Attachment B

MARKED-UP TECHNICAL SPECIFICATIONS

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REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

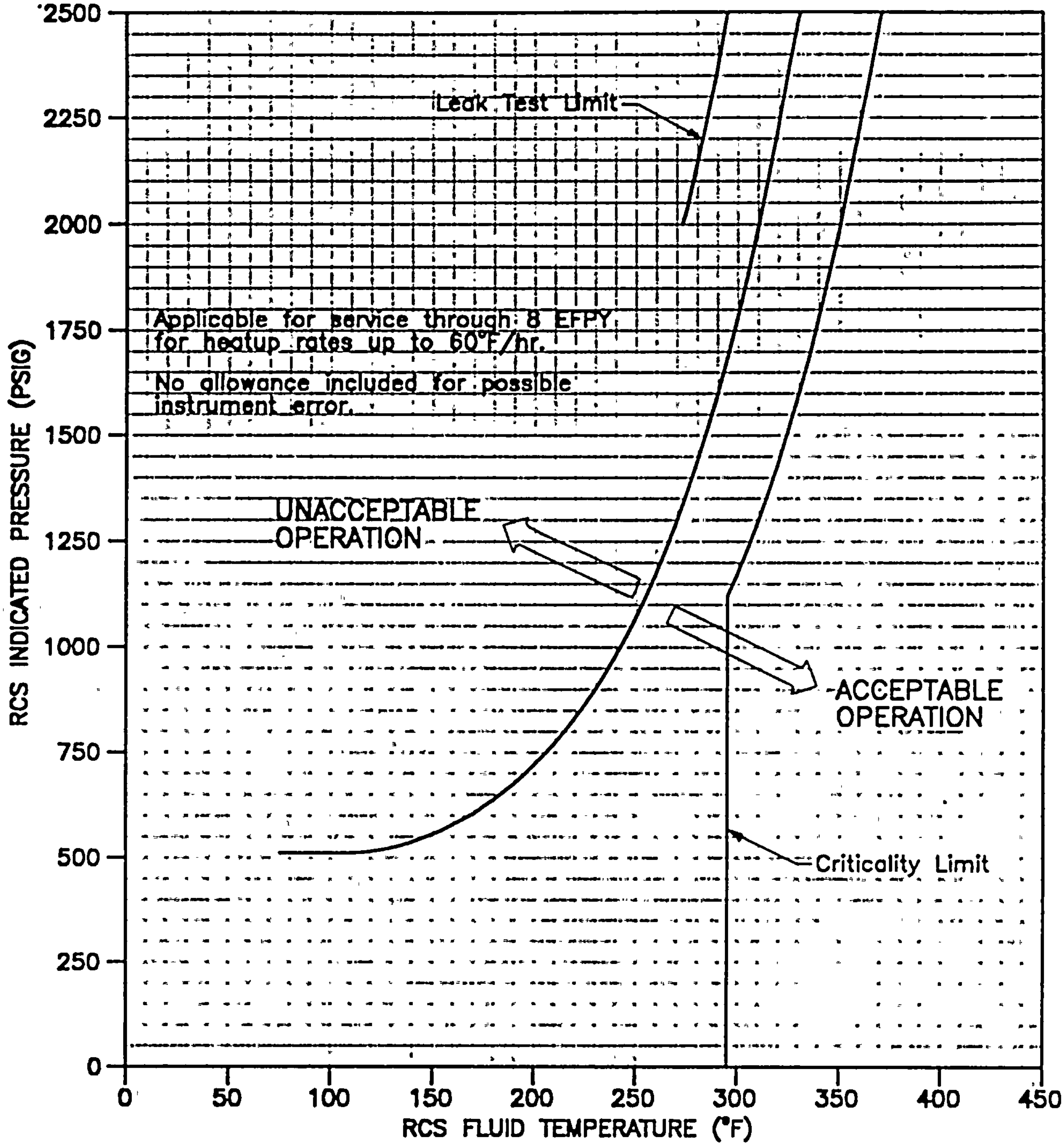
ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity 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°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 hour 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, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3, and the setpoint of Technical Specification 3.4.9.3.a.~~

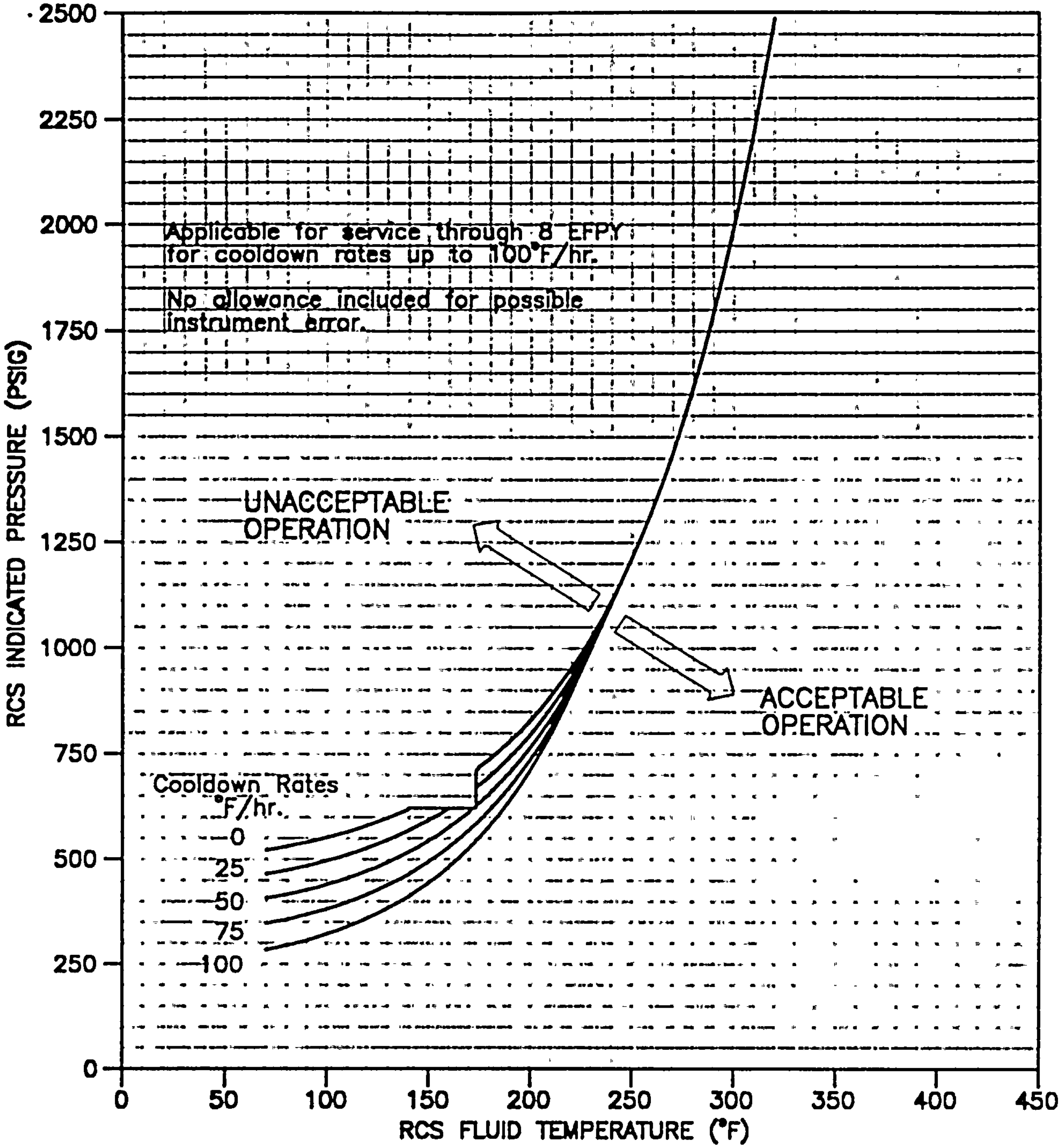


Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt.% Cu 0.59wt.% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

Figure 3.4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 8 EFPY



Controlling Material:

Unit 2 Intermediate Shell Plate B5454-2 0.14wt.% Cu 0.59wt.% Ni

Initial RT_{NDT}=67°F Projected RT_{NDT} 1/4T = 164°F 3/4T = 141°F

Figure 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 8 EYPY

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TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 1

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
S	320°	3.7	First refueling outage
T	140°	3.7	Standby
U	356°	1.1	12
V	184°	1.1	24
W	4°	1.1	38
X	176°	1.1	50
Y	40°	3.7	Standby
Z	220°	3.7	Standby

UNIT 2

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	56°	4.8	First refueling outage
X	236°	4.8	3
V	58.5°	4.0	6
Y	238.5°	4.0	10
W	124°	4.8	15
Z	304°	4.8	Standby

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Presure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI. *Insert*

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1966 Edition for Unit 1 and the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil ductility reference temperature, RT_{NDT}, at the end of ~~6~~ 8 effective full power years (EFPY) of service life. The ~~6~~ 8 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region

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Allowable pressures and temperatures for inservice leak and hydrostatic tests are given in Figure 3.4-2.

6. The criticality limit on Figure 3.4-2 is based on the minimum allowable temperature of 295°F for an inservice hydrostatic test of 110% of operating pressure.

TABLE B 3/4.4-1a

REACTOR VESSEL TOUGHNESS DATA-UNIT 1

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COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (Wt%)	P (Wt%)	NDTT °F	MINIMUM 50 FT-LB/35 Mil TEMP °F LONG	MINIMUM 50 FT-LB/35 Mil TEMP °F TRANS	RT_{NDT} °F	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
Cl.Hd.Dome	B4108	A533B1			-30	51	71*	11		72*
Cl.Hd.Seg.	B4109-1	A533B1			0	53	73*	13		81*
Cl.Hd.Seg.	B4109-2	A533B1			-10	50	70*	10		89*
Cl.Hd.Seg.	B4109-3	A533B1			-20	50	70*	10		7[illegible]
Cl.Hd.Flg.	B4102	A508,2			53*	30	50*	53		103*
Ves.Sh.Flg.	B4101	A508,2			35*	-5	15*	35		99*
Inlet Noz.	B4103-1	A508,2			60*	17	37*	60		77*
Inlet Noz.	B4103-2	A508,2			60*	27	47*	60		75*
Inlet Noz.	B4103-3	A508,2			43*	10	30*	43		108*
Inlet Noz.	B4103-4	A508,2			48*	2	22*	48		106*
Outlet Noz.	B4104-1	A508,2			60*	-13	7*	60		77*
Outlet Noz.	B4104-2	A508,2			43*	-3	17*	43		74*
Outlet Noz.	B4104-3	A508,2			54*	-12	8*	54		86*
Outlet Noz.	B4104-4	A508,2			60*	30	50*	60		84*
Upper Shl.	B4105-1	A533B1	0.12	0.010	10	68	88*	28		80*
Upper Shl.	B4105-2	A533B1	0.12	0.008	0	49	69*	9		74*
Upper Shl.	B4105-3	A533B1	0.14	0.010	0	54	74*	14		81*
Inter.Shl.	B5106-1	A533B1	0.14	0.013	-10	57	40	-10	134	116
Inter.Shl.	B4106-2	A533B1	0.13	0.013	-10	36	57	- 3	132	114
Inter.Shl.	B4106-3	A533B1	0.10	0.011	10	70	90*	30	119	7[illegible]*
Lower Shl.	B4107-1	A533B1	0.13	0.011	-10	59	75	15	127	110
Lower Shl.	B4107-2	A533B1	0.12	0.010	-10	64	80	20	127	108
Lower Shl.	B4107-3	A533B1	0.12	0.010	-50	52	38	-22	135	116
Bot.Hd.Seg.	B4111-1	A533B1			-20	33	53*	- 7		82*
Bot.Hd.Seg.	B4111-2	A533B1			-40	16	36*	-14		90*
Bot.Hd.Seg.	B4111-3	A533B1			-40	21	41*	-19		85*
Bot.Hd Seg.	B4110	A553B1			-10	60	80*	20		75*

*Estimated per NRC Standard Review Plan Section 5.3.2.

TABLE B 3/4.4-1b

REACTOR VESSEL TOUGHNESS DATA-UNIT 2   DELETE

COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (Wt%)	P (Wt%)	NDTT °F	MINIMUM 50 FT-LB/35 Mil TEMP °F LONG	MINIMUM 50 FT-LB/35 Mil TEMP °F TRANS	RT_{NDT} °F	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
Cl.Hd.Dome	B5457	A533BCL1			-20	35	55*	-5	135	88*
Cl.Hd.Seg.	B5456-1	A533BCL1			-50	23	43*	-17	134	87*
Cl.Hd.Seg.	B5456-2	A533BCL1			-20	62	82*	22	131	[illegible]*
Cl.Hd.Seg.	B5456-3	A533BCL1			-20	15	35*	-20	124	81*
Cl.Hd.Flg.	B5452	A508CL2			20	15	35*	20	151	98*
Ves.Sh.Flg.	B5451	A508CL2			-10	-10	10*	-10	158	103*
Inlet Noz.	B5461-1	A508CL2			-20	23	43*	-17	116	75
Inlet Noz.	B5461-2	A508CL2			-20	- 2	18*	-20	119	77*
Inlet Noz.	B5461-3	A508CL2			-40	-45	-25*	-40	127	83*
Inlet Noz.	B5461-4	A508CL2			-40	-48	-28*	-40	129	84*
Outlet Noz.	B5462-1	A508CL2			-50	- 4	16*	-44	145	94
Outlet Noz.	B5462-4	A508CL2			-40	-10	10*	-40	137.5	89*
Outlet Noz.	B5462-2	A508CL2			-40	14	34*	-26	135.5	88*
Outlet Noz.	B5462-3	A508CL2			-50	17	37*	-23	131.5	85*
Upper Shl.	B5453-1	A533BCL1			0	85	88	28	92	82
Upper Shl.	B5453-3	A533BCL1			-10	45	65*	5	136.5	89*
Upper Shl.	B5011-1	A533BCL1			10	40	60*	0	110	72*
Inter.Shl.	B5454-1	A533BCL1	0.15	0.010	-40	14	112	52	128	91
Inter.Shl.	B5454-2	A533BCL1	0.14	0.012	0	60	127	67	113	[illegible]
Inter.Shl.	B5454-3	A533BCL1	0.15	0.012	-40	30	93	33	129	90
Lower Shl.	B5455-1	A533BCL1	0.14	0.010	-20	42	45	-15	134	112
Lower Shl.	B5455-2	A5338CL1	0.14	0.011	0	25	45	0	137	122
Lower Shl.	B5455-3	A533BCL1	0.10	0.010	0	55	75	15	128	100
Bot.Hd.Seg.	B5009-2	A533BCL1			-10	110	130*	70	85	55*
Bot.Hd.Seg.	B5009-3	A533BCL1			-20	-12	8*	-20	131	84
Bot.Hd.Seg.	B5009-1	A533BCL1			0	88	108*	48	95	62*
Bot.Hd.Seg.	B5010	A533BCL1			-30	20	40*	-20	114	74

*Estimated per NRC Standard Review Plan Section 5.3.2.

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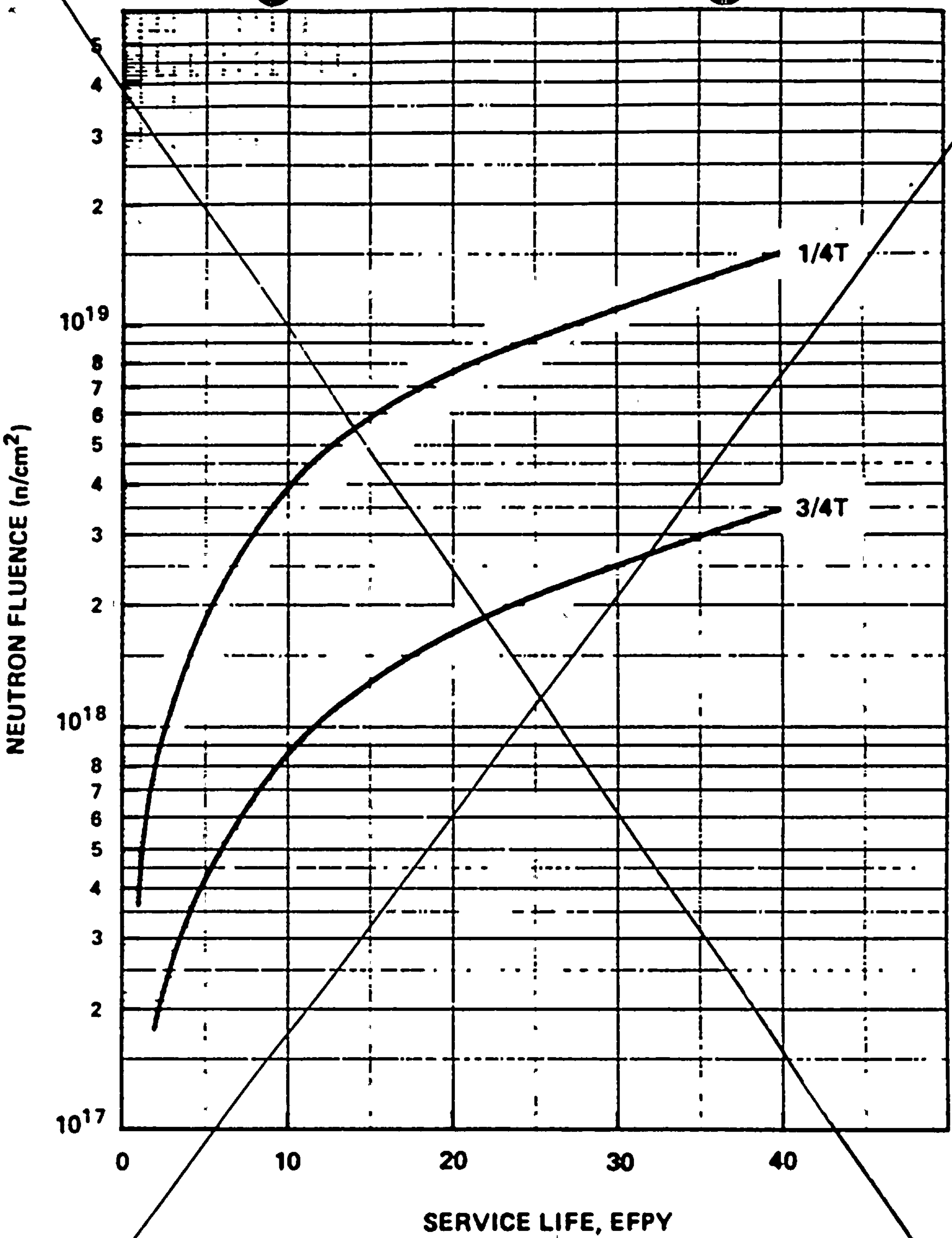

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

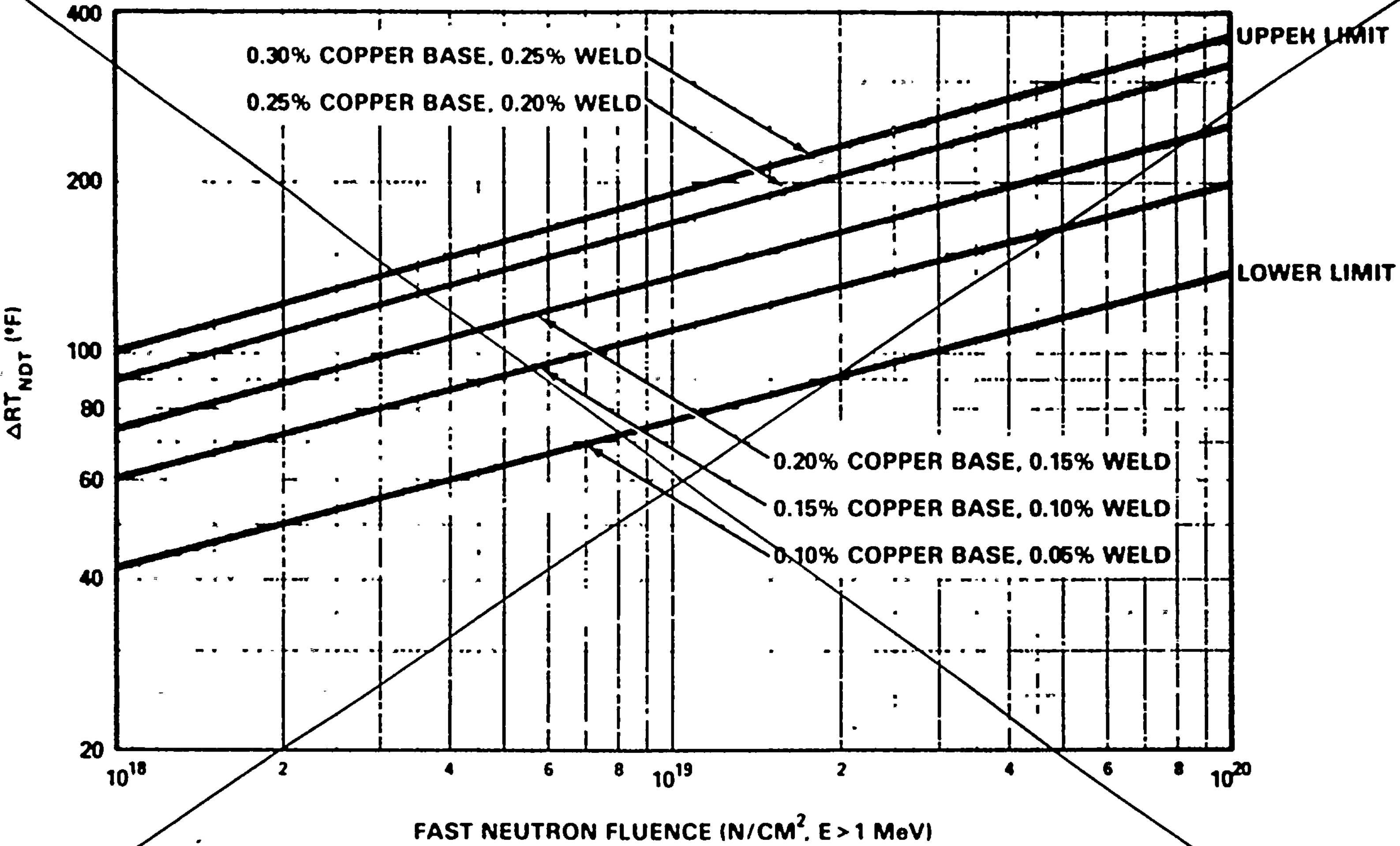

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FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

for the maximum neutron fluence at the locations of interest.

is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in ~~Table B 3/4.4-1a for Unit 1 and Table B 3/4.4-1b for Unit 2~~ the FSAR Update. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and ~~phosphorous~~ nickel content of the material in question, can be predicted using ~~Figure B 3/4.4-1 and the largest~~ value of ΔRT_{NDT} computed by ~~either~~ Regulatory Guide 1.99, Revision ~~1~~ 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," ~~or the Westinghouse Copper Trend Curves shown in Figure 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 ~~6~~ 8 EFPY.

Values of ΔRT_{NDT} determined in this manner ~~may~~ will be used until the results from the material surveillance program, evaluated according to ASTM E185-~~73~~ 82, ~~are available~~ can be used. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule ~~is shown in Table 4.4-5~~ will be maintained in the FSAR Update. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the

PRESSURE/TEMPERATURE LIMITS (Continued)

calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where, K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = ~~constant~~ reference stress intensity factor provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature of the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factors, K_{It}, for the reference flaw ~~is~~ are computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.