Attachment 1

Proposed Changed Pages

Unit 2

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	R. V. INTERMEDIATE SHELL
COPPER CONTENT	:	0.20 WT%
PHOSPHORUS CONTENT	:	0.018 WT%
INITIAL RT_{NDT}	:	$-10^{\circ}F$
RT_{NDT} AFTER 8 EFPY	:	1/4T, $146^{\circ}F$
	:	3/4T, $83^{\circ}F$

CURVES APPLICABLE FOR HEATUP RATES UP TO $60^{\circ}F/HR$ FOR THE SERVICE PERIOD UP TO 8 EFPY

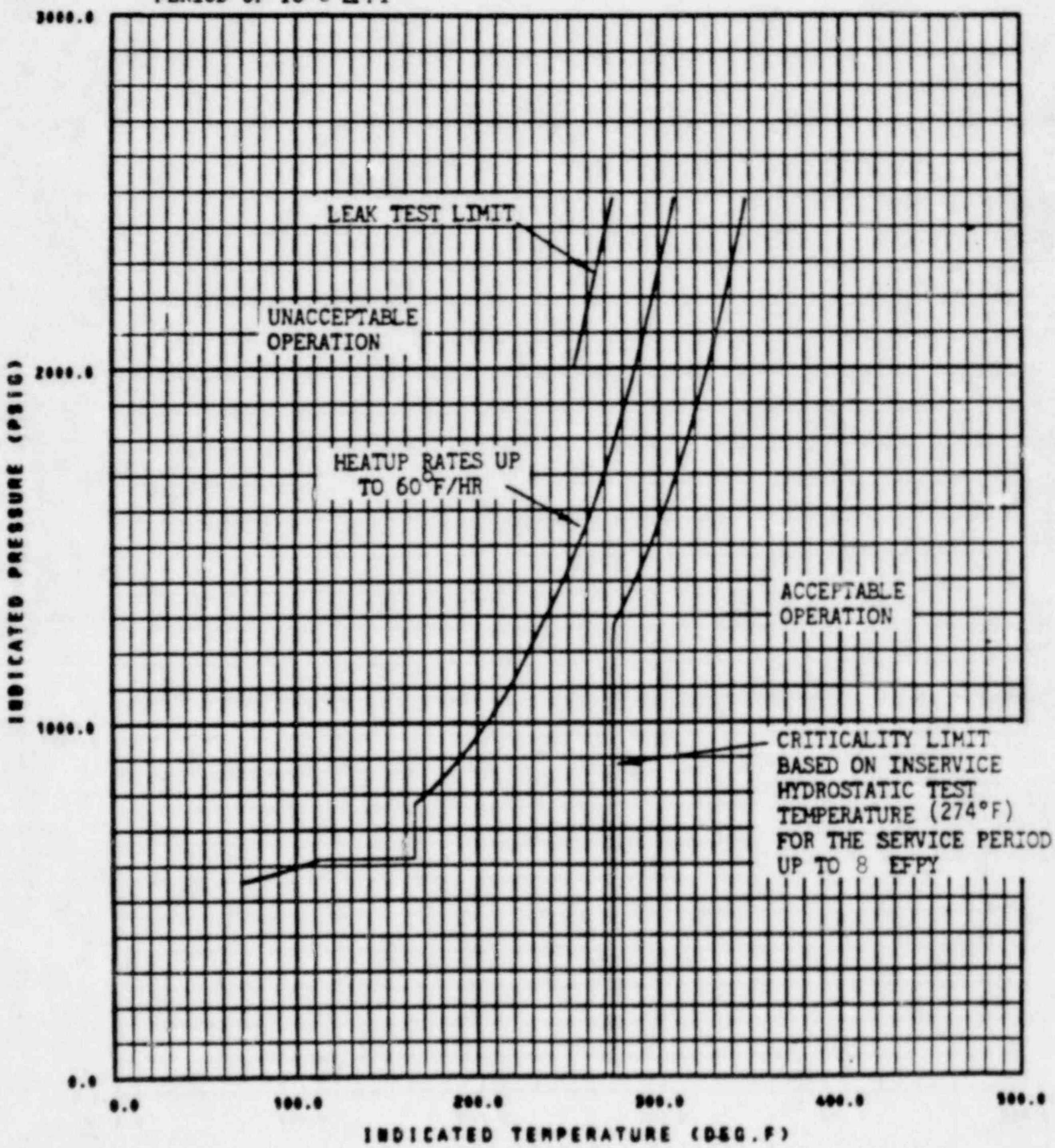


FIGURE 3.4-2 FARLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	R. V. INTERMEDIATE SHELL
COPPER CONTENT	:	0.20 WT%
PHOSPHORUS CONTENT	:	0.018 WT%
INITIAL RT_{NDT}	:	$-10^{\circ}F$
RT_{NDT} AFTER 8 EFPY	:	1/4T, $146^{\circ}F$
	:	3/4T, $83^{\circ}F$

CURVES APPLICABLE FOR COOLDOWN RATES UP TO $100^{\circ}F/HR$ FOR THE SERVICE PERIOD UP TO 8 EFPY

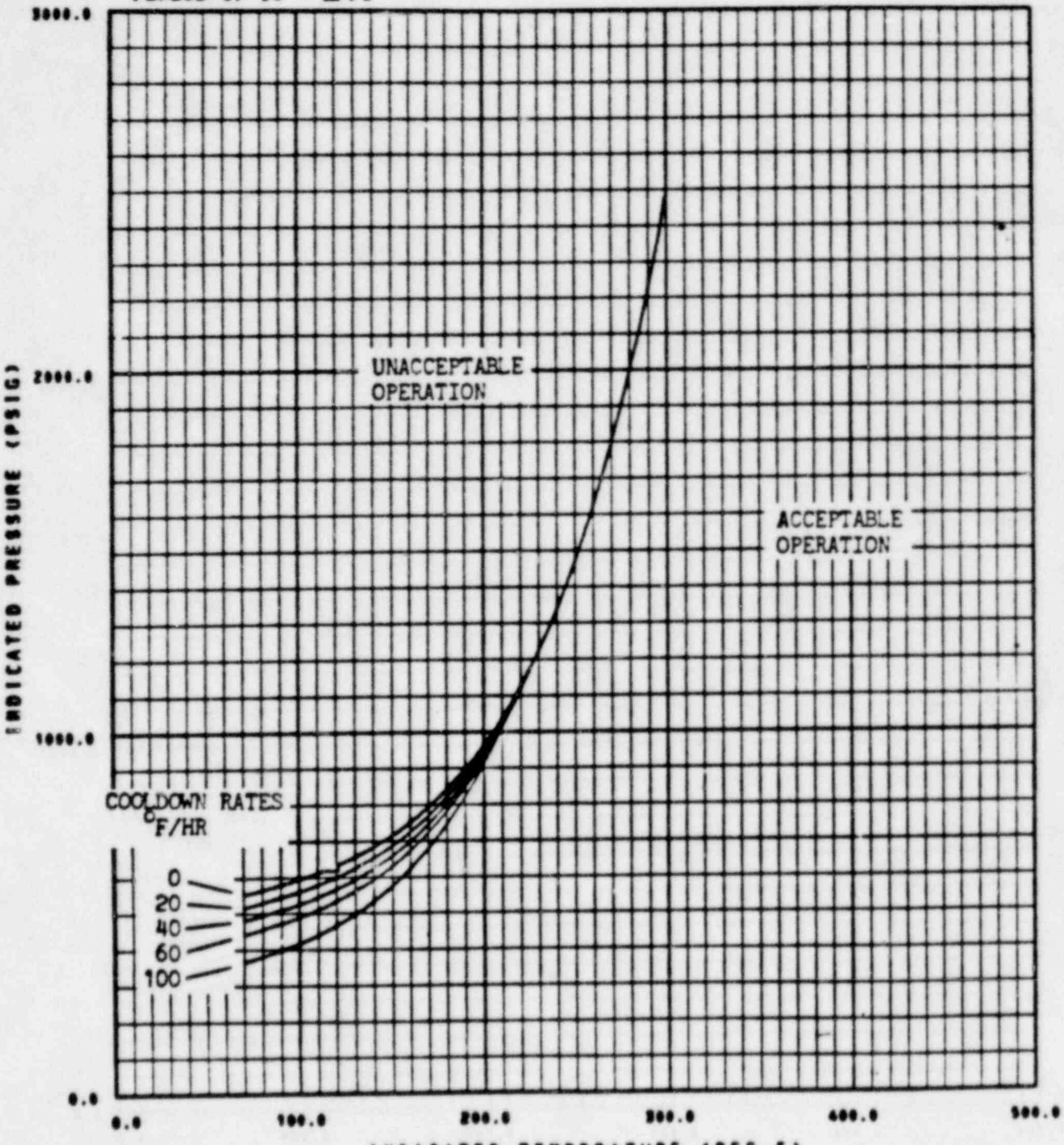


FIGURE 3.4-3 FARLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 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 Pressure Vessel Code, Section III, Appendix G as required per 10CFR Part 50 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 first full-power service period.

   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.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nonductile 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 320°F.

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.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of 8 effective full power years of service life. The 8 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region 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 3/4.4-1. 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 and copper content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials". 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 8 EFPY.

Values of ΔRT_{ndt} determined in this manner may be used until the next results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the next 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 WCAP-7924-A.

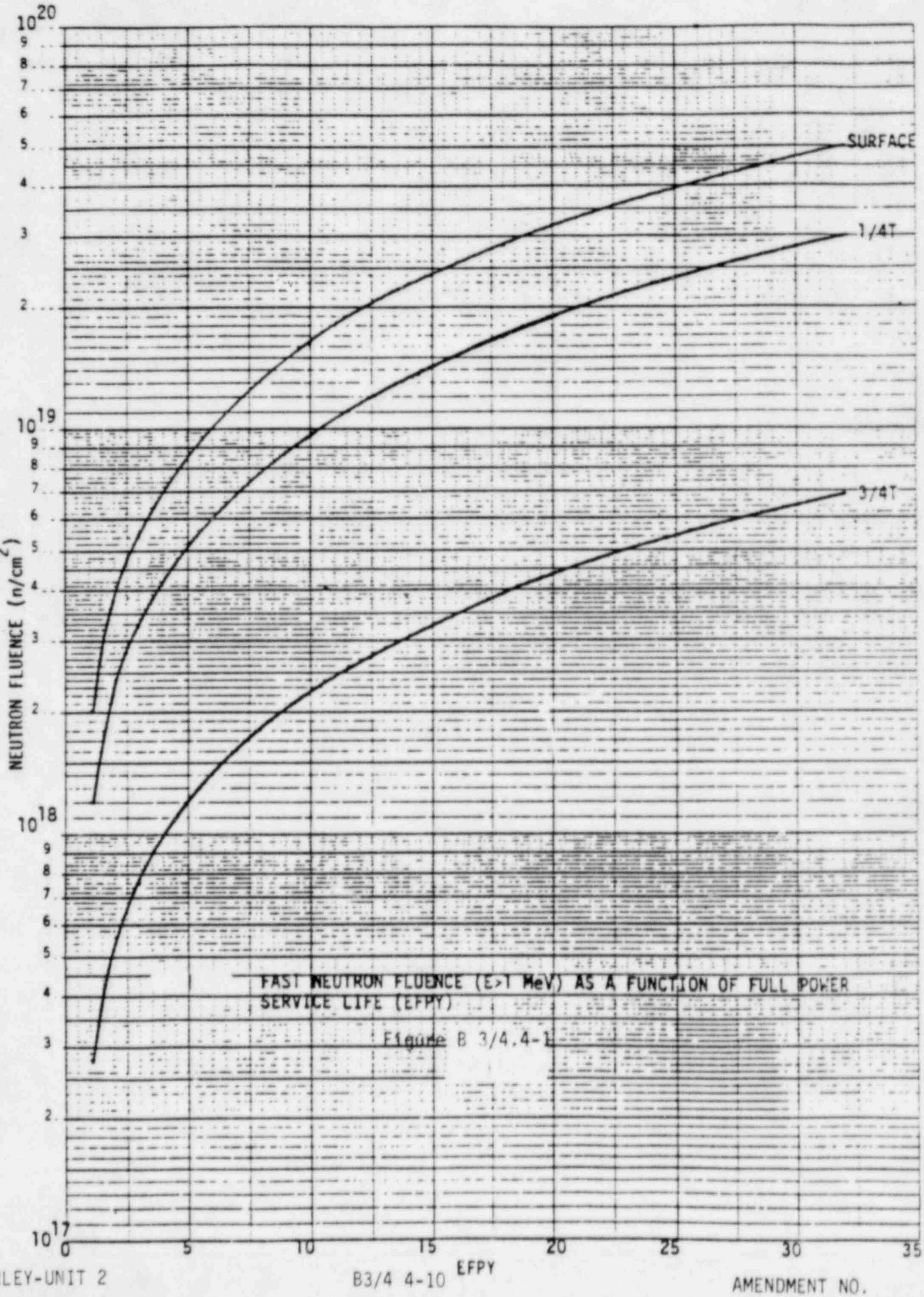
The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed 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 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.

TABLE B3/4.4-1

REACTOR VESSEL TOUGHNESS DATA

Component	Code No.	Grade	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
CL. HD. Dome	B7215-1	A533,B,CL.1	0.17	0.010	0.49	-30	16[a]	83[a]	128
CL. HD. Flange	B7207-1	A508,CL.2	0.14	0.011	0.65	60[a]	60[a]	>56[a]	>86[c]
VES. Flange	B7206-1	A508,CL.2	0.10	0.012	0.67	60[a]	60[a]	>71[a]	>109
Inlet Noz.	B7218-2	A508,CL.2	-	0.010	0.68	50[a]	50[a]	103[a]	158
Inlet Noz.	B7218-1	A508,CL.2	-	0.010	0.71	32[a]	32[a]	112[a]	172
Inlet Noz.	B7218-3	A508,CL.2	-	0.010	0.72	60[a]	60[a]	98[a]	150
Outlet Noz.	B7217-1	A508,CL.2	-	0.010	0.73	60[a]	60[a]	100[a]	154
Outlet Noz.	B7217-2	A508,CL.2	-	0.010	0.72	6[a]	6[a]	108[a]	167
Outlet Noz.	B7217-3	A508,CL.2	-	0.010	0.72	48[a]	48[a]	103[a]	158
Upper Shell	B7216-1	A508,CL.2	-	0.010	0.73	30	30[a]	97[a]	149
Inter Shell	B7203-1	A533,B,CL.1	0.14	0.010	0.60	-40	15	99	140
Inter Shell	B7212-1	A533,B,CL.1	0.20	0.018	0.60	-30	-10	99	134
Lower Shell	B7210-1	A533,B,CL.1	0.13	0.010	0.56	-40	18	103	128
Lower Shell	B7210-2	A533,B,CL.1	0.14	0.015	0.57	-30	0	99	145
Trans. Ring	B7208-1	A508,CL.2	-	0.010	0.73	40	40[a]	89[a]	137
Bot. HD. Dome	B7214-1	A533,B,CL.1	0.11	0.007	0.48	-30	-2[a]	87[a]	134
Inter. Shell	A1.46	SMAW	0.02	0.009	0.96	0[a]	0[a]	>131	-
Long Seams	A1.40	SMAW	0.02	0.010	0.93	-60	-60	>106	-
Inter Shell to Lower Shell	G1.50	SAW	0.13	0.016	<.20[b]	-40	-40	>102	-
Lower Shell Long Seams	G1.39	SAW	0.05	0.006	<.20[b]	-70	-70	>126	-

(a) Estimate per NUREG 0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
(b) Estimated.
(c) Upper shelf not available, value represents minimum energy at the highest test temperature.



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

Figure B 3/4.4-1

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 8 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHR relief valves or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief 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 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10CFR50.44(c)(3)(iii).

Attachment 2

"Heatup and Cooldown Limit Curves for the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel", WCAP 10910, Revision 1

Westinghouse Nuclear Energy Systems