Attachment

Unit 1

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TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
Y	343°	3.12	Removed 1.13 EFPY
U	107°	3.12	Removed 3.02 EFPY
X	287°	3.12	6 EFPY
W	110°	2.70	12 EFPY
V	290°	2.70	21 EFPY
Z	340°	2.70	Standby

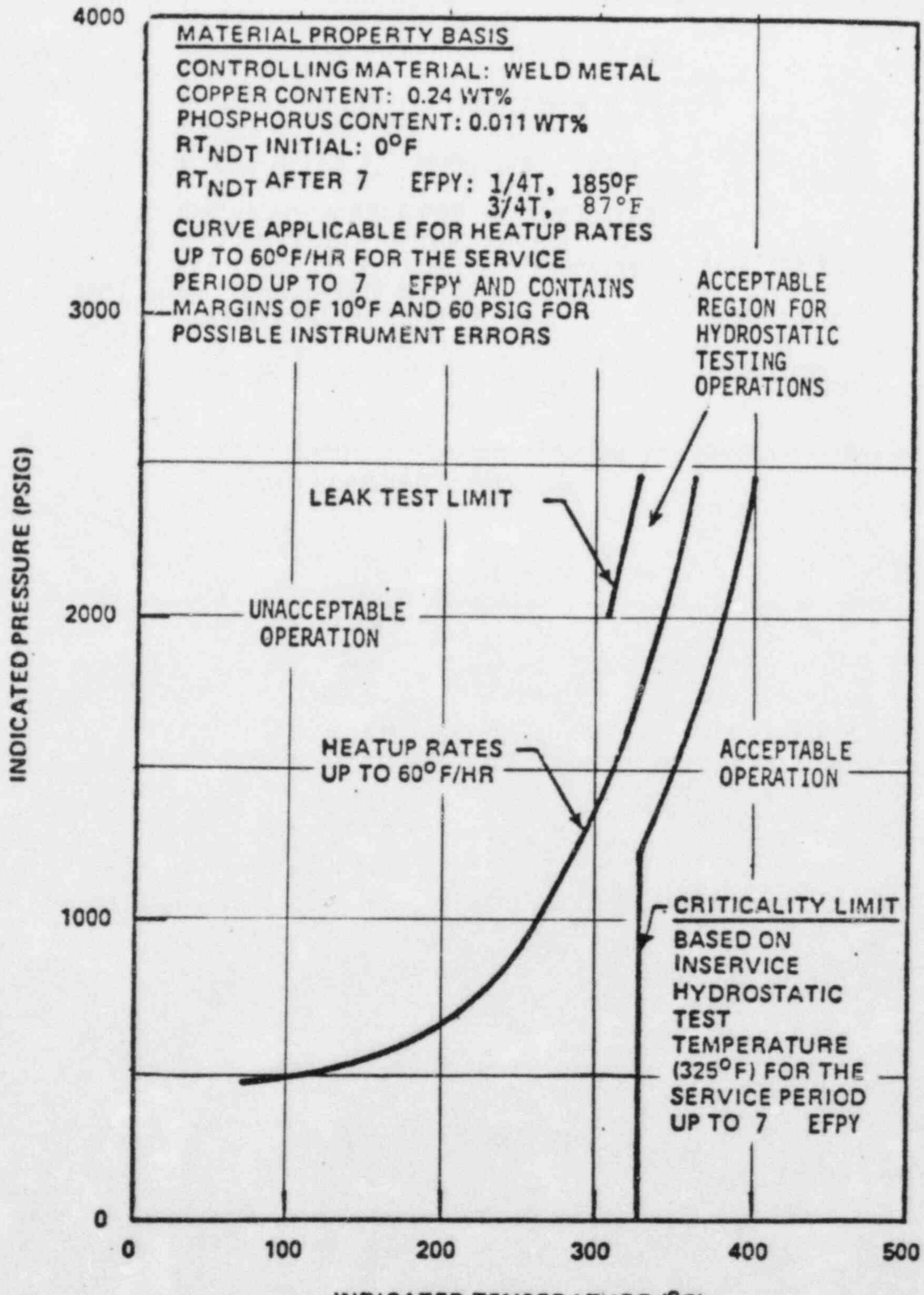

Figure 3.4-2 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 7 EFPY

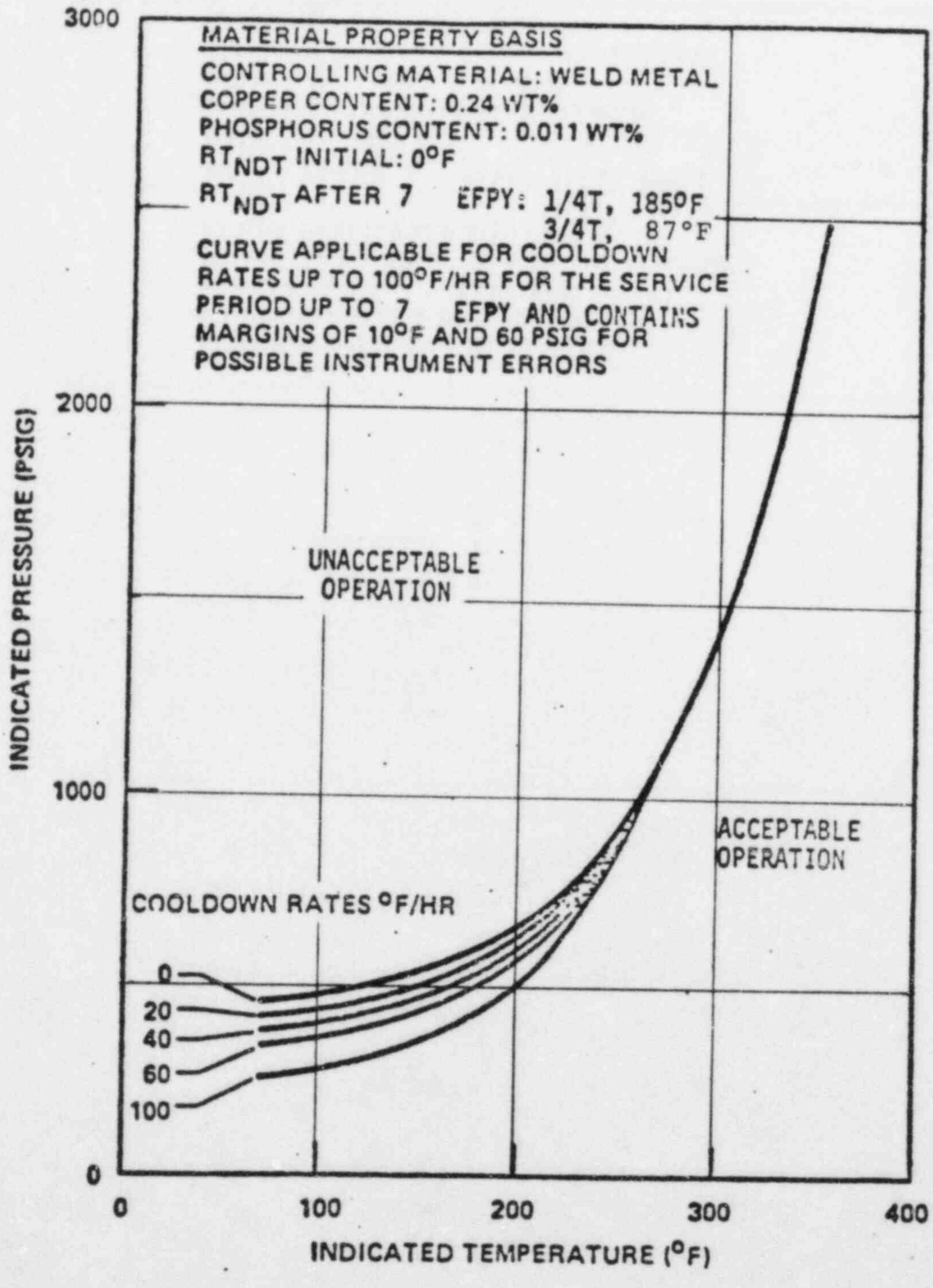

Figure 3.4-3 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 7 EFPY

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 7 effective full power years of service life. The 7 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 Predicated 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 7 EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

Values of ΔRT_{ndt} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10CFR50, 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 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 10CFR 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 B 3/4.4-1

REACTOR VESSEL TOUGHNESS DATA

Component	Code No.	Material Type	Cu (%)	P (%)	T_{NDT} (°F)	MWD (°F)	NMWD (°F)	RT_{NDT} (°F)	Upper Shelf Energy MWD	Upper Shelf Energy NMWD
Closure head dome	B6901	A533B, Cl.1	0.16	0.009	-30	20	40[a]	-20	140	-
Closure head segment	B6902-1	A533B, Cl.1	0.17	0.007	-20	-10	10[a]	-20	138	-
Closure head flange	B6915-1	A508, Cl.2	0.10	0.012	60[a]	-20	0(a)	60	75[a]	-
Vessel flange	B6913-1	A508, Cl.2	0.17	0.011	60[a]	-30	-10[a]	60	106[a]	-
Inlet nozzle	B6917-1	A508, Cl.2	-	0.010	60[a]	-	45	60	-	110
Inlet nozzle	B6917-2	A508, Cl.2	-	0.008	60[a]	-	115	60	-	80
Inlet nozzle	B6917-3	A508, Cl.2	-	0.008	60[a]	-	35	60	-	98
Outlet nozzle	B6916-1	A508, Cl.2	-	0.007	60[a]	-	60	60	-	96.5
Outlet nozzle	B6916-2	A508, Cl.2	-	0.011	60[a]	-	30	60	-	97.5
Outlet nozzle	B6916-3	A508, Cl.2	-	0.009	60[a]	-	50	60	-	100
Nozzle shell	B6914-1	A508, Cl.2	-	0.010	30	70	90[a]	30	148	-
Inter. shell	B6903-2	A533B, Cl.1	0.13	0.011	0	-25	40	0	151.5	97
Inter. shell	B6903-3	A533B, Cl.1	0.12	0.014	10	5	52	10	134.5	100
Lower shell	B6919-1	A533B, Cl.1	0.14	0.015	-20	-5	75	15	133	90.5
Lower shell	B6919-2	A533B, Cl.1	0.14	0.015	-10	0	65	5	134	97
Bottom head ring	B6912-1	A508, Cl.1	-	0.010	10	-25	-5[a]	10	163.5	-
Bottom head segment	B6906-1	A533B, Cl.1	0.15	0.011	-30	-50	-30[a]	-30	147	-
Bottom head dome	B6907-1	A533B, Cl.1	0.17	0.014	-30	-10	-10[a]	-30	143.5	-
Inter. shell long. weld seams			0.27	0.015	0[a]	-	<60	0	-	-
Inter. lo lower shell weld seam			0.24	0.011	0[a]	-	<60	0	-	-
Lower shell long. weld seam			0.17	0.022	0[a]	-	<60	0	-	-

(a) Estimated per NRC Regulatory Standard Review Plan, section 5.3.2.

MWD - Major Working Direction
NMWD - Normal to Major Working Direction