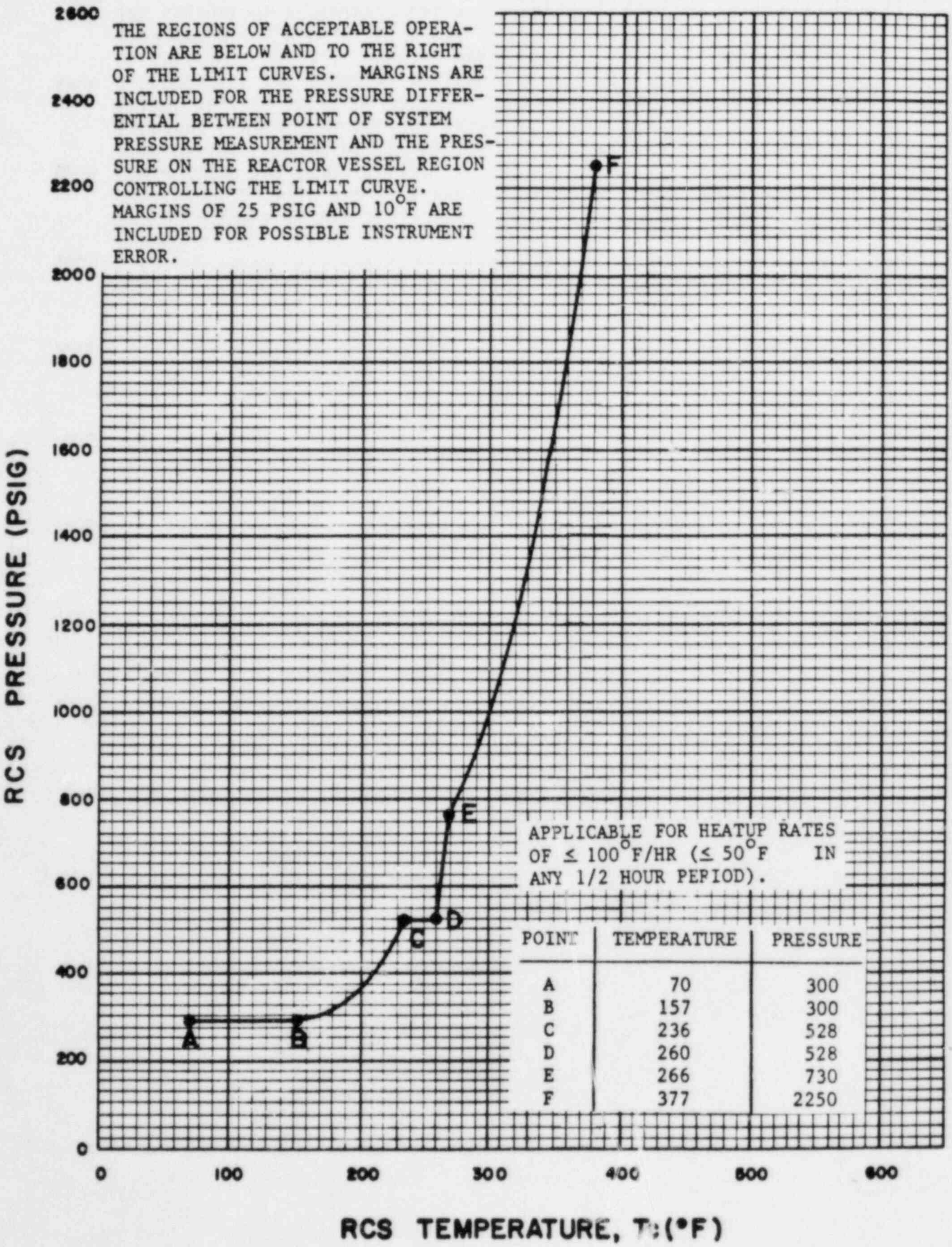
With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

FIGURE 3.4-2

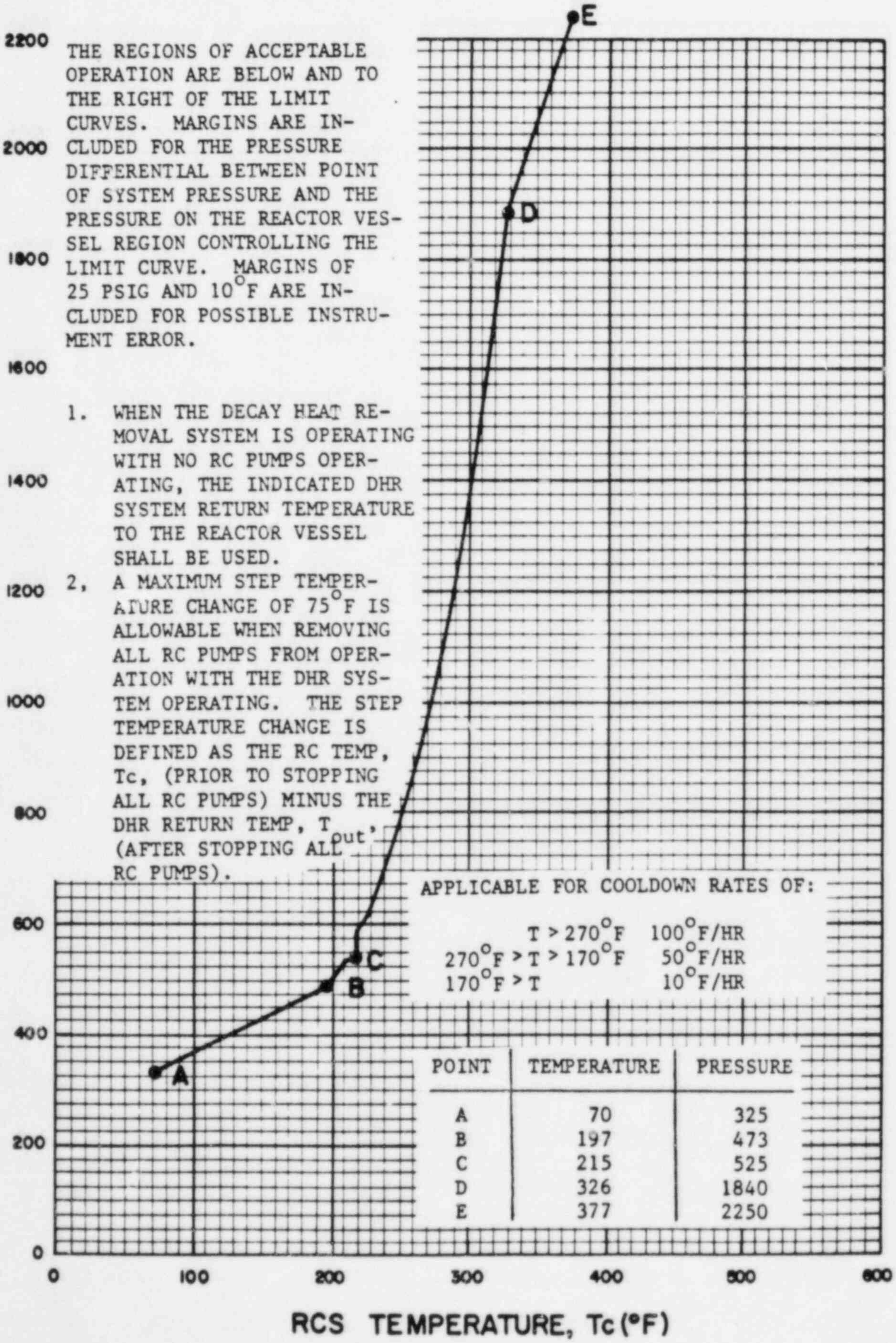
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR HEATUP FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	300
B	157	300
C	236	528
D	260	528
E	266	730
F	377	2250

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR COOLDOWN FIRST 8 EFPY



THE REGIONS OF ACCEPTABLE OPERATION ARE BELOW AND TO THE RIGHT OF THE LIMIT CURVES. MARGINS ARE INCLUDED FOR THE PRESSURE DIFFERENTIAL BETWEEN POINT OF SYSTEM PRESSURE AND THE PRESSURE ON THE REACTOR VESSEL REGION CONTROLLING THE LIMIT CURVE. MARGINS OF 25 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERROR.

1. WHEN THE DECAY HEAT REMOVAL SYSTEM IS OPERATING WITH NO RC PUMPS OPERATING, THE INDICATED DHR SYSTEM RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.
2, A MAXIMUM STEP TEMPERATURE CHANGE OF 75°F IS ALLOWABLE WHEN REMOVING ALL RC PUMPS FROM OPERATION WITH THE DHR SYSTEM OPERATING. THE STEP TEMPERATURE CHANGE IS DEFINED AS THE RC TEMP, Tc, (PRIOR TO STOPPING ALL RC PUMPS) MINUS THE DHR RETURN TEMP, T_{out}, (AFTER STOPPING ALL RC PUMPS).

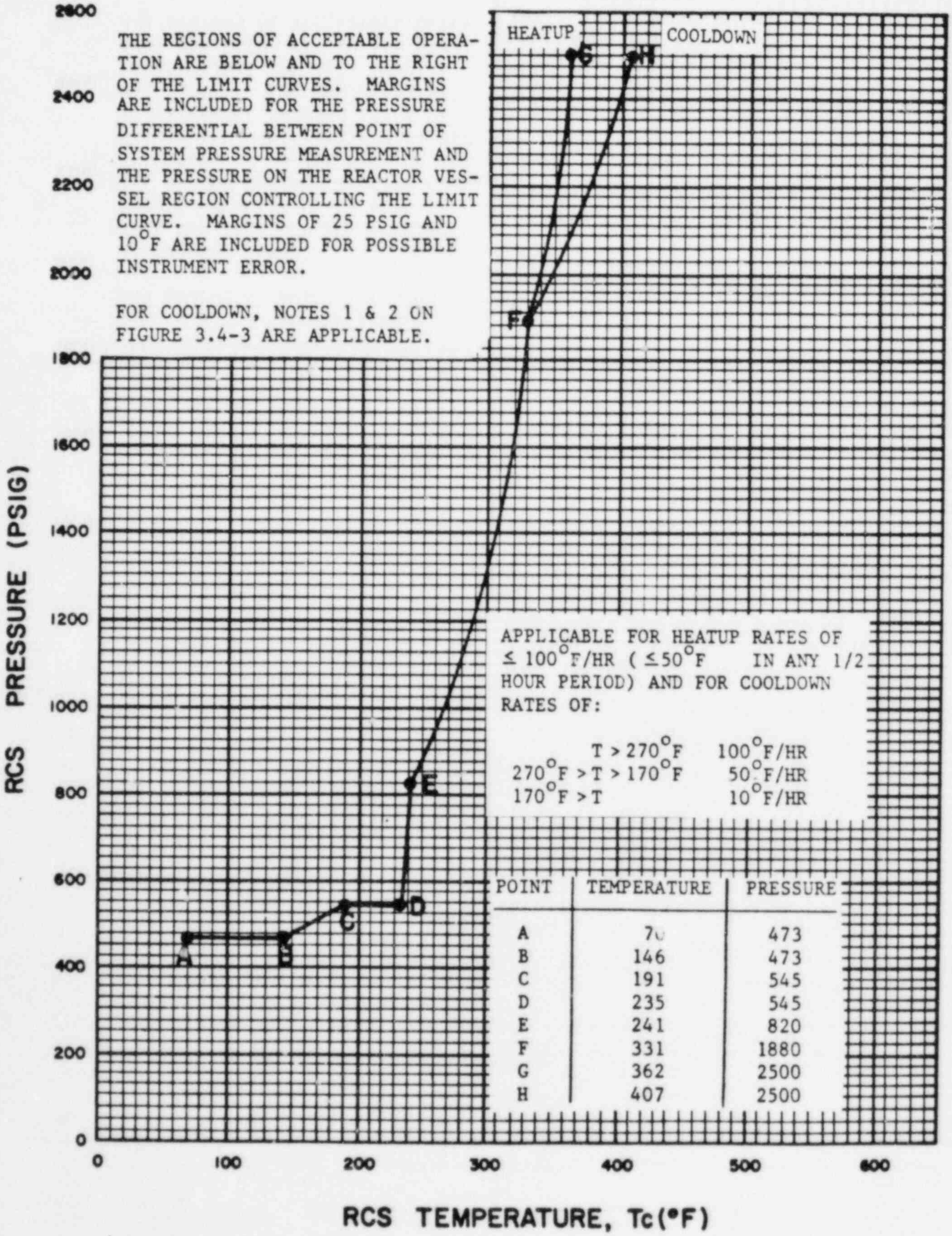
APPLICABLE FOR COOLDOWN RATES OF:

$T > 270°F$	100°F/HR
$270°F > T > 170°F$	50°F/HR
$170°F > T$	10°F/HR

POINT	TEMPERATURE	PRESSURE
A	70	325
B	197	473
C	215	525
D	326	1840
E	377	2250

FIGURE 3.4-4

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR HEATUP & COOLDOWN LIMITS FOR INSERVICE LEAK AND HYDROSTATIC TESTS FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	473
B	146	473
C	191	545
D	235	545
E	241	820
F	331	1880
G	362	2500
H	407	2500

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BASES TABLE 4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU %	P %	S %	RT NDT F	TRANS UPPER SHELF FT-LB	RT ADJUSTED NDT FOR 8 FULL POWER YEARS @ 1/4 T, °F	@ 3/4 T, °F
Nozzle Belt	SA-508 CL 2	.054	.008	.006	+10	183	26	17
*Upper Shell	SA-533B	.20	.008	.016	+20	88	90	54
**Upper Shell	SA-533B	.20	.008	.016	+20	90	90	54
Lower Shell	SA-533B	.12	.013	.015	-20	119	26	2
Lower Shell	SA-533B	.12	.013	.015	+45	88	91	67
***Surveillance	Weld	.30	.020	.005	+43	63		
Upper Long	Weld	.20	.009	.009	(+20)****	66****	130	73
Upper Long	Weld	.105	.091	.004	(+20)****	66****	130	73
Upper Circum (60%)	Weld	.106	.014	.013	(+20)****	66****	NA	53
Upper Circum (40%)	Weld	.19	.021	.016	(+20)****	66****	128	NA
Middle Circum (100%)	Weld	.27	.014	.011	(+20)****	66****	177	96
Lower Long (100%)	Weld	.22	.015	.013	(+20)****	66****	134	75
Lower Circum (100%)	Weld	.20	.015	.021	(+20)****	66****	30	20
Out 1st Nozzle	Weld	.19	.021	.016	(+20)****	66****		
Middle Circum	Atypical weld	--	---	---	+90		136	112

* Surveillance Base Metal A
** Surveillance Base Metal B
*** Surveillance Weld
**** Estimated Value

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 100°F per hour. During cooldown, similar types of thermal stress occur. Thus, the cooldown limit curve, Figure 3.4-3, is also a composite curve which was prepared based upon the same type analysis as the heatup curve 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. Additionally, during cooldown and heatup at the higher temperatures, the most conservative limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-3 and 3.4-4, respectively. (These limits will not require adjustments due to the neutron fluences.)

During the first several years of service life, the most limiting Reactor Coolant System regions are the closure head region (due to mechanical loads resulting from bolt pre-load) and the reactor vessel outlet nozzles. Nozzle sensitivity is caused by the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the beltline region of the reactor vessel becomes the most limiting region due to material irradiation.

For the service period for which the limit curves are established, the pressure/temperature limits were obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and the most sensitive material in the beltline region. The lowest pressure calculated for these three regions becomes the maximum allowable pressure for the fluid temperature used in the calculation. The calculated pressure/temperature curves are adjusted by 25 PSI and 10°F for possible instrument errors. The pressure limit is also adjusted for the pressure differential between the point of pressure measurement and the limiting component for all combinations of reactor coolant pump operations.

Irradiation damage to the beltline region can be quantified by determining the decrease in the temperature at which the metal changes from ductile to brittle fracture (ΔRT_{NDT}). The unirradiated transverse impact properties of the beltline region have been determined for those materials for which sufficient amounts of materials were available and are listed on Table 4-1. The adjusted reference temperatures on Table 4-1 are calculated by adding the predicted radiation-induced change in the reference temperature (ΔRT_{NDT}) and the unirradiated reference temperature. (The assumed unirradiated RT_{NDT} of the closure head region and of the outlet nozzle steel forgings was 60°F.) The adjusted RT_{NDT}s of the beltline region materials at the end of the eighth full power year are listed on Table 4-1 for the one-quarter and three-quarter wall thickness of the vessel wall.

Bases Figure 4-1 illustrates the calculated peak neutron fluence, for several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules, as a function of exposure time. Bases Figure 4-2 illustrates the design curves for predicting the radiation-induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. Thus, using these two figures and information on Table 4-1, shifts in the RT_{NDT} can be predicted over the full service life of the vessel.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating the 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 the 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 limit curves must be recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure. The pressure and temperature limits shown on Figures 3.4-2 and 3.4-4 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 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.

3/4 4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, 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.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

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