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 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of $60^{\circ}F$ in any one hour period.

b. A maximum cooldown of $100^{\circ}F$ in any one hour period.

c. A maximum temperature of less than or equal to $5^{\circ}F$ in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure within the limit 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 the RCS T_{avg} and pressure to less than $200^{\circ}F$ 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

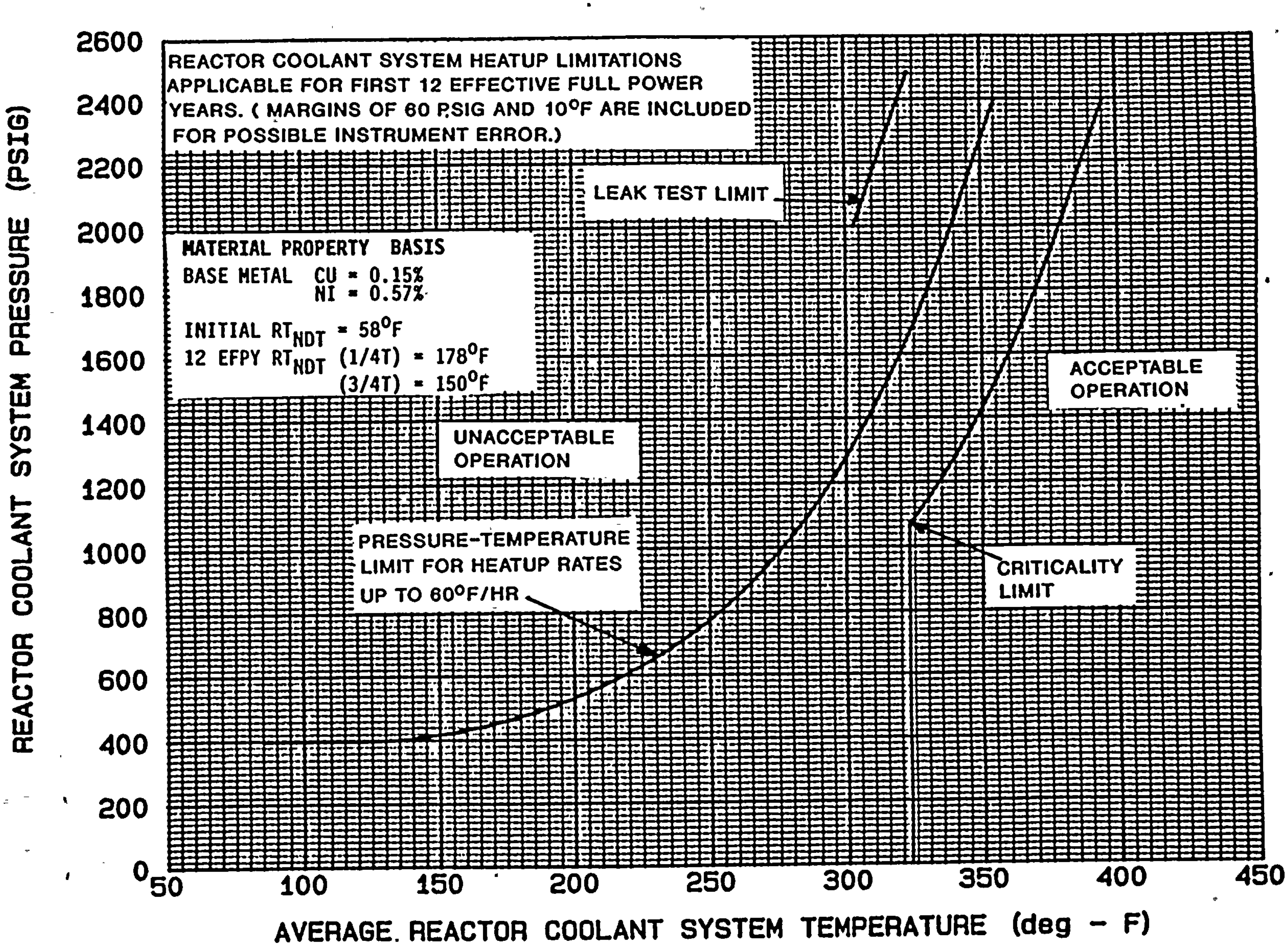

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS 60 deg-F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

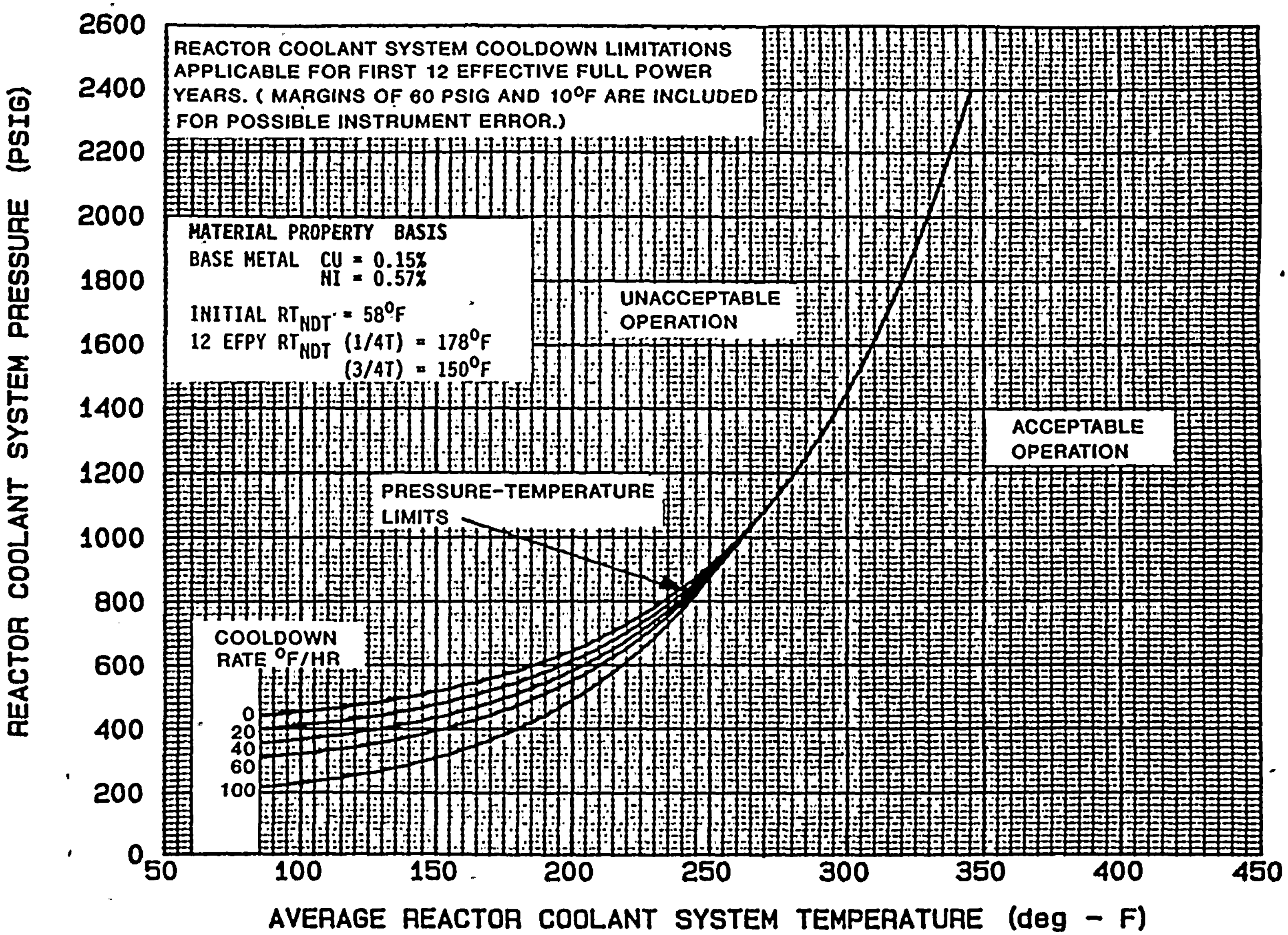

FIGURE 3.4-3

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 on 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 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.

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TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	CODE NO.	MATERIAL TYPE	CU (%)	NI (%)	T_{NDT} (°F)	50 FT-LB 35 MIL Temp(a) (°F)	RT_{NDT} (°F)	USE MWD (ft-lb)	USE NMWD(a) (ft-lb)
CL. HD. DOME	B0048-2	A533B CL.1	NA	0.64	-20	30	-20	148	96
CL. HD. SEG.	B9883-2	A533B CL.1	NA	0.66	-20	-3	-20	143.5	93
CL. HD. SEG.	A5189-2	A533B CL.1	NA	0.63	10	72	12	140.5	91
CL. HEAD FLG.	4437-V-1	A508 CL.2	NA	0.70	-20	5	-20	239	155
VESSEL FLANGE	4436-V-2	A508 CL.2	NA	0.70	30	15	30	161	105
INLET NOZZLE	269T-2	A508 CL.2	NA	0.85	-20	-15	-20	201.5	131
INLET NOZZLE	270T-1	A508 CL.2	NA	0.91	-20	-3	-20	239.5	156
INLET NOZZLE	269T-1	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
INLET NOZZLE	270T-2	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
OUTLET NOZZLE	271T-1	A508 CL.2	NA	0.80	0	12	0	>179	NA
OUTLET NOZZLE	271T-2	A508 CL.2	NA	0.80	0	-15	0	181	117.5
OUTLET NOZZLE	272T-1	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
OUTLET NOZZLE	272T-2	A508 CL.2	NA	NA	0	NA	0	NA	NA
UPPER SHELL	C5518-2	A533B CL.1	.12	0.61	10	88	28	107.5	70
UPPER SHELL	C5521-1	A533B CL.1	.14	0.59	0	93	33	112	73
UPPER SHELL	C5518-1	A533B CL.1	.12	0.57	10	66	10	> 82.5	NA
INTER SHELL	C5556-2	A533B CL.1	.15	0.57	0	118(b)	58(b)	109.5	90(b)
INTER SHELL	C5521-2	A533B CL.1	.14	0.58	10	98(b)	38(b)	111.5	86(b)
LOWER SHELL	C5540-2	A533B CL.1	.11	0.64	-20	35(b)	-20(b)	113	110(b)
LOWER SHELL	C5592-1	A533B CL.1	.14	0.59	-20	25(b)	-20(b)	107	103(b)
BOT. HD. SEG.	C5823-2	A533B CL.1	NA	0.57	-10	45	-10	129	84
BOT. HD. SEG.	A4957-3	A533B CL.1	NA	0.51	-10	20	-10	149	97
BOT. HD. SEG.	B0019-18	A533B CL.1	NA	0.61	-50	0	-50	177	115
INTER. & LOWER SHELL LONG. and GIRTH WELD SEAM	(HT S3986 & SAW Linde 124 Flux Lot No. 0934)		.06	0.97	-40	25	-35(b)	NA	97(b)

a) Estimated per NRC Standard Review Plan
b) Actual values
NA - Not available or not applicable, as appropriate

MWD - Major Working Direction
NMWD - Normal to MWD

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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 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 12 EFPY heatup and cooldown curves were developed based on the following:

1. The projected fluence values established by specimen analysis.

2. Intermediate shell plate C5556-2 being the limiting material as determined by Position 1 of Regulatory Guide 1.99, Revision 2, with a copper and nickel content of 0.15% and 0.57%, respectively.

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 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 or (2) the start of a charging pump and its injection into a water solid RCS.

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.