ATTACHMENT 2

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HEATUP AND COOLDOWN LIMIT CURVES FOR THE ALABAMA POWER COMPANY JOSEPH M. FARLEY UNIT 1 REACTOR VESSEL

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1. INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature). The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (T_{NDT}) or the temperature at which the material exhibits at least 50 ft lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper, nickel and phosphorus) present in reactor vessel steels. Westinghouse, other NSSS vendors, the U.S. Nuclear Regulatory Commission and others have developed trend curves for predicting adjustment of RT_{NDT} as a function of fluence and copper, nickel and/or phosphorus content. The Nuclear Regulatory Commission (NRC) trend curve is published in Regulatory Guide 1.99 (Effects of Residual Elements on Predicting Radiation Damage to Reactor Vessel Materials)[1]. Regulatory Guide 1.99 was originally published in July 1975 with a Revision 1 being issued in April 1977. Currently, a Revision 2 to Regulatory Guide 1.99 is under consideration within the NRC. The chemistry factor, "CF", °F, a function of copper and nickel content identified in Regulatory Guide 1.99, Revision 2 is given in Table I for welds and Table II for base metal (plates and forgings). Interpolation is permitted. The value, "f", given in Figure 1 is the calculated value of the neutron fluence at the location of interest (inner surface, 1/4T, or 3/4T) in the vessel at the location of the postulated defect, n/cm^2 (E > MeV) divided by 10^{19}. The fluence factor is determined from Figure 1.

Given the copper and nickel contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Tables I and II and Figure 1. The maximum fast-neutron fluence (E > 1 MeV) at the inner surface, 1/4T (wall thickness) and 3/4T (wall thickness) vessel locations is given as a function of full-power service in Figure 2 and 3 for the vessel core region girth weld and longitudinal welds, respectively, and plates shown in Figure 4. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to ensure that no other component will be limiting with respect to RT_{NDT}.

2. FRACTURE TOUGHNESS PROPERTIES

The preirradiation fracture-toughness properties of the Farley Unit 1 reactor vessel materials are presented in Table III. A review of submerged arc welding practice at the time of vessel fabrication by Combustion Engineering showed that B-4 type weld wire was used and the as deposited nickel content of the welds resulted in low nickel (~0.20% Ni). The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[2]. The postirradiation fracture-toughness properties of the reactor vessel beltline material were obtained directly from the Farley Unit 1 Vessel Material Surveillance Program.

3. FLUENCE CALCULATIONS

For the purpose of revising heatup and cooldown curves for Farley Unit 1, it is necessary to know vessel fast fluence (ϕ (E > 1 MeV)) at the azimuthal peak location which has limiting embrittlement characteristics. This peak location is at 0° for plates and girth welds and 45° for longitudinal weld seams shown in Figure 4. The calculations performed for this purpose consist of adjoint analyses, relating the fast flux (ϕ (E > 1 MeV)) at the vessel IR to the power distributions in the reactor core. The adjoint (importance) functions used, when combined with cycle specific core power distributions, yield the plant specific exposure data for each operating fuel cycle.

The adjoint function was generated using the DOT discrete ordinates code[3] and the SAILOR cross-section library[4]. The SAILOR library is a 47 group, ENDF-B/IV based data set produced specifically for light water reactor applications. In generating the adjoint function, anisotropic scattering was treated with a P_3 expansion of the cross-sections. The adjoint source location was chosen along the inner diameter of the pressure vessel. This calculation was run in R, θ geometry to provide a power distribution importance function for the exposure parameter of interest (ϕ (E > 1 MeV)). Having the adjoint importance function and appropriate core power distributions, the response of interest is calculated as

$$R_{R,\theta} = \int_R \int_\theta I(R,\theta)\ F(R,\theta)\ R\ dR\ d\theta$$

where:

$R_{R,\theta}$ = Response of interest (ϕ (E > 1.0 MeV), dPa, etc.) at radius R and azimuthal angle θ.

$I(R,\theta)$ = Adjoint importance function at radius R and azimuthal angle θ.

$F(R,\theta)$ = Full power fission density at radius R and azimuthal angle θ.

It should be noted that as written in the above equation, the importance function I(R,θ) represents an integral over the fission distribution so that the response of interest can be related directly to the spatial distribution of fission density within the reactor core.

Core power distributions for Farley Unit 1 were taken from the following Westinghouse fuel cycle design reports for each operating cycle to date:

Fuel Cycle	Report
1	WCAP-8515 and Ref. 5
2	Ref. 5 and WCAP-9761
3	WCAP-9761 and WCAP-10036
4A	WCAP-10036 and WCAP-10308
5	WCAP-10308 and WCAP-10525
6	WCAP-10525 and WCAP-10795
7	WCAP-10795 and Ref. 6

Of these, Cycles 1 through 4A utilized out-in fuel loading patterns, and Cycles 5, 6 and 7 implemented low leakage fuel loading patterns.

The power distributions employed represent cycle averaged relative assembly powers. Therefore, the adjoint results are in terms of fuel cycle averaged neutron flux, which when multiplied by the fuel cycle length yields the incremental fast neutron fluence. Fast fluences at 1/4T and 3/4T are obtained from those at vessel IR through fast flux ratios obtained from the DOT transport analysis performed in support of WCAP-10474, "Analysis of Capsule U from the Alabama Power Company, Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program". As a result, the following neutron fluences for E > 1.0 MeV were calculated:

Cumulative Lifetime	Fluence (E > 1 MeV) at 0° (n/cm^2)		
EFPY	Vessel IR	Vessel 1/4T	Vessel 3/4T
1.05	1.943×10^{18}	1.151×10^{18}	2.674×10^{17}
1.81	3.526×10^{18}	2.088×10^{18}	4.852×10^{17}
2.19	4.231×10^{18}	2.505×10^{18}	5.822×10^{17}
2.98	5.738×10^{18}	3.398×10^{18}	7.896×10^{17}
3.81	7.041×10^{18}	4.169×10^{18}	9.688×10^{17}
4.72	8.296×10^{18}	4.913×10^{18}	1.142×10^{18}
4.79	8.406×10^{18}	4.978×10^{18}	1.157×10^{18}
32.00	5.037×10^{19}	2.983×10^{19}	6.932×10^{18}

Projection of fluence at 32.0 EFPY was made assuming a power distribution unchanged from that used in Cycle 7 (i.e., that which generated the fluence estimate for 4.79 EFPY).

4. CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

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 and 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(7). The K_{IR} curve is given by the equation:

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

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}. Thus, the governing equation of the heatup-cooldown analysis is defined in Appendix G to the ASME Code(7) as follows:

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

where:

K_{IM} is the stress intensity factor caused by membrane (pressure) stress

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

K_{IR} is a function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value of 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 are computed. From Equation (2), the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation.

Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, 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. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves in Figures 5 through 8. In addition, heatup and cooldown curves without instrument errors are presented in Figures 9 through 12.

The Farley Unit 2 fracture analysis results from Reference 8 are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. As a result, the 16 EFPY and 32 EFPY cooldown curves with and without instrument errors are impacted by the 10CFR50 rule as shown by Figures 6, 8, 10, and 12; also, the heatup curves without instrument errors for 16 and 32 EFPY and with instrument error for 16 EFPY are impacted by the 10CFR50 rule as shown by Figures 5, 9 and 11. However, the heatup curve with instrument errors for 32 EFPY shown by Figure 7 is not impacted by the 10CFR50 rule. Since there are many conservatisms (safety factor of 2 on pressure, K_{IR} toughness and 1/4T flaw) built into the ASME Appendix G analysis method[7], Appendix G does not require that instrument error margins be included in the analysis. Therefore, plant operation can be based on heatup and cooldown curves without instrument errors.

An evaluation has been performed to determine the acceptability of the Overpressure Mitigation System (OMS) presently in Farley Unit 1 (Technical Specification 3/4.4.10.3) with respect to the 16 EFPY heatup and cooldown curves shown in Figures 9 and 10 respectively. For the purpose of the evaluation it was assumed that the RHR relief valve lifts at 495 psig which includes 10% accumulation. The heatup curve in Figure 9 does not fall below 495 psig at any temperature. A comparison of cooldown curves in Figure 10 shows that in the low temperature range (<130°F) cooldown rates of 20°F/hr and lower fall well above 495 psig. Although the cooldown curves for rates of 40°F/hr and above do fall below 495 psig, it is not expected that the Appendix G curves will be violated during an actuation of the OMS since cooldown rates greater than or equal to 40°F/hr are highly unlikely at low temperature conditions. Therefore, the Appendix G curves as illustrated in Figures 9 and 10 will not be violated as the result of an actuation of the OMS.

5. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed previously. The derivation of the limit curves is presented in the NRC Regulatory Standard Review Plan(2).

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown on the heatup and cooldown curves. The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line, shown in Figures 5, 7, 9 and 11. This is in addition to other criteria which must be met before the reactor is made critical.

The leak test limit curve shown in Figures 5, 7, 9 and 11 represent minimum temperature requirements at the leak test pressure specified by applicable codes(2,7).

6. AVAILABLE SURVEILLANCE CAPSULE DATA AND ADJUSTED REFERENCE TEMPERATURE

Credible surveillance data is available for lower plate B6919-1[9,10], however, data for the weld metal is not considered credible since it was not fabricated with the same heat of weld wire and lot of flux as used in the vessel beltline region. The following surveillance data was used to determine a chemistry factor (CF) for plate B6919-1 using USNRC Regulatory Guide 1.99 Rev. 2.[11]

From Regulatory Guide 1.99 Rev. 2 the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{3}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of Initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} \text{ surface} = [CF]f^{(0.28-0.10 \log f)} \tag{4}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following attenuation formula was used:

$$\Delta RT_{NDT} = [\Delta RT_{NDT} \text{ surface}]e^{-0.067x} \tag{5}$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface.

CF (°F) is the chemistry factor, a function of copper and nickel content. CF is given in Table I for welds and in Table II for base metal (plates and forgings). Linear interpolation is permitted. In Tables I and II "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld.

Surveillance Base Material Data (Plate B6919-1)

Fluence (n/cm^2)	ΔRT_{NDT} °F	Fluence Factor (FF)	ΔRT_{NDT} x FF
5.83×10^{18}	55	0.849	46.695
1.65×10^{19}	90	1.138	102.420
5.83×10^{18}	85	0.849	72.165
1.65×10^{19}	105	1.138	119.49
			340.77

Sum of the squares of the fluence factor = 4.0317.

$$\text{Then Chemistry Factor (CF)} = \frac{\Delta RT_{NDT} \times FF}{\sum (FF)^2} = \frac{340.77}{4.0317} = 84.54$$

Beltline Base Material Evaluation

Plate No.	Initial RT_{NDT}(°F)	Cu	Ni	CF	32 EFPY ΔRT_{NDT} (°F) Surface	1/4 T*	3/4 T*
B6903-2	0	.13	.60	91	127.8	112.0	86.0
B6903-3	10	.12	.56	82.2	115.4	101.1	77.7
B6919-1	15	.14	.55	84.5	118.7	104.0**	79.9**
B6919-2	5	.14	.56	98.2	137.9	120.9	92.8

*$\Delta RT_{NDT} = [\Delta RT_{NDT} \text{ Surface}]e^{-0.067x}$

**Base on surveillance data

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

$$2\sqrt{0 + 17^2} = 34°F$$

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

Plate No.	Initial RT_{NDT}(°F)	ART (°F) (32 EFPY) Margin (°F)	1/4 T*	3/4 T*
B6903-2	0	34	146.0	120.0
B6903-3	10	34	145.1	121.7
B6919-1	15	34	136.0*	111.9*
B6919-2	5	34	159.9	131.8

*Based on surveillance data

Weld Metal Evaluation

Weld No.	Initial RT_{NDT}(°F)	Cu	Ni	CF	32 EFPY ΔRT_{NDT} (°F) Surface*	1/4 T**	3/4 T**
19-894A&B	-56	.25	.21	127.1	142.1	124.5	95.7
11-894	-56	.22	.20	112.0	157.2	137.8	105.8
20-894A&B	-56	.17	.20	92.0	102.8	90.1	69.2

*ΔRT_{NDT} $[CF]f^{(0.28 - 0.10 \log f)}$

**ΔRT_{NDT} $[\Delta RT_{NDT} \text{ surface}]e^{-0.067x}$

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

$$2\sqrt{17^2 + 28^2} = 65.5°F$$

Weld No.	Initial RT_{NDT}(°F)	ART (°F) (32 EFPY) Margin (°F)	1/4 T	3/4 T
19-894A&B	-56	65.5	134.0	105.2
11-894	-56	65.5	147.3	115.3
20-894A&B	-56	65.5	99.6	78.7

Data for 16 EFPY was also generated as above. Based on the above analysis, plate B6919-2 is considered the limiting material and was used to develop heatup and cooldown curves shown in Figures 5 through 12.

7. SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The surveillance capsule withdrawal schedule for Unit 1 (Table IV) should remain the same as identified in the Technical Specifications and WCAP-10474[9]. The dosimetry analysis of the third capsule to be removed after 6 EFPY should be used to re-evaluate the withdrawal schedule for the remaining capsules.

REFERENCES

(1) Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, April 1977.

(2) "Fracture Toughness Requirements," Branch Technical Position - MTEB No. 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

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(4) "Sailor RSIC Data Library Collection DLC-76," Coupled, Self-Shielded, 47 Neutron, 20 Gamma-Ray, P_3, Cross Section Library for Light Water Reactors, Radiation Shielding Information Center, Oak Ridge National Laboratory.

(5) Radcliffe, R. and Holmes, R., "Revised Nuclear Design Data for Cycle 2 of Farley Unit 1," Westinghouse Nuclear Fuels Division, NDS-79-096, April 25, 1979.

(6) Erwin, R., "New Power Distributions for Cycle 7," J. M. Farley Unit 1 Project File, Westinghouse Nuclear Fuels Division, August 19, 1985.

(7) <u>ASME Boiler and Pressure Vessel Code</u>, Section III, Division 1 - Appendices, "Rules for Construction of Nuclear Vessels," Appendix G, "Protection Against Nonductile Failure," pp. 559-564, 1983 Edition, American Society of Mechanical Engineers, New York, 1983.

(8) Miller, J. C., "Response to NRC Comments on Farley Unit 2," ALA-85-706, July 31, 1985.

(9) Boggs, R. S., Yanichko, S. E., Cheney, C. A. and Kaiser, W. T. "Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-10474, February 1984.

(10) Yanichko, S. E., Anderson, S. L., and Kaiser, W. T., "Analysis of Capsule Y from the Alabama Power Company Farley Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-9717, June, 1980.

(11) Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" (Proposed Draft), U.S. Nuclear Regulatory Commission, June 1984.

TABLE I

CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt. %	Nickel, Wt. % 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE II

CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt. %	Nickel, Wt. % 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE III

FARLEY UNIT 1 REACTOR VESSEL TOUGHNESS PROPERTIES

Component	Code No.	Material Type	Cu (%)	P (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shell Energy MWD[c]	Upper Shell Energy NMWD[d]
Closure head dome	B6901	A533,B,Cl.1	0.16	0.009	0.50	-30	-20[a]	140	-
Closure head segment	B6902-1	A533,B,Cl.1	0.17	0.007	0.52	-20	-20[a]	138	-
Closure head flange	B6915-1	A508, Cl.2	0.10	0.012	0.64	60[a]	60[a]	75[a]	-
Vessel flange	B6913-1	A508, Cl.2	0.17	0.011	0.69	60[a]	60[a]	106[a]	-
Inlet nozzle	B6917-1	A508, Cl.2	-	0.010	0.83	60[a]	60[a]	-	110
Inlet nozzle	B6917-2	A508, Cl.2	-	0.008	0.80	60[a]	60[a]	-	80
Inlet nozzle	B6917-3	A508, Cl.2	-	0.008	0.87	60[a]	60[a]	-	98
Outlet nozzle	B6916-1	A508, Cl.2	-	0.007	0.77	60[a]	60[a]	-	96.5
Outlet nozzle	B6916-2	A508, Cl.2	-	0.011	0.78	60[a]	60[a]	-	97.5
Outlet nozzle	B6916-3	A508, Cl.2	-	0.009	0.78	60[a]	60[a]	-	100
Nozzle shell	B6914-1	A508, Cl.2	-	0.010	0.68	30	30[a]	148	-
Inter. shell	B6903-2	A533,B,Cl.1	0.13	0.011	0.60	0	0	151.5	97
Inter. shell	B6903-3	A533,B,Cl.1	0.12	0.014	0.56	10	10	134.5	100
Lower shell	B6919-1	A533,B,Cl.1	0.14	0.015	0.55	-20	15	133	90.5
Lower shell	B6919-2	A533,B,Cl.1	0.14	0.015	0.56	-10	5	134	97
Bottom head ring	B6912-1	A508, Cl.2	-	0.010	0.72	10	10[a]	163.5	-
Bottom head segment	B6906-1	A533,B,Cl.1	0.15	0.011	0.52	-30	-30[a]	147	-
Bottom head dome	B6907-1	A533,B,Cl.1	0.17	0.014	0.60	-30	-30[a]	143.5	-
Inter. shell long. weld seam	M1.33	Sub Arc Weld	0.25	0.017	0.21	0[a]	0[a]	-	-
Inter. to lower shell weld seams	G1.18	Sub Arc Weld	0.22	0.011	<0.20[b]	0[a]	0[a]	-	-
Lower shell long. weld seams	G1.08	Sub Arc Weld	0.17	0.022	<0.20[b]	0[a]	0[a]	-	-

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[a] Estimate per NUREG-0800 "USNRC Standard Review Plan" Branch Technical Position MTEB 5-2.
[b] Estimated (low nickel weld wire used in fabricating vessel weld seams).
[c] Major working direction.
[d] Normal to major working direction

TABLE IV

SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule	Lead Factor	Removal Time[a]	Estimated Fluence $n/cm^2 \times 10^{19}$
Y	3.12	Removed (1.13)	.583 (Actual)
U	3.12	Removed (3.02)	1.65 (Actual)
X	3.12	6	3.05[b]
W	2.70	12	5.28[c]
V	2.70	21	9.28
Z	2.70	Standby	-

[a] Effective full power years from plant startup

[b] Approximates vessel end of life 1/4 thickness wall location fluence

[c] Approximates vessel end of life inner wall location fluence

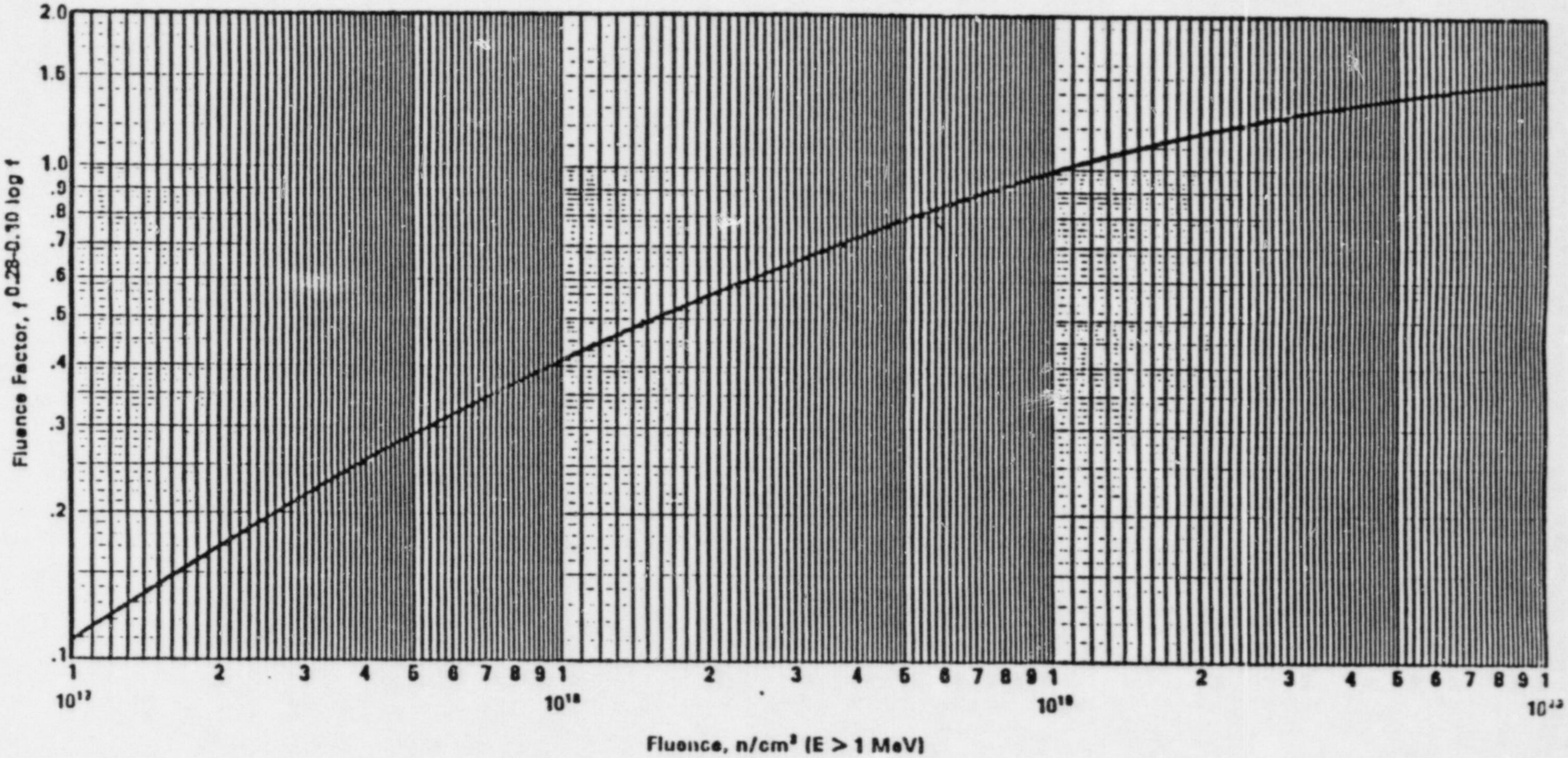

FIGURE 1 FLUENCE FACTOR FOR USE IN THE EXPRESSION FOR ΔRT_{NDT}

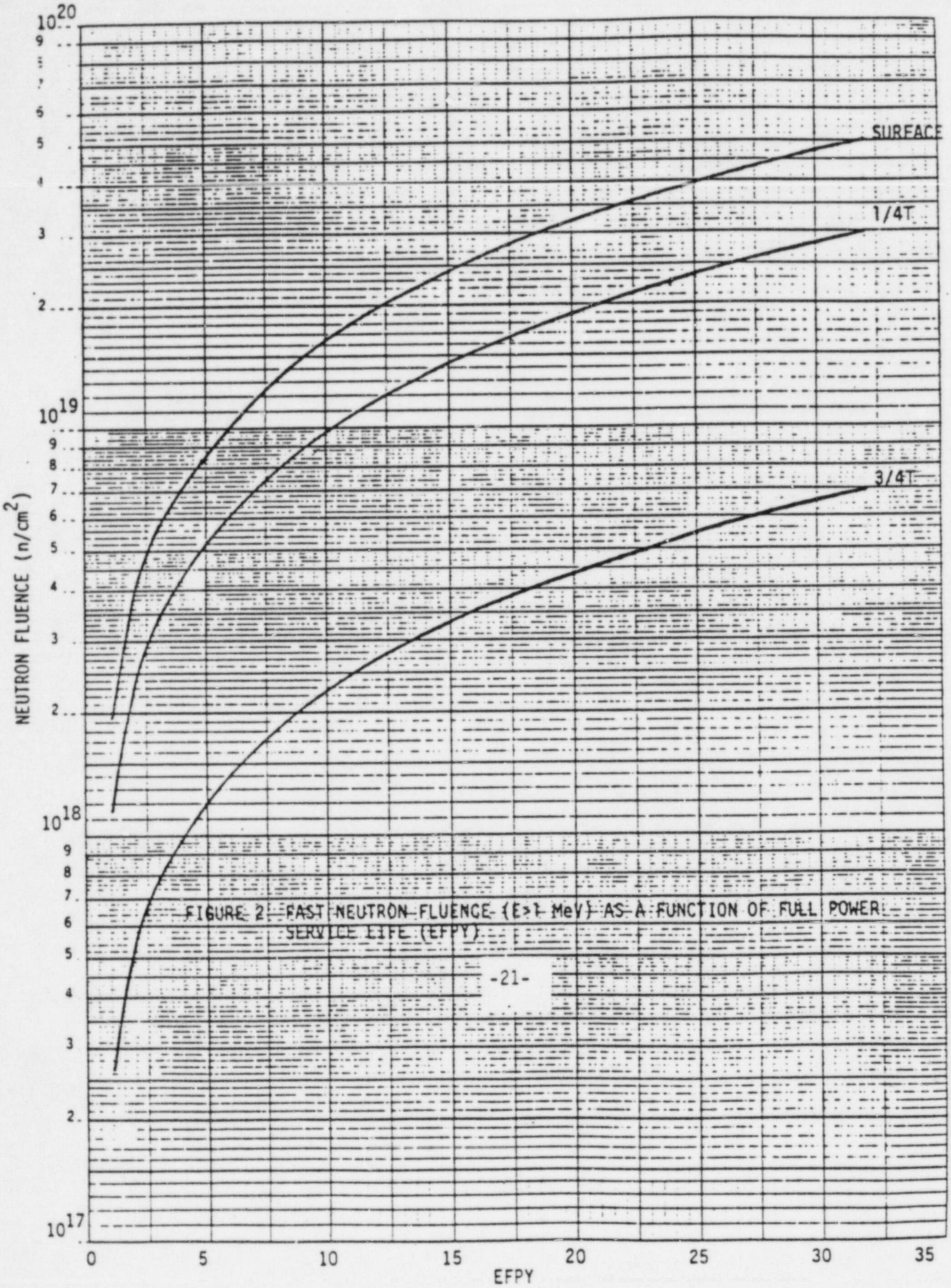

FIGURE 2 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

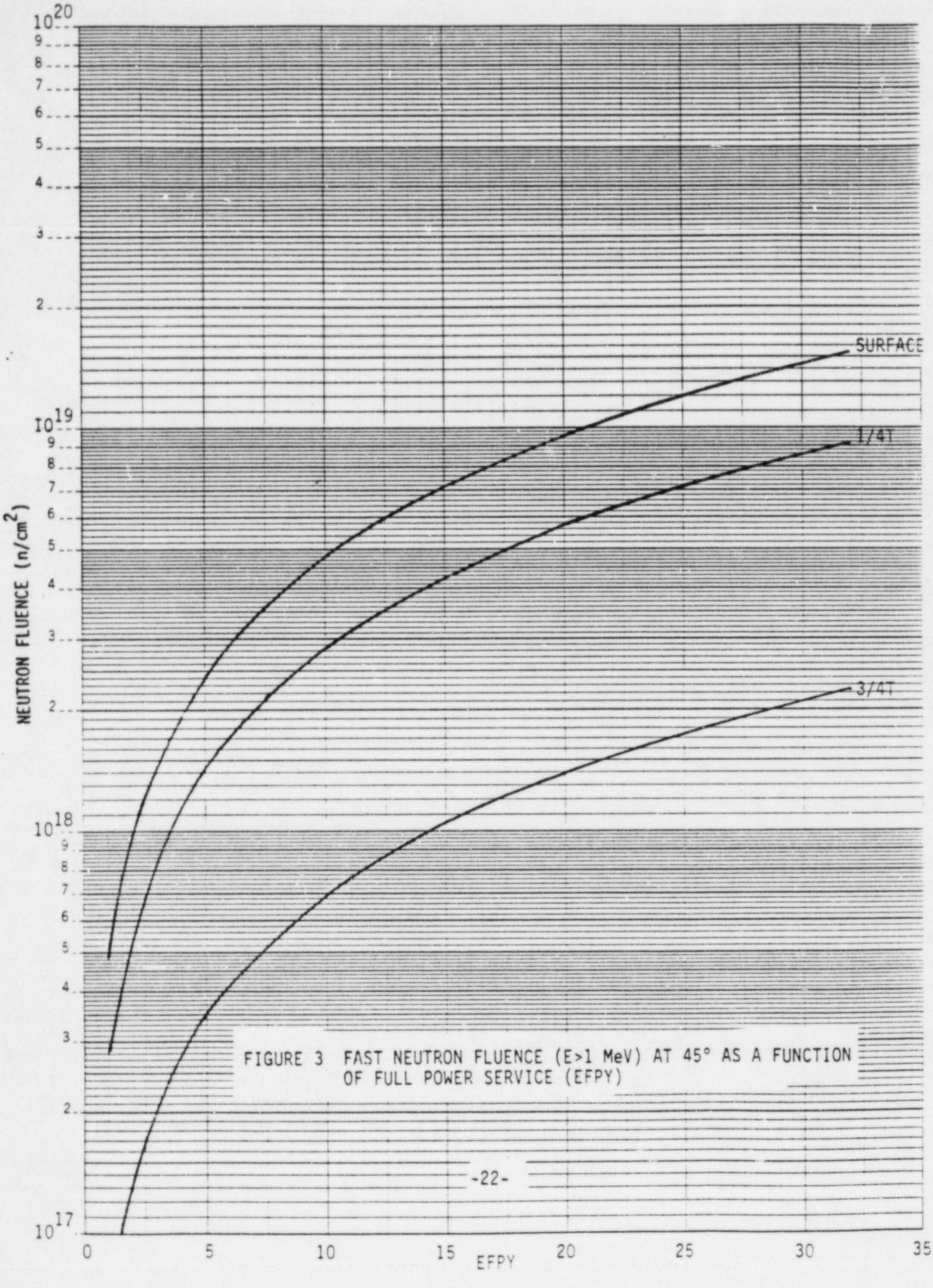

FIGURE 3 FAST NEUTRON FLUENCE (E>1 MeV) AT 45° AS A FUNCTION OF FULL POWER SERVICE (EFPY)

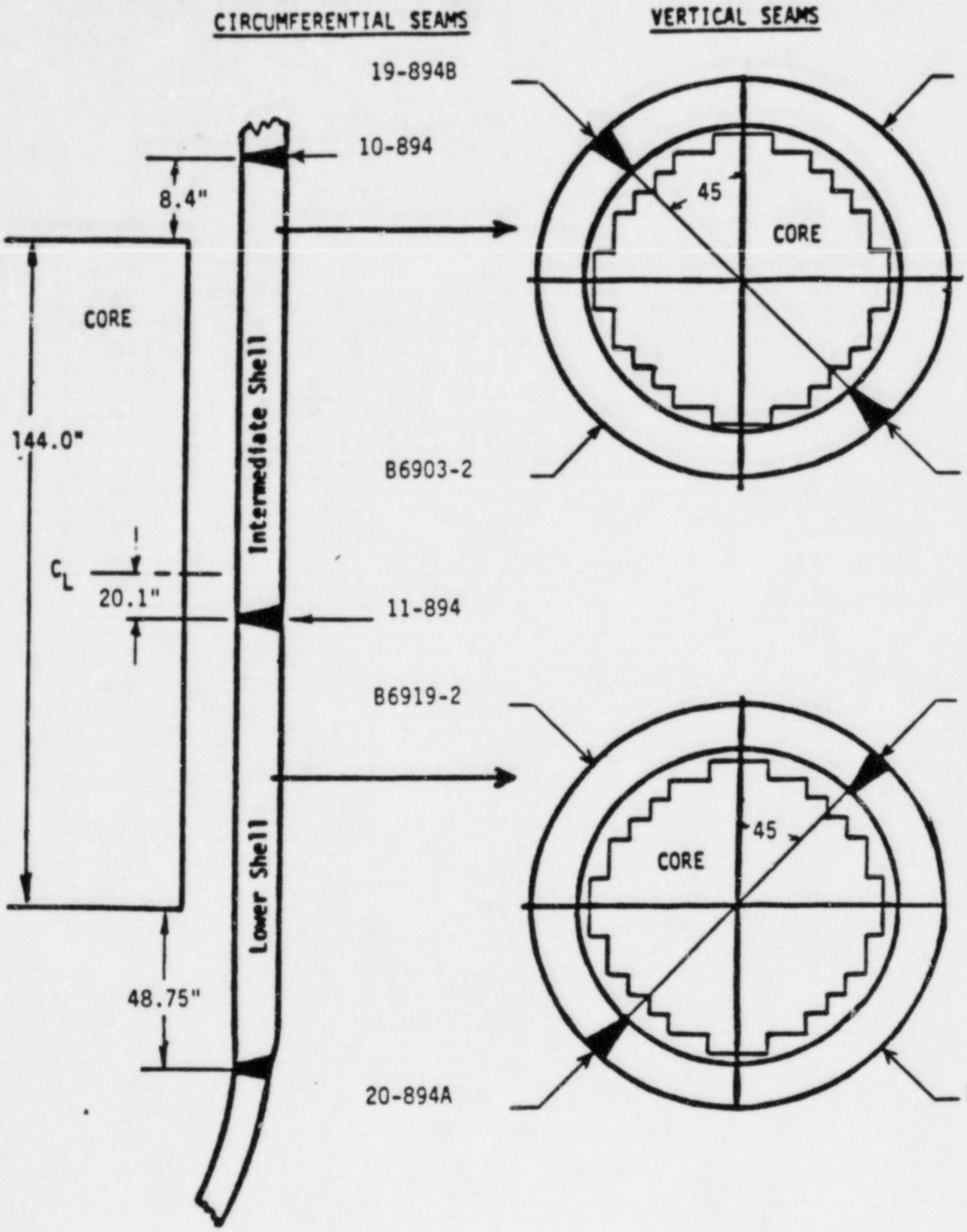

FIGURE 4 IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE J.M. FARLEY UNIT 1 REACTOR VESSEL

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	5°F
RT_{NDT} After 16 EFPY	:	1/4T, 146.4°F
	:	3/4T, 121.5°F

Curve applicable for heatup rates up to 60°F/hr for the service period up to 16 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

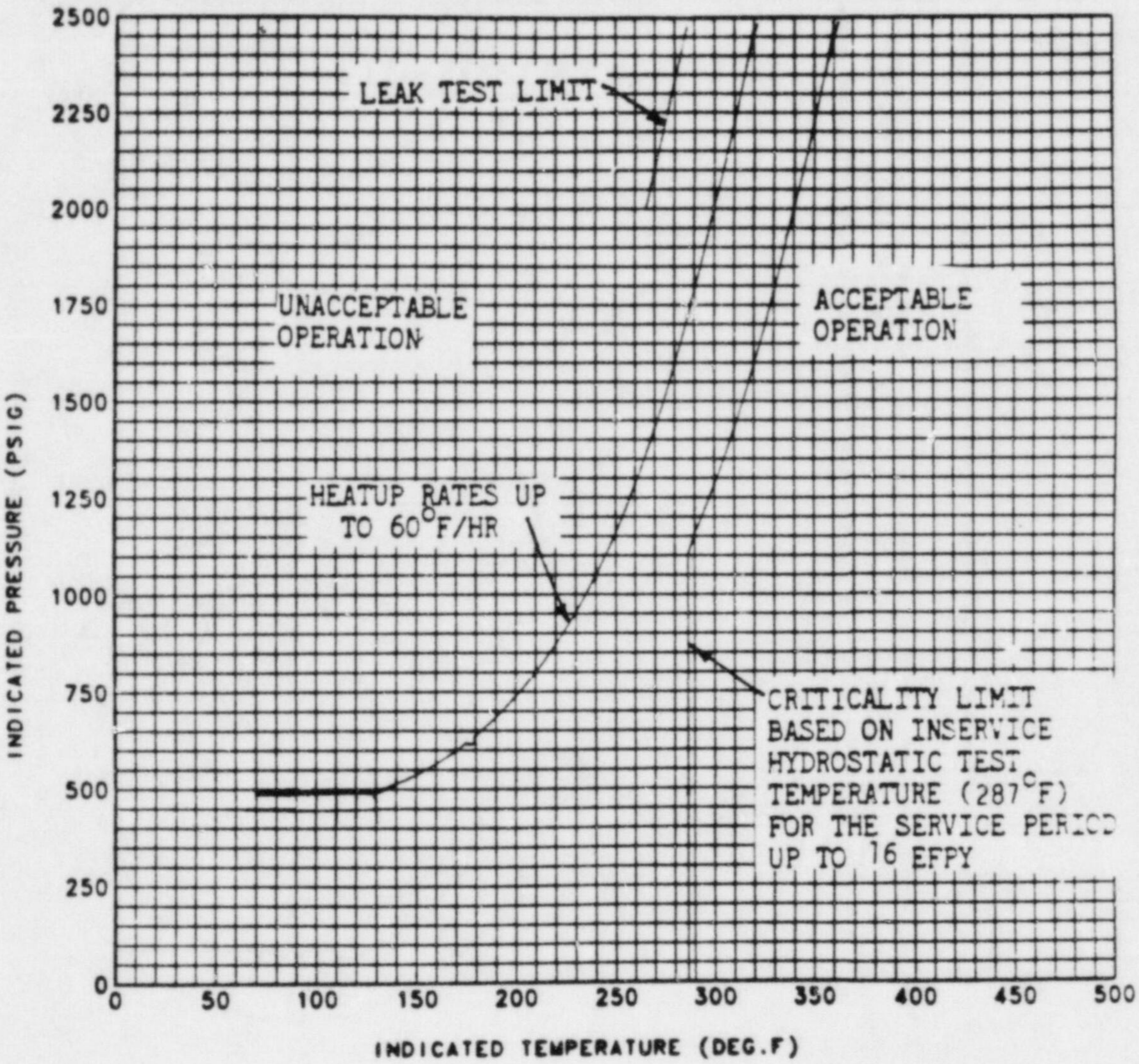

Figure 5 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	5°F
RT_{NDT} After 16 EFPY	:	1/4T, 146.4°F
	:	3/4T, 121.5°F

Curves applicable for cooldown rates up to 100°F/hr for the service period up to 16 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

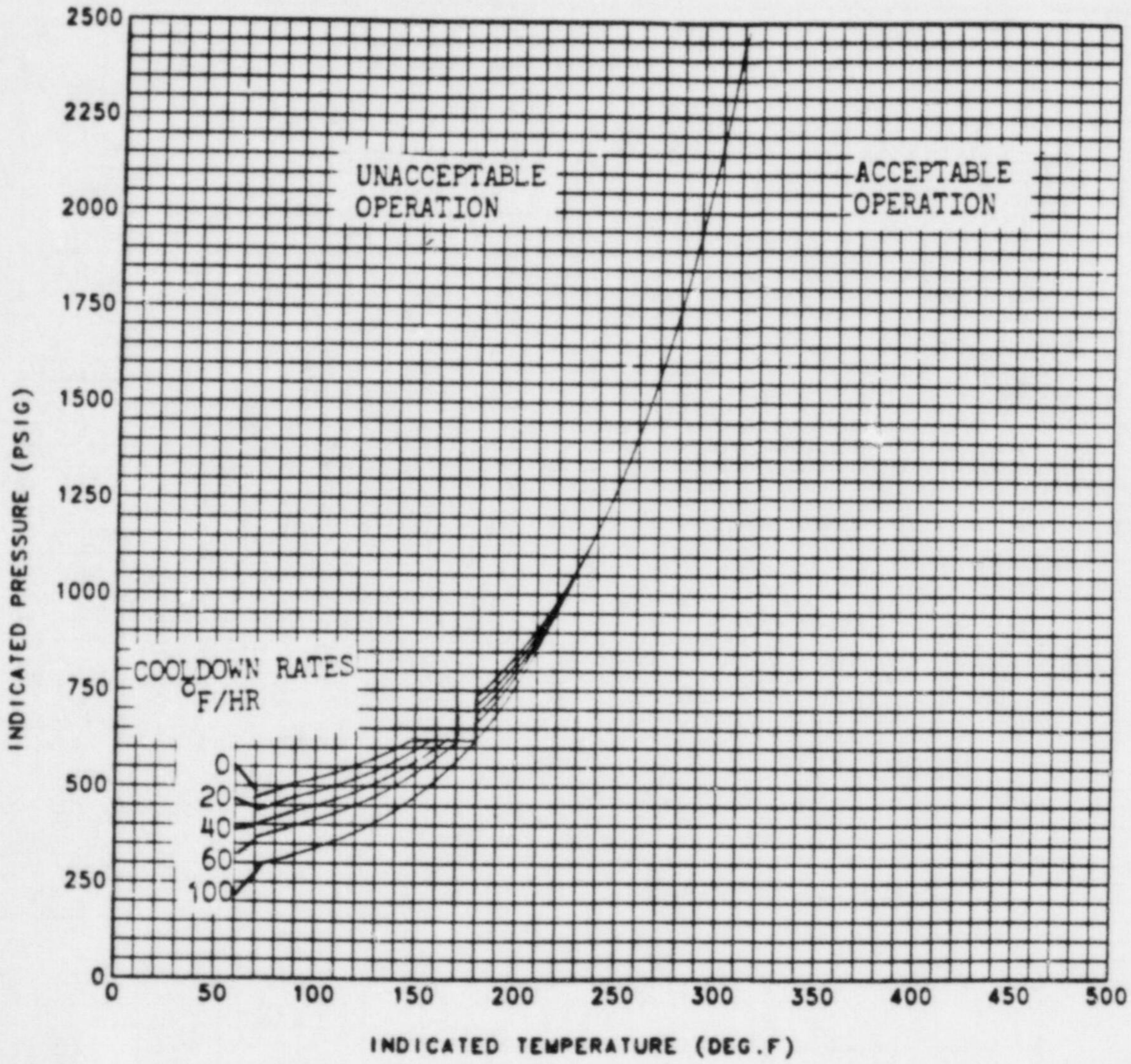

Figure 6 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	$5^{\circ}F$
RT_{NDT} After 32 EFPY	:	1/4T, $160^{\circ}F$
	:	3/4T, $132^{\circ}F$

Curve applicable for heatup rates up to $60^{\circ}F/hr$ for the service period up to 32 EFPY and contains margins of $10^{\circ}F$ and 60 psig for possible instrument errors

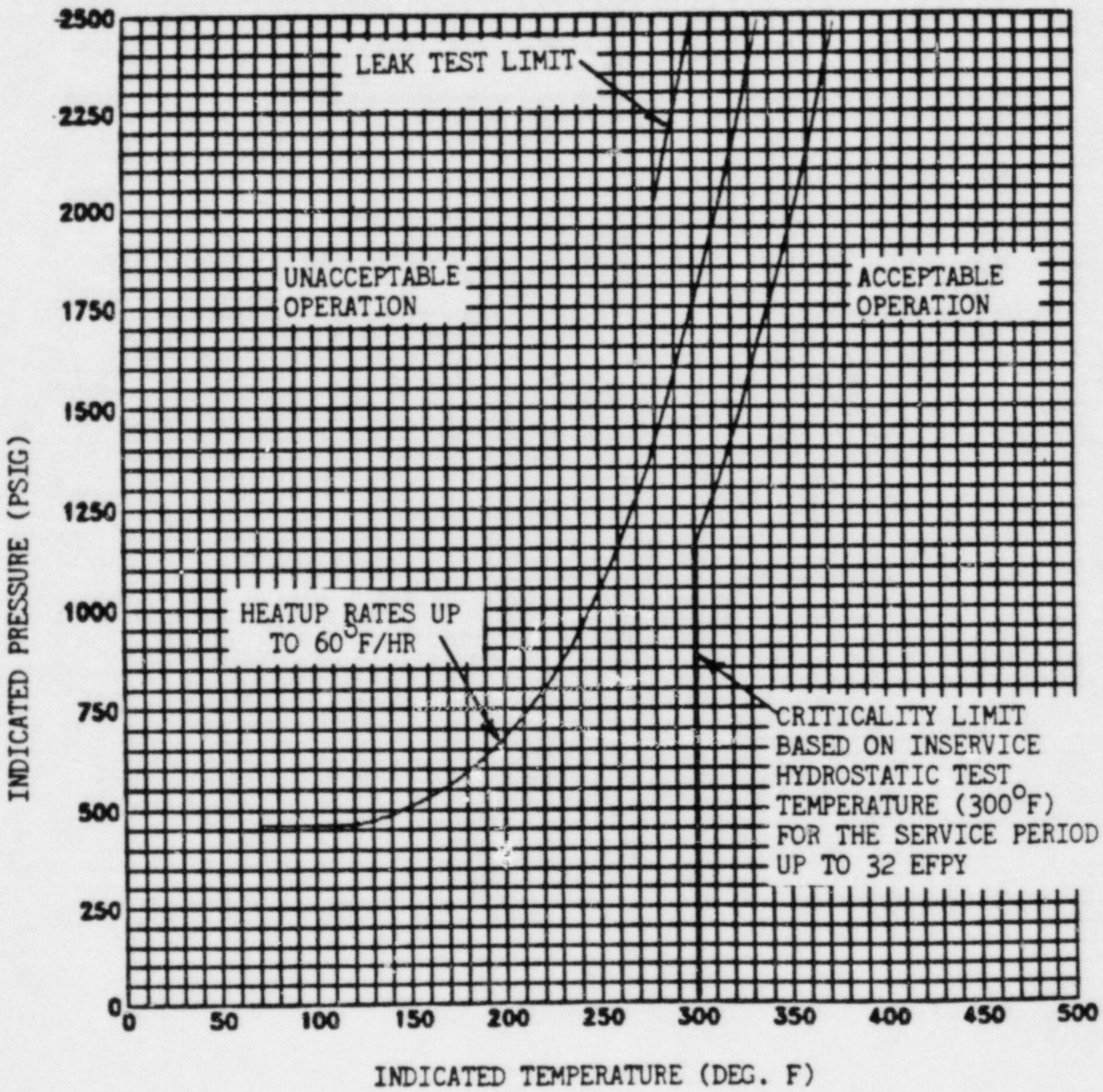

Figure 7 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable for up to 32 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	$5^{\circ}F$
RT_{NDT} After 32 EFPY	:	1/4T, $160^{\circ}F$
	:	3/4T, $132^{\circ}F$

Curves applicable for cooldown rates up to $100^{\circ}F$/hr for the service period up to 32 EFPY and contains margins of $10^{\circ}F$ and 60 psig for possible instrument errors

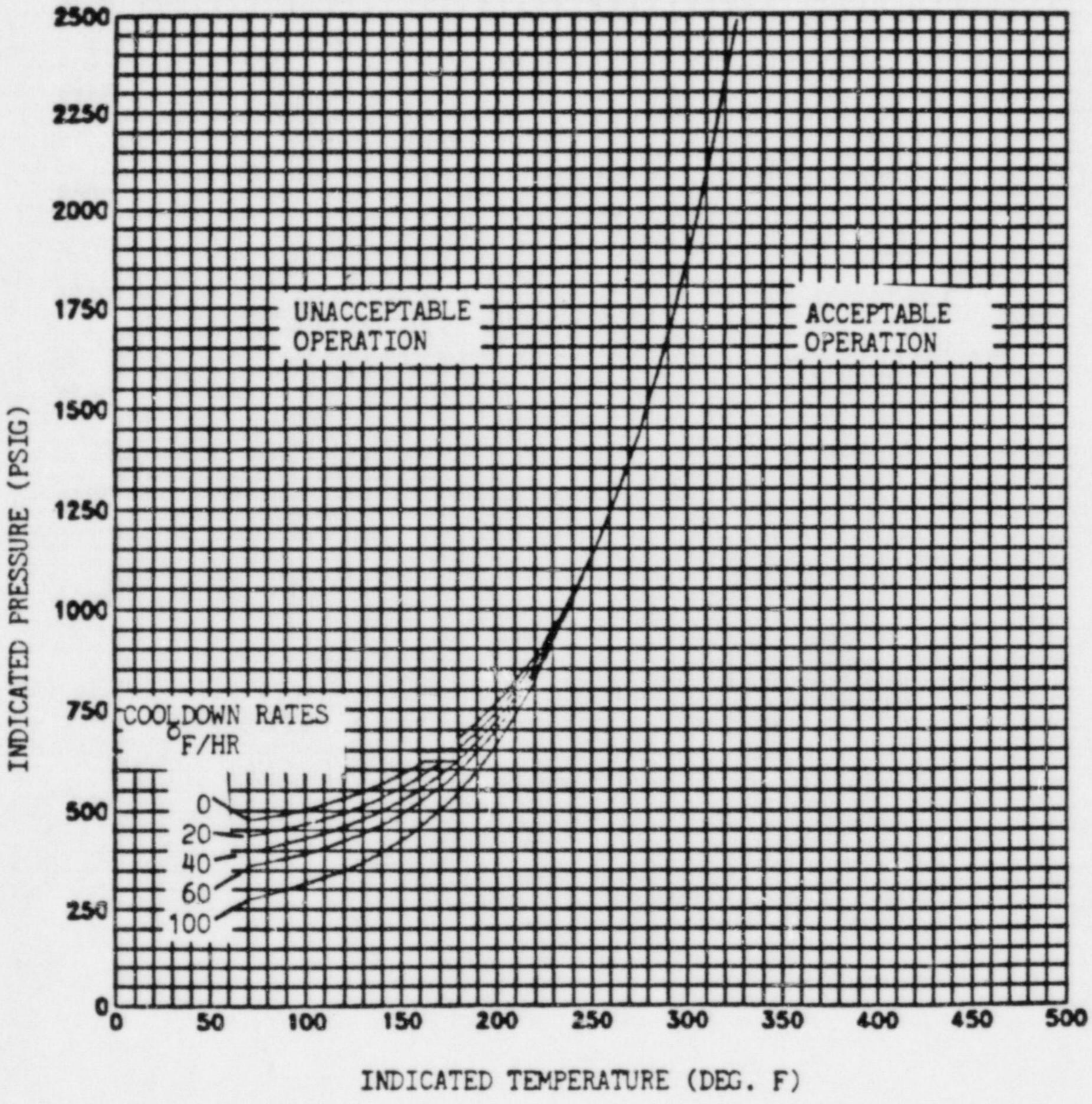

Figure 8 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for up to 32 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	5°F
RT_{NDT} After 16 EFPY	:	1/4T, 146.4°F
	:	3/4T, 121.5°F

Curve applicable for heatup rates up to 60°F/hr for the service period up to 16 EFPY

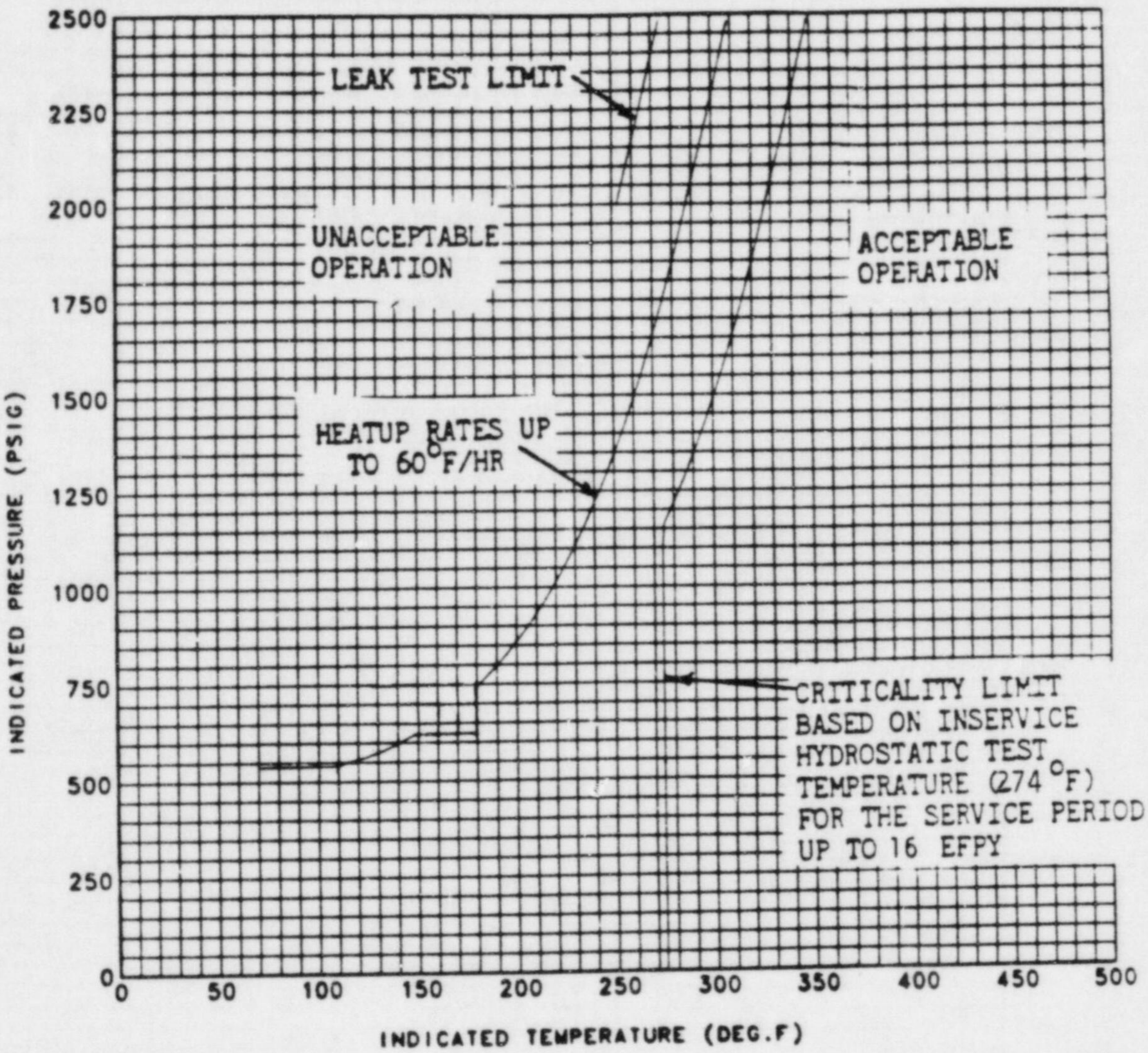

Figure 9 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	5°F
RT_{NDT} After 16 EFPY	:	1/4T, 146.4°F
	:	3/4T, 121.5°F

Curves applicable for cooldown rates up to 100°F/hr for the service period up to 16 EFPY

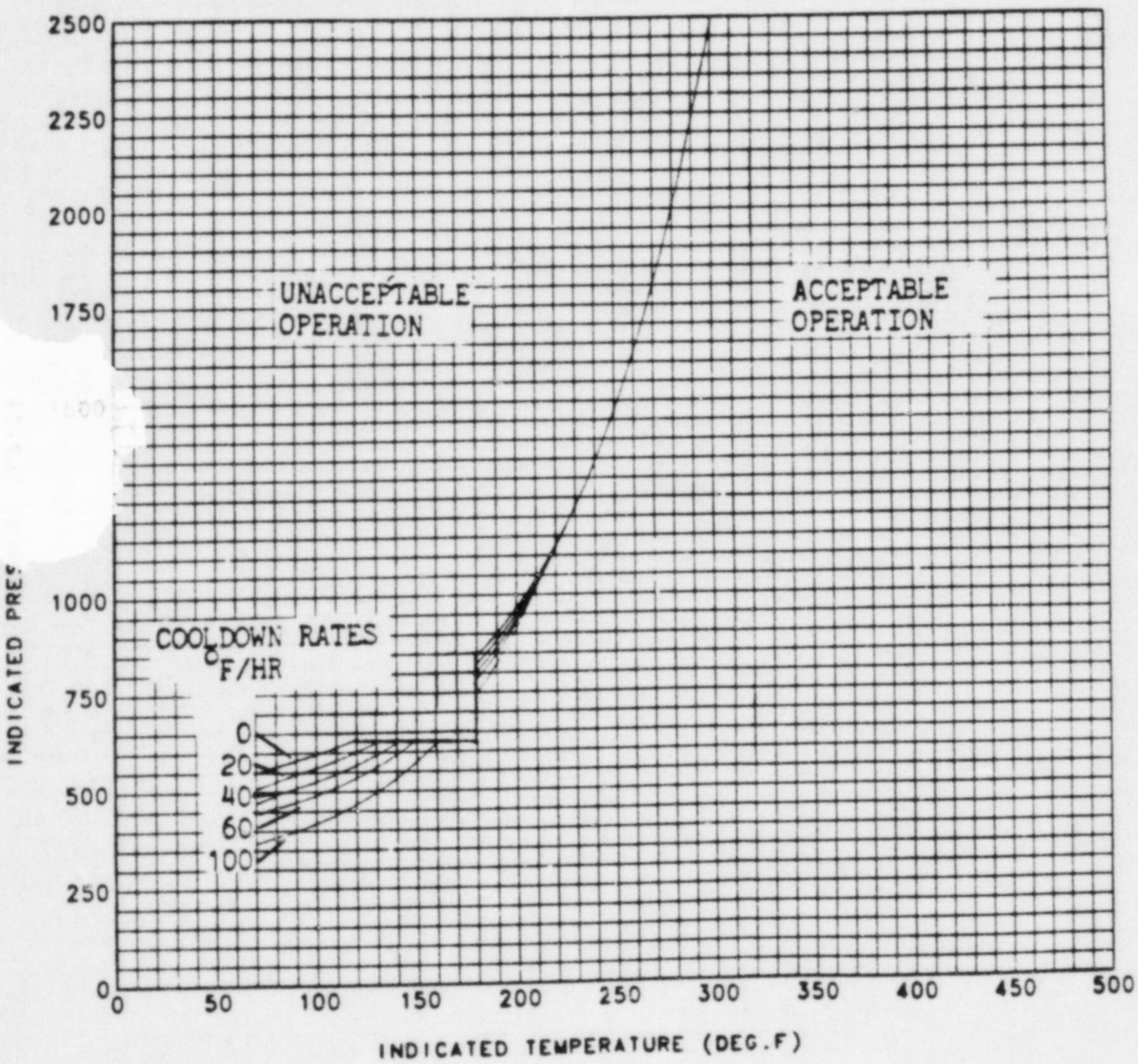

Figure 10 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	$5^{o}F$
RT_{NDT} After 32 EFPY	:	1/4T, $160^{o}F$
	:	3/4T, $132^{o}F$

Curve applicable for heatup rates up to $60^{o}F/hr$ for the service period up to 32 EFPY

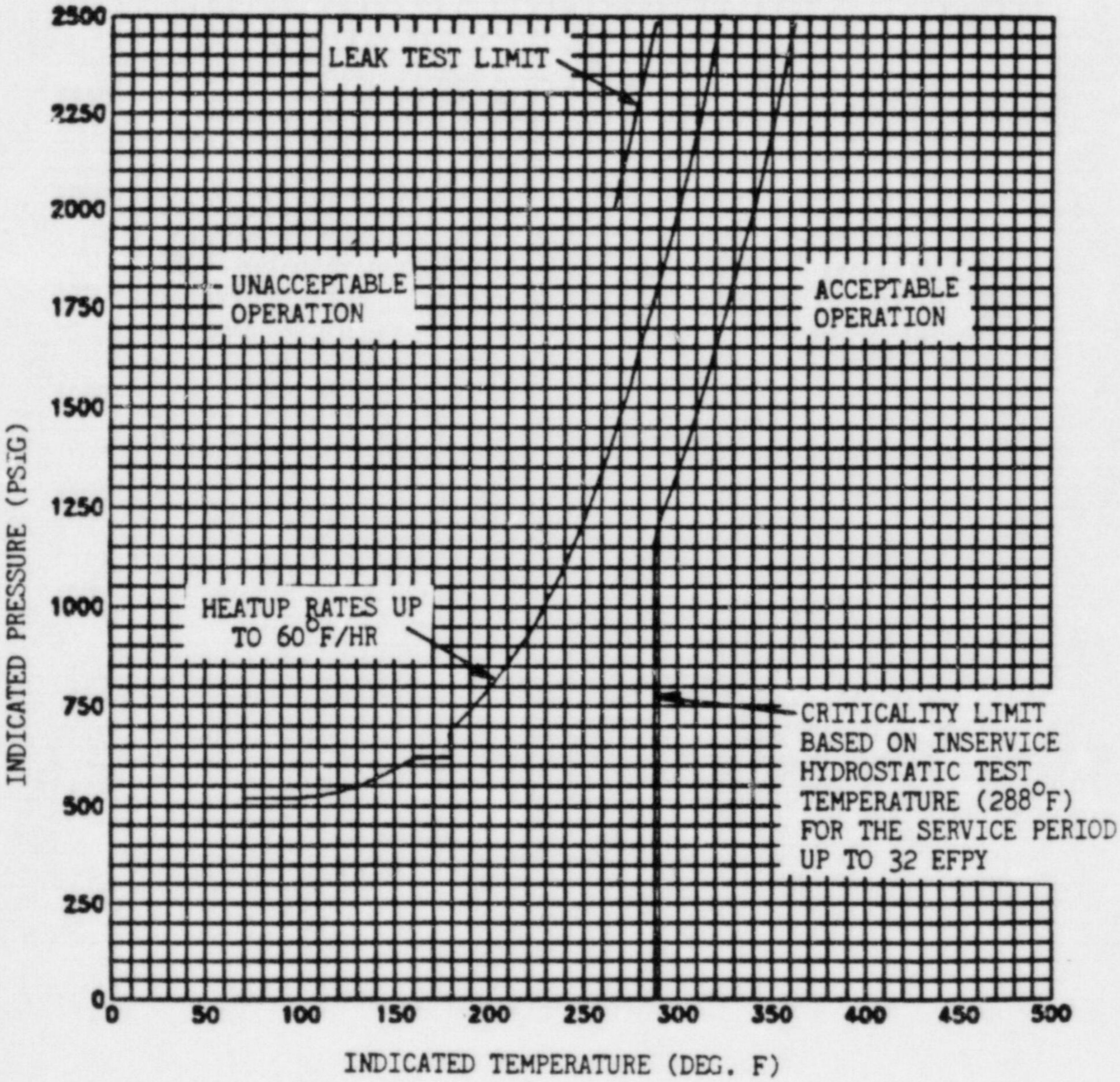

Figure 11 Farley Unit 1 Reactor Coolant System Heatup Limitations Applicable for up to 32 EFPY

MATERIAL PROPERTY BASIS :

Controlling Material	:	Lower Shell (plate no. B6919-2)
Copper Content	:	0.14 WT%
Nickel Content	:	0.56 WT%
Initial RT_{NDT}	:	$5^{\circ}F$
RT_{NDT} After 32 EFPY	:	1/4T, $160^{\circ}F$
	:	3/4T, $132^{\circ}F$

Curves applicable for cooldown rates up to $100^{\circ}F/hr$ for the service period up to 32 EFPY

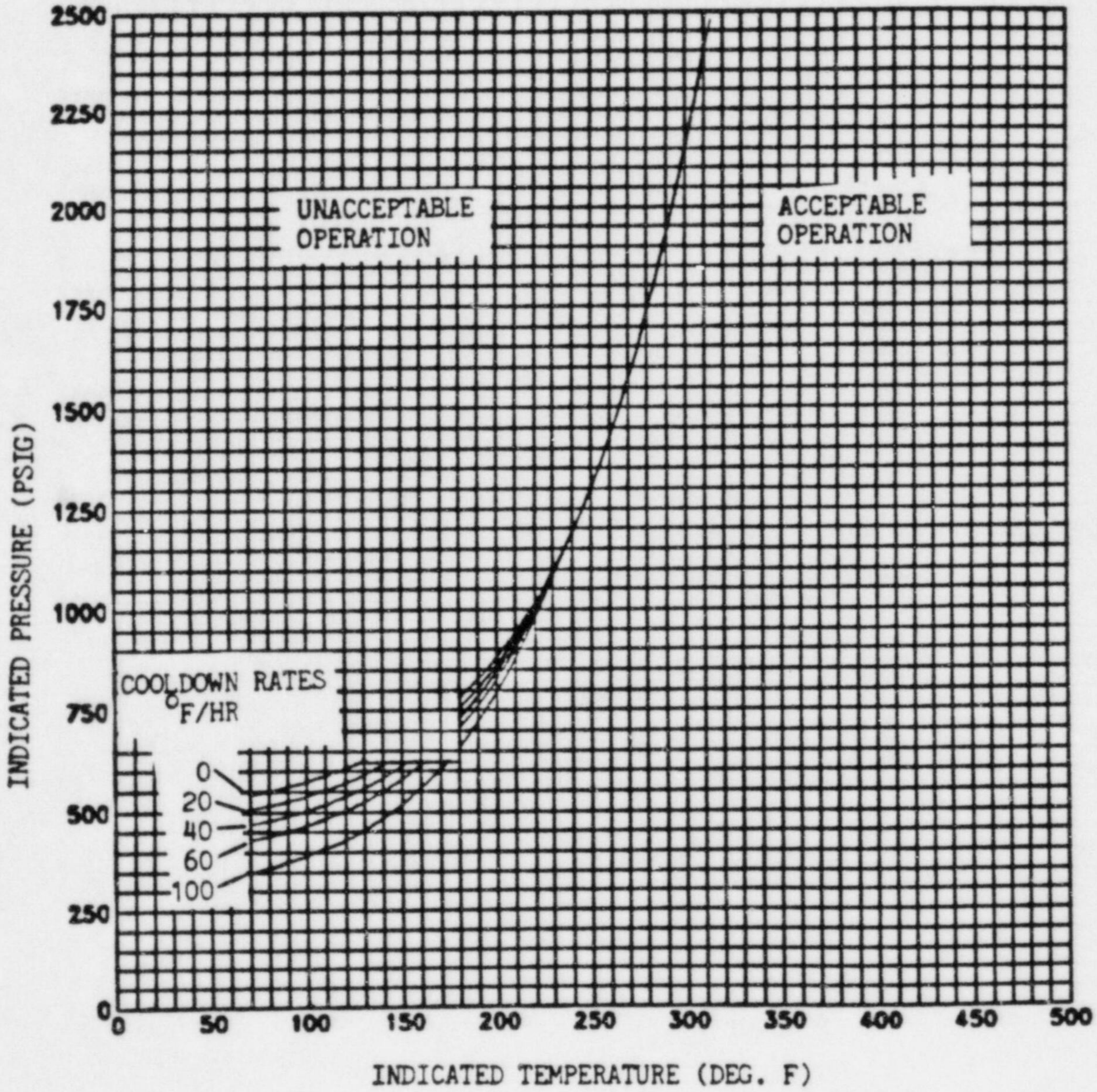

Figure 12 Farley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for up to 32 EFPY