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HEATUP AND COOLDOWN LIMIT CURVES FOR THE ALABAMA POWER COMPANY JOSEPH M. FARLEY UNIT 2 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 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.

Given the copper and phosphorus contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Figure 1. Fast-neutron fluence (E > 1 MeV) at the inner surface, 1/4T (wall thickness) and 3/4T (wall thickness) vessel locations are given as a function of full-power service in Figure 2. 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 2 reactor vessel materials are presented in Table I. 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 2 Vessel Material Surveillance Program.

3. FLUENCE CALCULATIONS

For the purpose of revising heatup and cooldown curves for Farley Unit 2, which has limiting embrittlement characteristics in the intermediate shell course plate B7212-1, it is necessary to know vessel fast fluence (ϕ (E > 1 MeV)) at the azimuthal peak location. This peak location is at 0° relative to the core cardinal axes, and at this angle, fast fluences are required at vessel inner radius, vessel 1/4T, and vessel 3/4T. 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 $I(R,\theta)$ 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 2 were taken from the following Westinghouse fuel cycle design reports for each operating cycle to date:

Fuel Cycle	Report
1	WCAP-9710
2	WCAP-10187
3	WCAP-10410
4	WCAP-10674

Of these, Cycles 1 through 2 utilized out-in fuel loading patterns, and Cycles 3 an. 4 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-10425, "Analysis of Capsule U from the Alabama Power Company, Joseph M. Farley Unit 2 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
1.09	2.027×10^{18}	1.200×10^{18}	2.789×10^{17}
1.86	3.625×10^{18}	2.147×10^{18}	4.989×10^{17}
2.95	5.527×10^{18}	3.273×10^{18}	7.606×10^{17}
3.20	5.898×10^{18}	3.493×10^{18}	8.116×10^{17}
32.0	4.999×10^{19}	2.960×10^{19}	6.878×10^{18}

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(5). 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 reference nil-ductility temperature RT_{NDT}. Thus, the governing equation of the heatup-cooldown analysis is defined in Appendix G to the ASME Code(5) as follows:

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

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 3 and 4. In addition, heatup and cooldown curves without instrument errors are presented in Figures 5 and 6.

Based on the Farley Unit 2 fracture analysis results from Reference 6, the heatup curves in Figures 3 and 5 and the cooldown curves in Figure 6 are impacted by the new 10CFR50 rule. This rule states that the minimum metal temperature of the closure flange regions should 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 Westinghouse plants). However, the cooldown curve in Figure 4 is not impacted by the 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[5], 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 Mitigatio System (OMS) presently in Farley Unit 2 (Technical Specification 3/4.4.10.3) with respect to the 8 EFPY Heatup and Cooldown curves shown in Figures 5 and 6 respectively. For the purpose of the evaluation it was assumed that the residual heat removal (RHR) relief valve lifts at 495 psig which includes 10% accumulation. The Heatup curve in Figure 5 does not fall below 495 psig at any temperature. A comparison to cooldown curves in Figure 6 shows that in the low temperature range (<130°F) cooldown rates of 20°F/Hr and lower fall 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 of equal to 40°F/Hr are highly unlikely at low temperature conditions. Therefore, the Appendix G curves as illustrated in Figures 5 and 6 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 3 and 5. 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 3 and 5 represent minimum temperature requirements at the leak test pressure specified by applicable codes[2,5].

6. AVAILABLE SURVEILLANCE CAPSULE DATA

Charpy test specimens from Capsule U irradiated to 5.61×10^{18} n/cm^2 indicate that the representative core region weld metal and limiting core region shell plate B7212-1 exhibited maximum shifts in RT_{NDT} of 10°F and 133°F, respectively(7). The shell plate shift of 133°F is less than the 157°F shift which is predicted by Regulatory Guide 1.99 Revision 1.

The ΔRT_{NDT}'s used to compute the heatup and cooldown curves were obtained from the radiation damage curve associated with the surveillance shell plate shift of 133°F at 5.61×10^{18} n/cm^2 shown in Figure 1.

7. SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The surveillance capsule withdrawal schedule for Unit 2 (Table II) should remain the same as identified in the Technical Specifications and WCAP-10425(7). The dosimetry analysis of the second capsule to be removed after 4 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.

(3) Soltesz, R. G., Disney, R. K., Jedruch, J. and Ziegler, S. L., "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation Vol. 5 - Two Dimensional, Discrete Ordinates Transport Technique," WANL-PR(LL)034, Vol. 5, August 1970.

(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 Shield Information Center, Oak Ridge National Laboratory.

(5) ASME Boiler and Pressure Vessel Code, 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.

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

(7) Kunka, M. K., Yanichko, S. E., Cheney, C. A. and Kaiser, W. T. "Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-10425, October 1983.

TABLE I

FARLEY UNIT 2 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.

TABLE II

SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following removal schedule is recommended for future capsules to be removed from the Farley Unit 2 reactor vessel:

Capsule	Lead Factor	Removal Time[a]	Estimated Fluence $n/cm^2 \times 10^{19}$
U	3.12	Removed (1.1)	.56 (Actual)
W	2.70	4	2.18
X	3.12	6	3.78[b]
Z	2.70	12	6.54[c]
V	3.12	18	11.34
Y	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

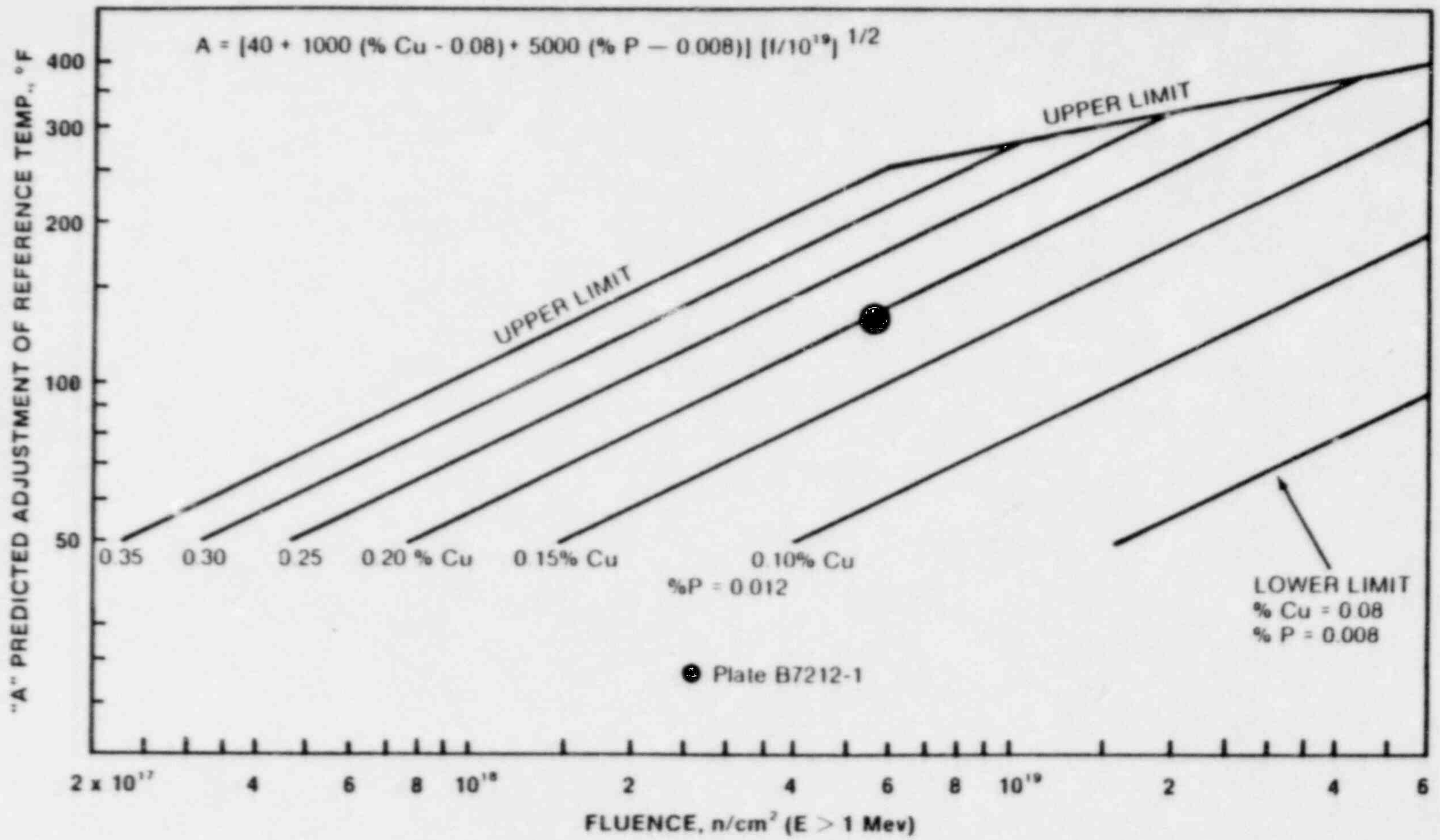

Figure 1. Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents other than those Plotted, use the Expression for "A" given on the Figure.

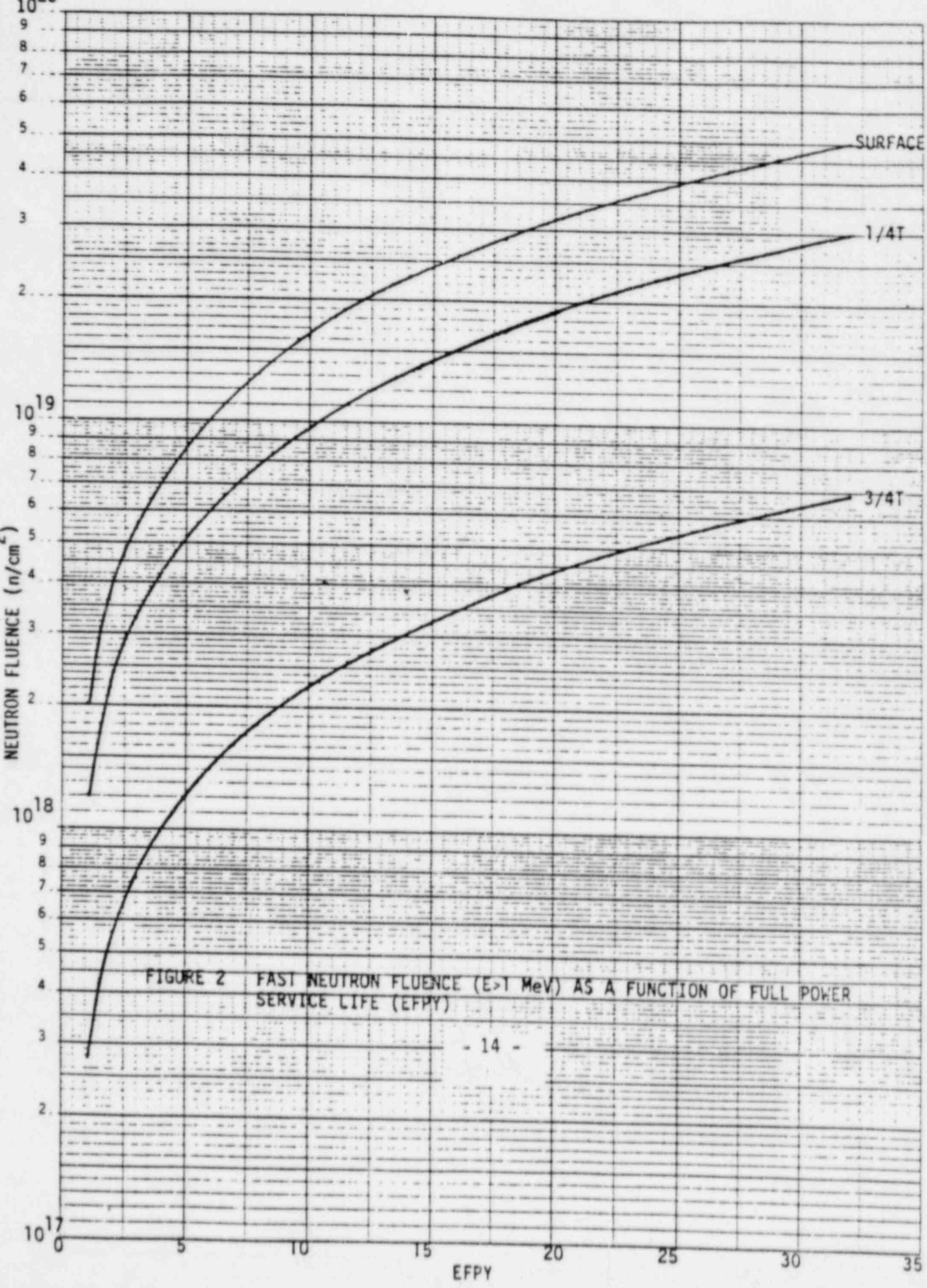

FIGURE 2 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

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 HEATUP RATES UP TO $60^{\circ}F/HR$ FOR THE SERVICE PERIOD UP TO 8 EFPY AND CONTAINS MARGINS OF $10^{\circ}F$ AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

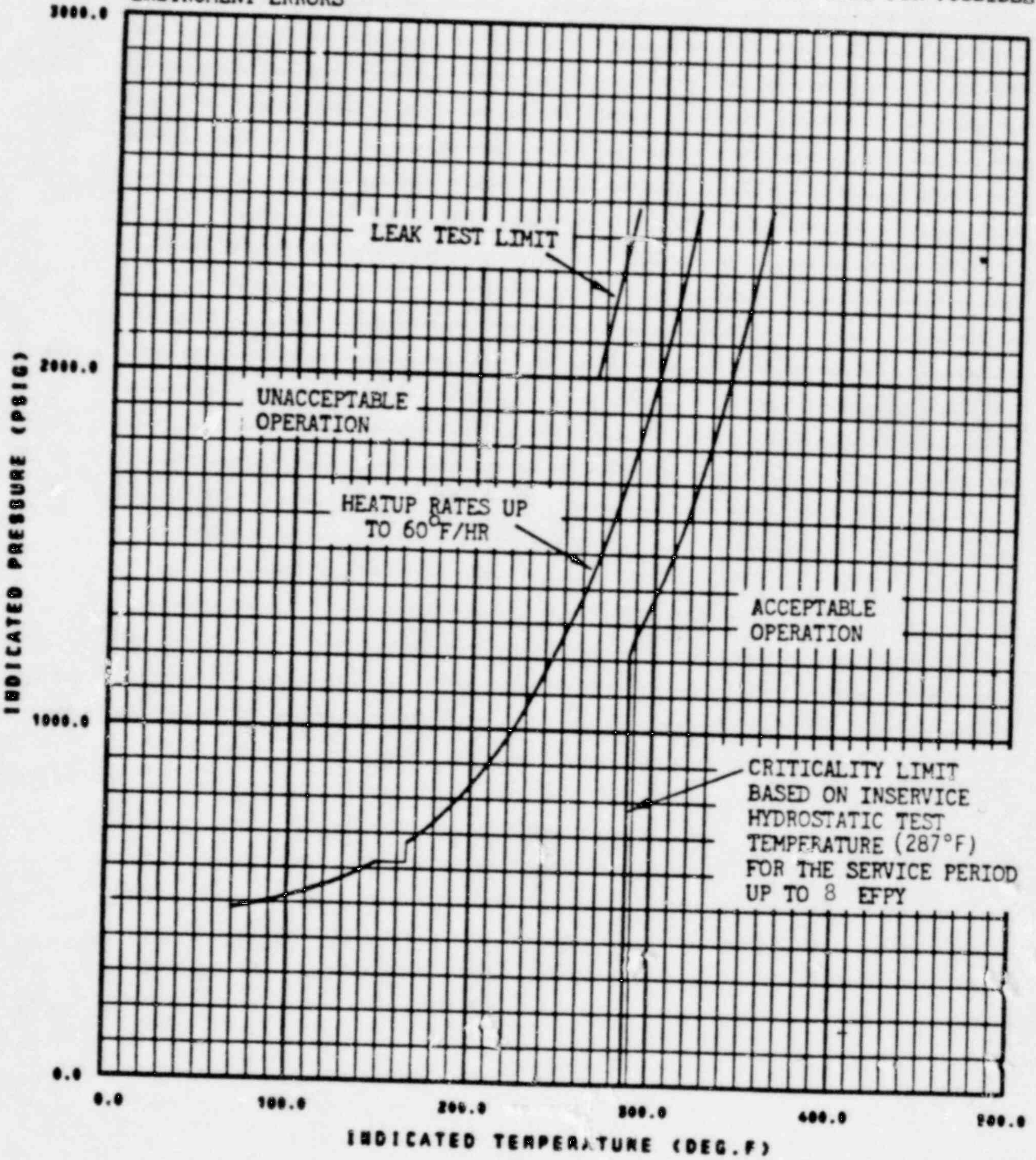

FIGURE 3 FARLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

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 AND CONTAINS MARGINS OF $10^{\circ}F$ AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

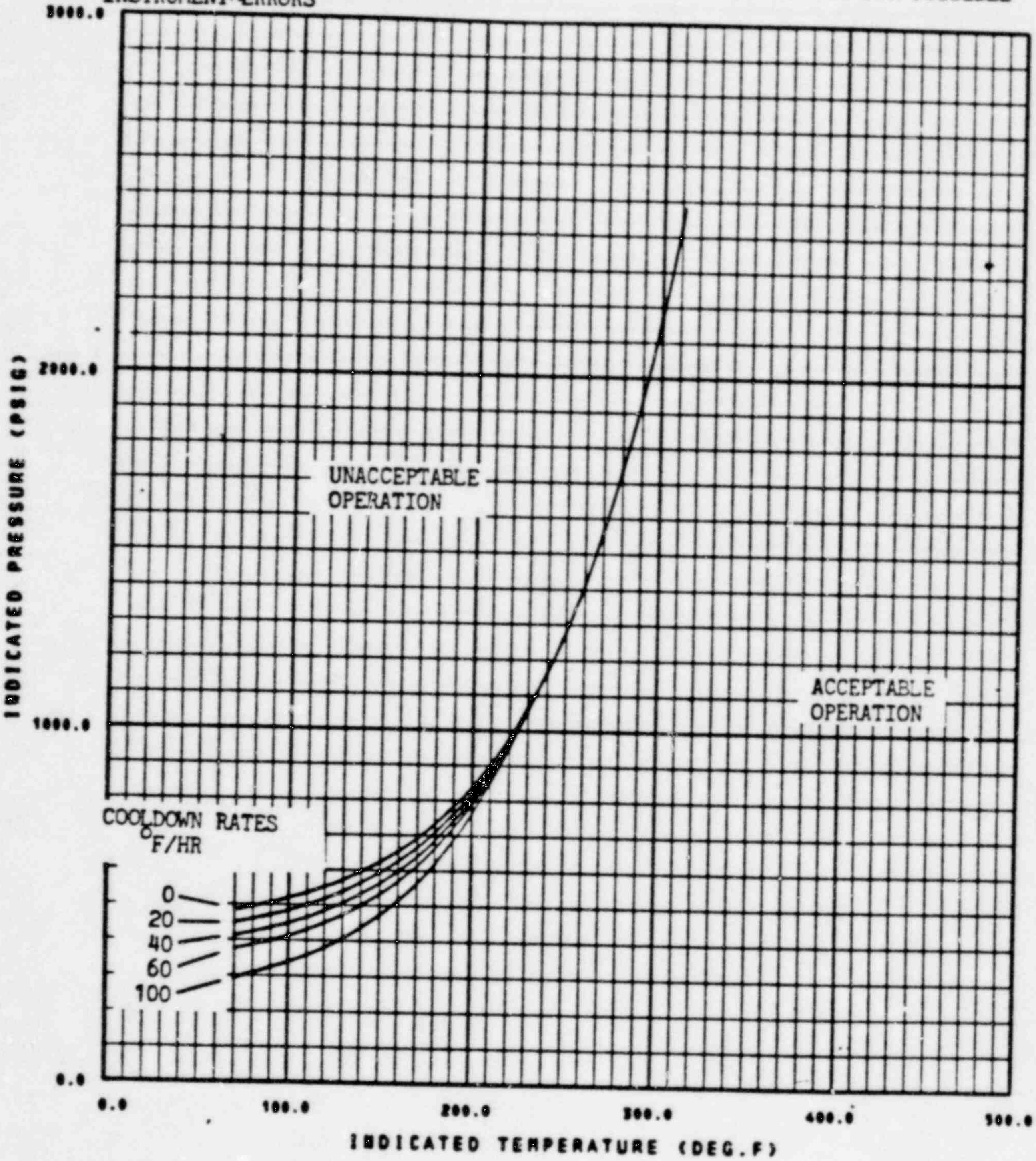

FIGURE 4 FARLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

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

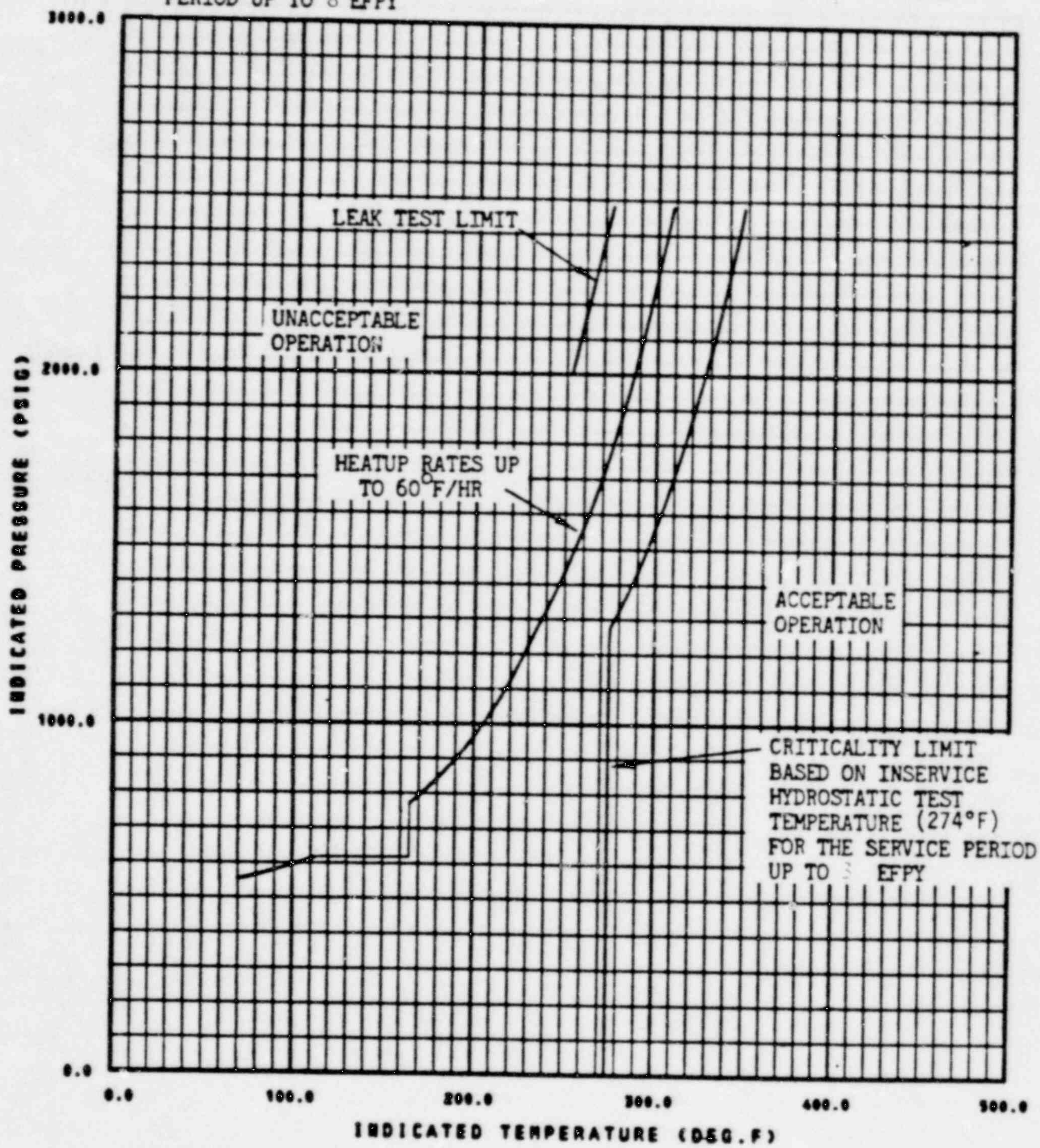

FIGURE 5 FARLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

MATERIAL PROPERTY BASIS

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

CURVES APPLICABLE FOR COOLDOWN RATES UP TO $100^{o}F/HR$ FOR THE SERVICE PERIOD UP TO 3 EFPY

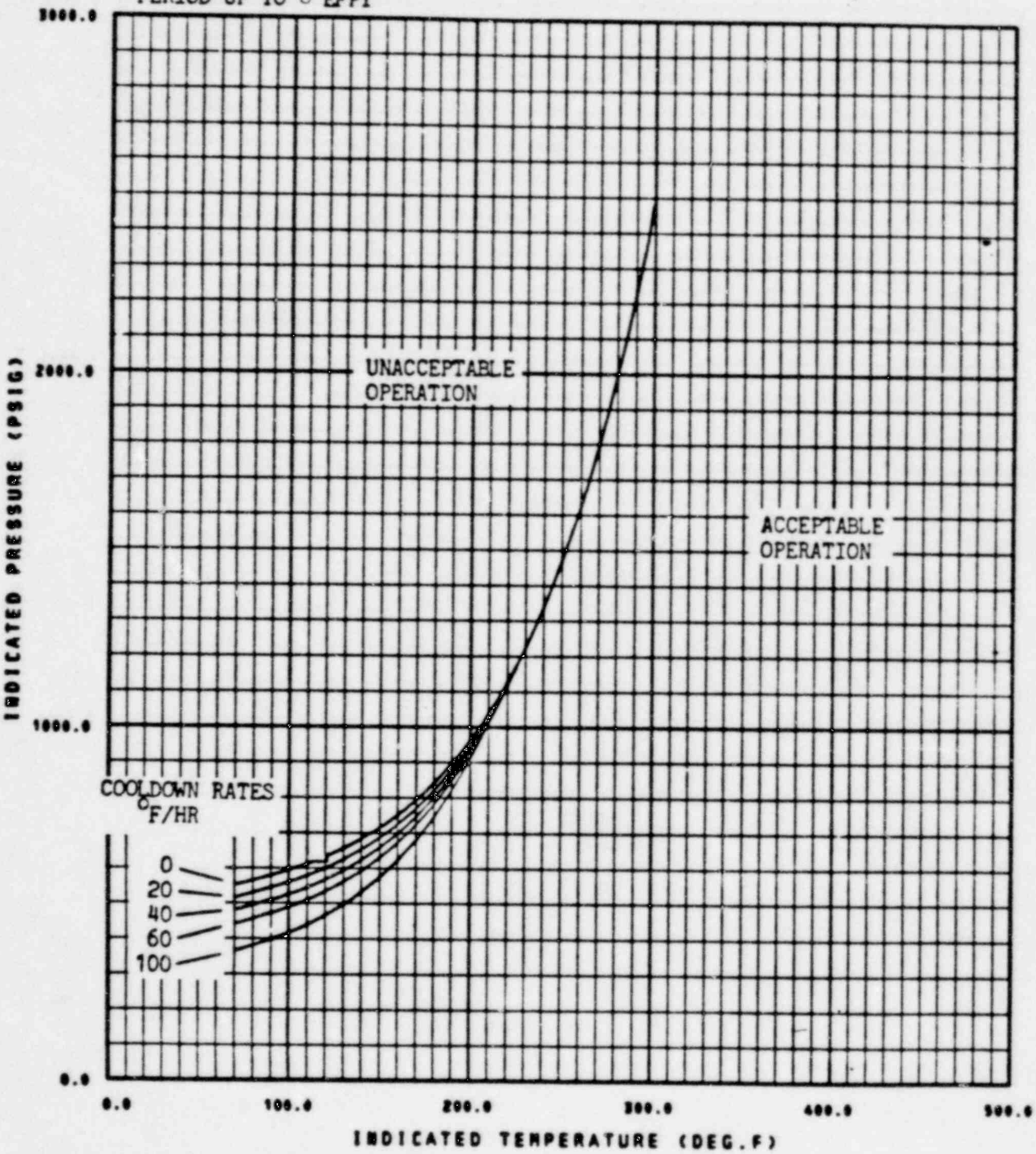

FIGURE 6 FARLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY