Portland General Electric Company

Bart D. Withers Vice President

January 23, 1986

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Director of Nuclear Reactor Regulation
ATTN: Mr. Steven A. Varga
Director, PWR-A
Project Directorate No. 3
U.S. Nuclear Regulatory Commission
Washington DC 20555

Dear Sir:

Trojan Reactor Vessel Radiation Surveillance Program

In accordance with Section III.A of Appendix H to 10 CFR 50, attached is the summary technical report for radiation specimen Capsule X. Capsule X was withdrawn from the Trojan reactor vessel at 4.28 EFPY and was tested in accordance with 10 CFR 50, Appendices G and H and ASTM Specification E185-82. The test results demonstrated the reactor vessel materials are less sensitive to irradiation than predicted by Regulatory Guide 1.99, Revision 1.

Please note Section 7 and Appendix A of the report identify changes to the surveillance capsule removal schedule and the heatup and cooldown limit curves based on identified changes in RT_{NDT}. The removal schedule and heatup and cooldown curves are contained in Trojan Technical Specifications and are currently the subject of License Change Application (LCA) 99, Revision 2 which you are reviewing. Therefore, we would like to withdraw LCA 99, Revision 2 so that we may incorporate the changes identified in the Capsule X report. A new LCA will then be submitted.

Sincerely,

BDWithers

Bart D. Withers
Vice President
Nuclear

Attachment

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy, w/o attachment

Mr. John B. Martin
Regional Administrator, Region V
U.S. Nuclear Regulatory Commission, w/o attach

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APPENDIX A

HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

A-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 (NDTT) 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. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's 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 and phosphorus) present in reactor vessel steels. Design curves which show the effect of fluence and copper and phosphorus contents on ΔRT_{NDT} for reactor vessel steels are shown in Figure A-1.

Given the copper and phosphorus contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Figure A-1. Fast neutron fluence ($E > 1$ MeV) at the vessel inner surface, the ¼ T (wall thickness), and ¾ T (wall thickness) vessel locations are given as a function of full-power service life in Figure A-2. The data for all ferritic materials in the reactor coolant pressure boundary are examined to ensure that no other component will be limiting with respect to RT_{NDT}.

A-2. FRACTURE TOUGHNESS PROPERTIES

The preirradiation fracture-toughness properties of the Trojan reactor vessel materials are presented in Table A-1. 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.[1] The postirradiation fracture-toughness properties of the reactor vessel beltline material were obtained directly from the Trojan Vessel Material Surveillance Program.

A-3. 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 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.[2] The K_{IR} curve is given by the equation:

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

where:

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}.

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[2] as follows:

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

where:

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

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

K_{IR} = 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 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 are computed. From Equation A-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 reference flaw of Appendix G to the ASME Code 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 ¼ T 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 ¼ T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so 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 ¼ T 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 insures 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 ¼ T 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 ¼ T crack during heatup is lower than the K_{IR} for the ¼ T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so 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 ¼ T 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 ¼ T 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 by constructing a composite curve 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 Figures A-3 and A-4.

Finally, the new 10CFR50[3] rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Trojan). Table A-1 indicates that the limiting

RT_{NDT} of 20°F occurs in the closure head flange of Trojan, and the minimum allowable temperature of this region is 140°F at pressures greater than 621 psig.

A-4. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using methods discussed in Section A-3. The derivation of the limit curves is presented in the NRC Regulatory Standard Review Plan.[4]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. Charpy test specimens from Capsule X indicate that the surveillance weld metal and core region lower shell plate heat No. C5583-1 exhibited shifts in RT_{NDT} of 50°F and 95°F, respectively. These shifts at a fluence of $1.77 \times 10^{19} n/cm^2$ are well within the appropriate design curve (Figure A-1) prediction. As a result, the heatup and cooldown curves are based on the ΔRT_{NDT} given in Figure A-1 for the most limiting beltline material which is the lower shell plate heat no. B9883-1. The resultant heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures A-3 and A-4 and represent an operational time period of 10 EFPY. These limit curves are impacted by the new 10CFR50 rule.

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 Figure A-3. This is in addition to other criteria which must be met before the reactor is made critical.

The leak test limit curve shown in Figure A-3 represents minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of References 2 and 4.

Figures A-3 and A-4 define limits for ensuring prevention of nonductile failure.

TABLE A-1
TROJAN REACTOR VESSEL TOUGHNESS TABLE

Component	Heat No.	Grade	Cu (%)	P (%)	NDT (°F)	Minimum 50 ft lb/ 35 mil Temp MWD (°F)	Minimum 50 ft lb/ 35 mil Temp NMWD (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy MWD (ft lb)	Average Upper Shelf Energy NMWD (ft lb)
Closure Head Dome	B0048-2	A533B,cl 1	—	—	− 20	8	28[a]	− 20	148	96[a]
Closure Head Torus	C6096-1	A533B,cl 1	—	—	0	10	30[a]	0	149.5	97[a]
Closure Head Torus	B0042-2	A533B,cl 1	—	—	− 20	13	33[a]	− 20	161.5	105[a]
Closure Head Flange	4436-V1	A508,cl 2	—	—	20	− 20	0[a]	20	151	98[a]
Vessel Flange	4437-V2	A508,cl 2	—	—	10	17	37[a]	10	132.5	86[a]
Inlet Nozzle	9-7987-1	A508,cl 2	—	—	− 80	10	30[a]	− 30	105	68[a]
Inlet Nozzle	9-7260-1	A508,cl 2	—	—	0	48	68[a]	8	111	72[a]
Inlet Nozzle	9-7235-1	A508,cl 2	—	—	− 120	45	65[a]	5	95	62[a]
Inlet Nozzle	9-7208-1	A508,cl 2	—	—	− 20	− 25	− 5[a]	− 20	115.5	75[a]
Outlet Nozzle	9-7315-1	A508,cl 2	—	—	− 30	0	20[a]	− 30	126	82[a]
Outlet Nozzle	9-7251-1	A508,cl 2	—	—	− 75	− 25	− 5[a]	− 65	120	78[a]
Outlet Nozzle	9-7301-1	A508,cl 2	—	—	− 100	20	40[a]	− 20	126	82[a]
Outlet Nozzle	9-7241-1	A508,cl 2	—	—	− 130	45	65[a]	5	101	66[a]
Nozzle Shell	C5570-2	A533B,cl 1	0.11	0.014	0	20	40[a]	0	127.5	83[a]
Nozzle Shell	C5571-1	A533B,cl 1	0.15	0.013	0	− 6	14[a]	0	118.5	77[a]
Nozzle Shell	C6529-2	A533B,cl 1	0.14	0.010	− 20	10	30[a]	− 20	131	85[a]
Intermediate Shell	C5582-1	A533B,cl 1	0.12	0.009	− 10	− 10	60	0	121.5	101
Intermediate Shell	C5587-1	A533B,cl 1	0.15	0.014	10	38	40	10	113	117

a) Estimated

MWD - Longitudinal axis of Charpy specimen oriented in the major working direction

NMWD - Longitudinal axis of Charpy specimen oriented normal to the major working direction

TABLE A-1(cont)
TROJAN REACTOR VESSEL TOUGHNESS TABLE

Component	Heat No.	Grade	Cu (%)	P (%)	NDT (°F)	Minimum 50 ft lb/ 35 mil Temp MWD (°F)	Minimum 50 ft lb/ 35 mil Temp NMWD (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy MWD (ft lb)	Average Upper Shelf Energy NMWD (ft lb)
Lower Shell	B9883-1	A533B,cl 1	0.16	0.012	−10	30	70	10	121	99
Lower Shell	C5583-1	A533B,cl 1	0.15	0.011	0	18	60	0	113	84
Bottom Head Torus	C6123-3	A533B,cl 1	—	—	−20	15	35[a]	−20	159	103[a]
Bottom Head Torus	C5823-1	A533B,cl 1	—	—	−20	4	16[a]	−20	143.5	93[a]
Bottom Head Dome	B0018-1	A533B,cl 1	—	—	−50	−25	−5[a]	−50	177	115[a]
Weld Metal			0.06	0.019	−20	—	35	−20	—	102.5
Weld HAZ			—	—	−60	—	−10	−60	—	110

a) Estimated

MWD - Longitudinal axis of Charpy specimen oriented in the major working direction

NMWD - Longitudinal axis of Charpy specimen oriented normal to the major working direction

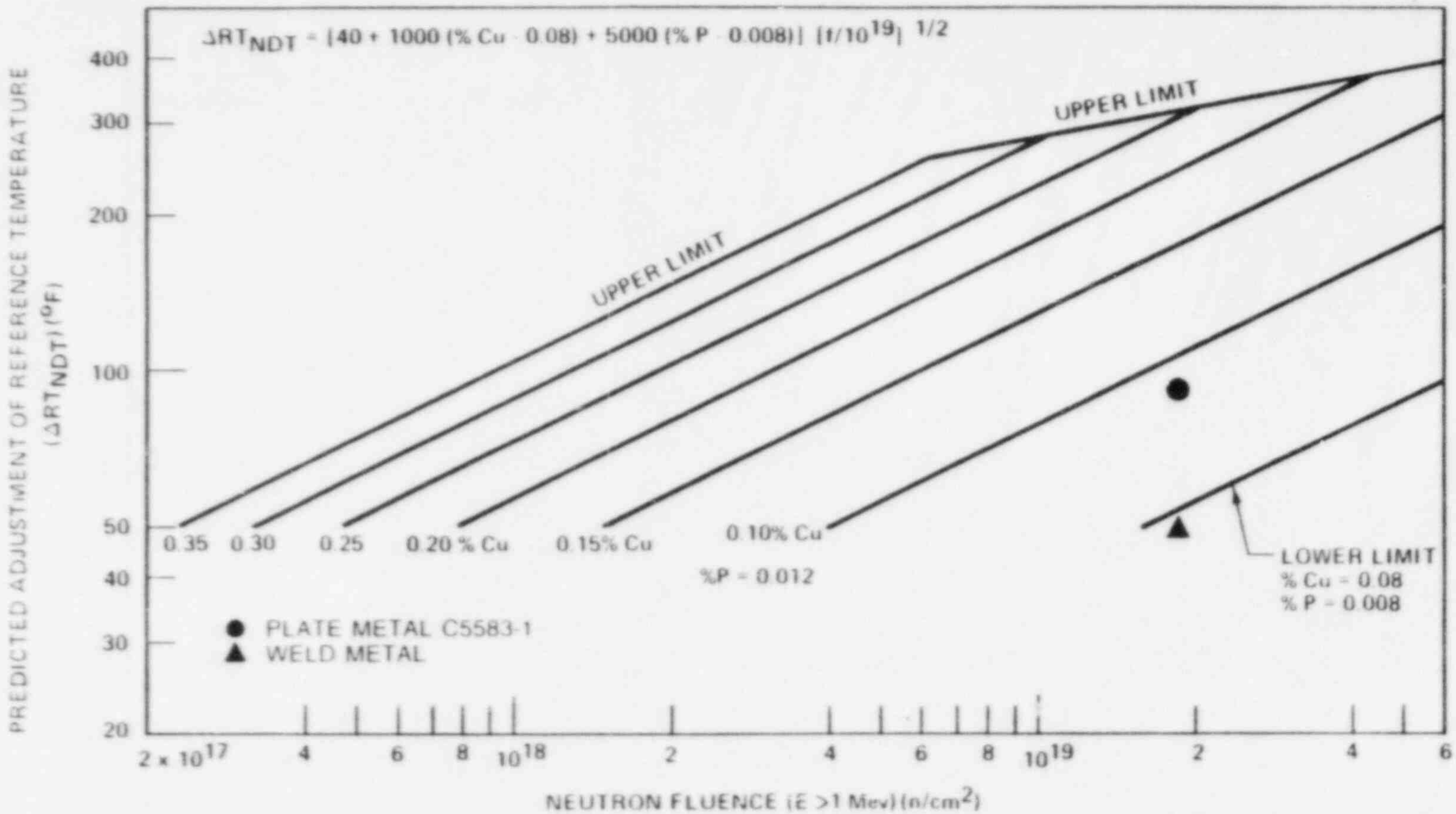

FIGURE A-1. PREDICTED ADJUSTMENT OF REFERENCE TEMPERATURE AS A FUNCTION OF FLUNCE, COPPER, AND PHOSPHORUS CONTENTS.

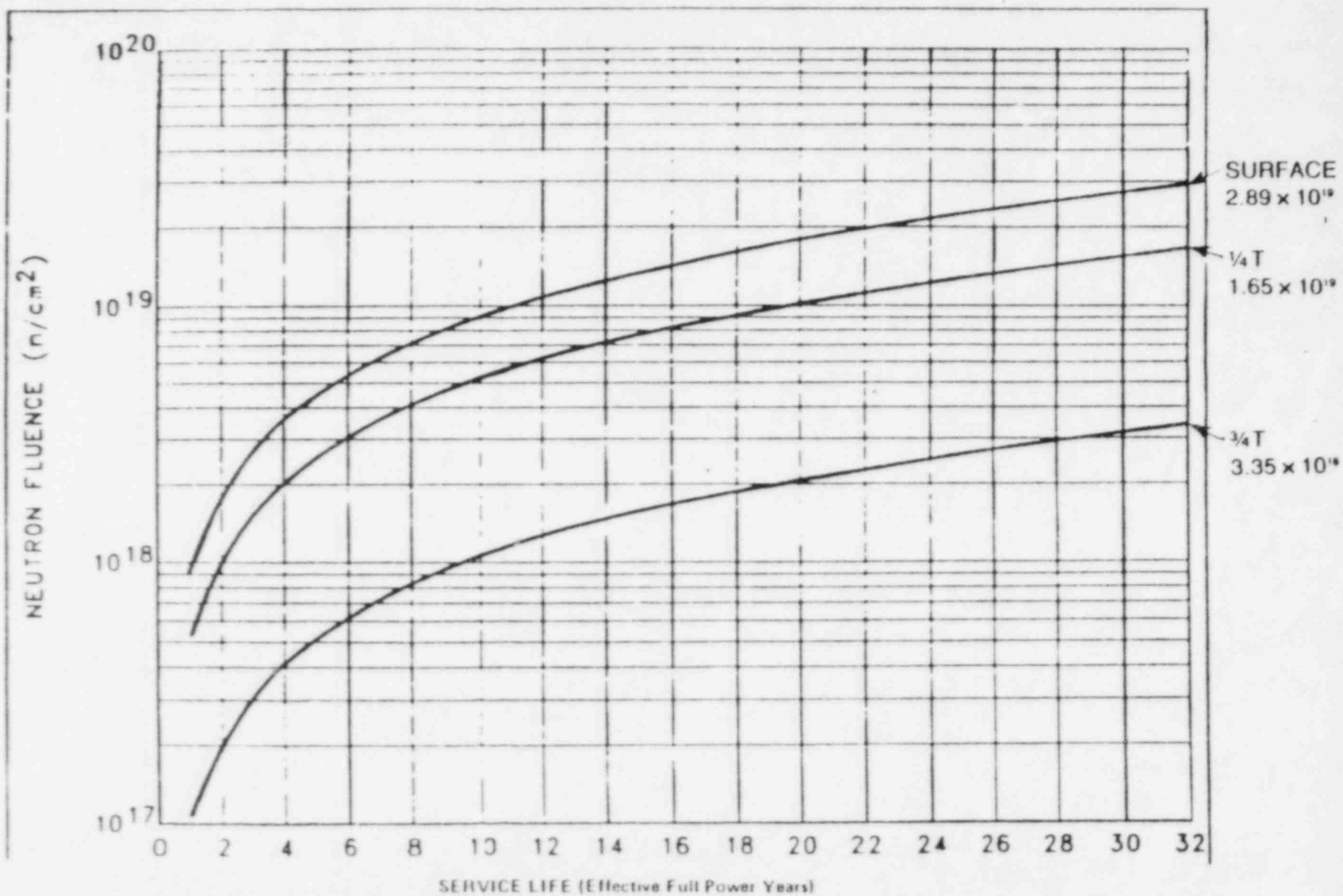

FIGURE A-2. FAST NEUTRON FLUENCE (E > 1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

MATERIAL PROPERTY BASIS:

CONTROLLING MATERIAL : R.V. LOWER SHELL
COPPER CONTENT : 0.16 WT%
PHOSPHORUS CONTENT : 0.012 WT%
INITIAL RT_{NDT} : 10°F
RT_{NDT} AFTER 10 EFPY : ¼T, 111°F
: ¾T, 55°F

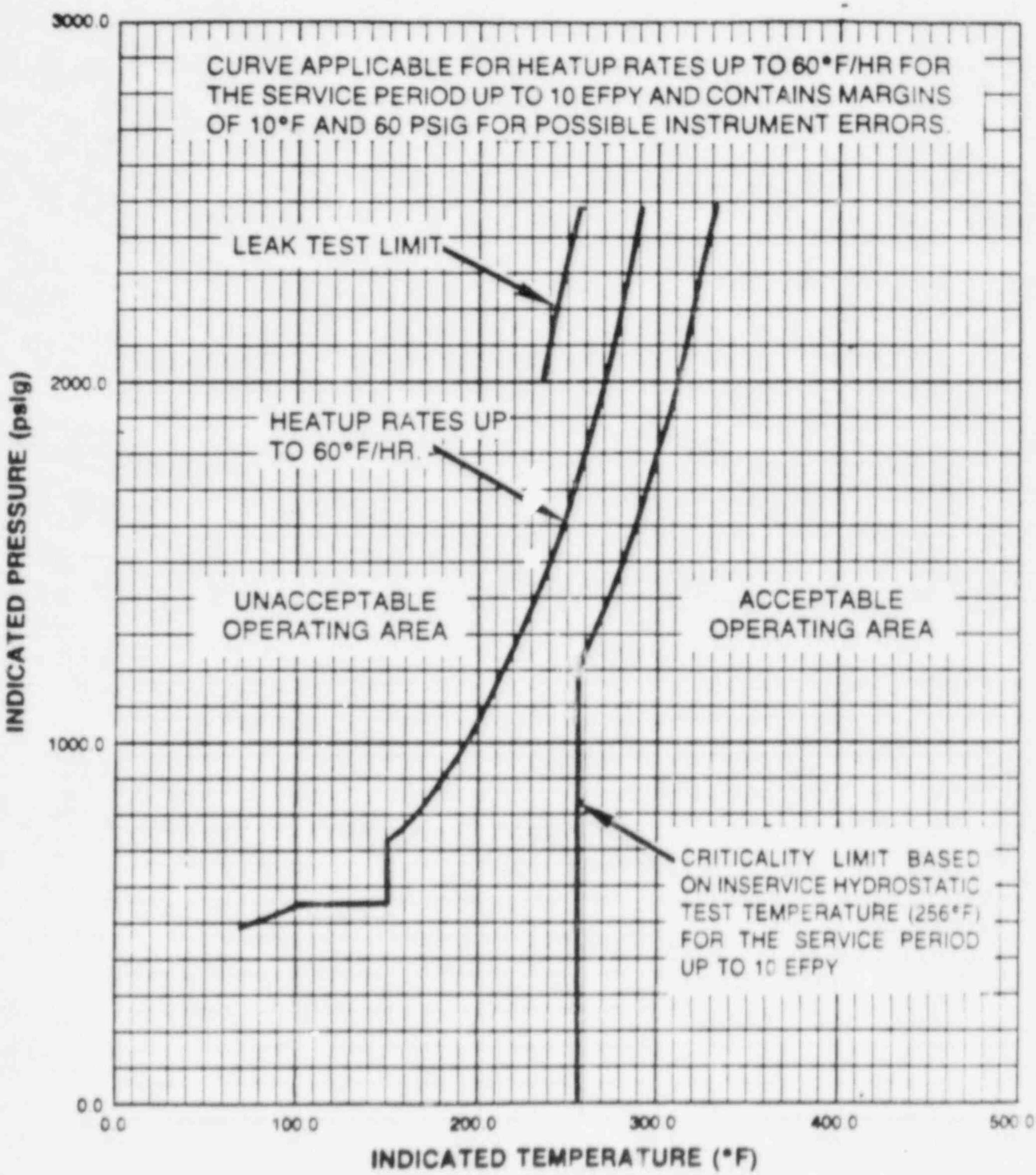

FIGURE A-3. TROJAN REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

MATERIAL PROPERTY BASIS:

CONTROLLING MATERIAL	R.V. LOWER SHELL
COPPER CONTENT	0.16 WT%
PHOSPHORUS CONTENT	0.012 WT%
INITIAL RT_{NDT}	10°F
RT_{NDT} AFTER 10 EFPY	¼T, 111°F
	¾T, 55°F

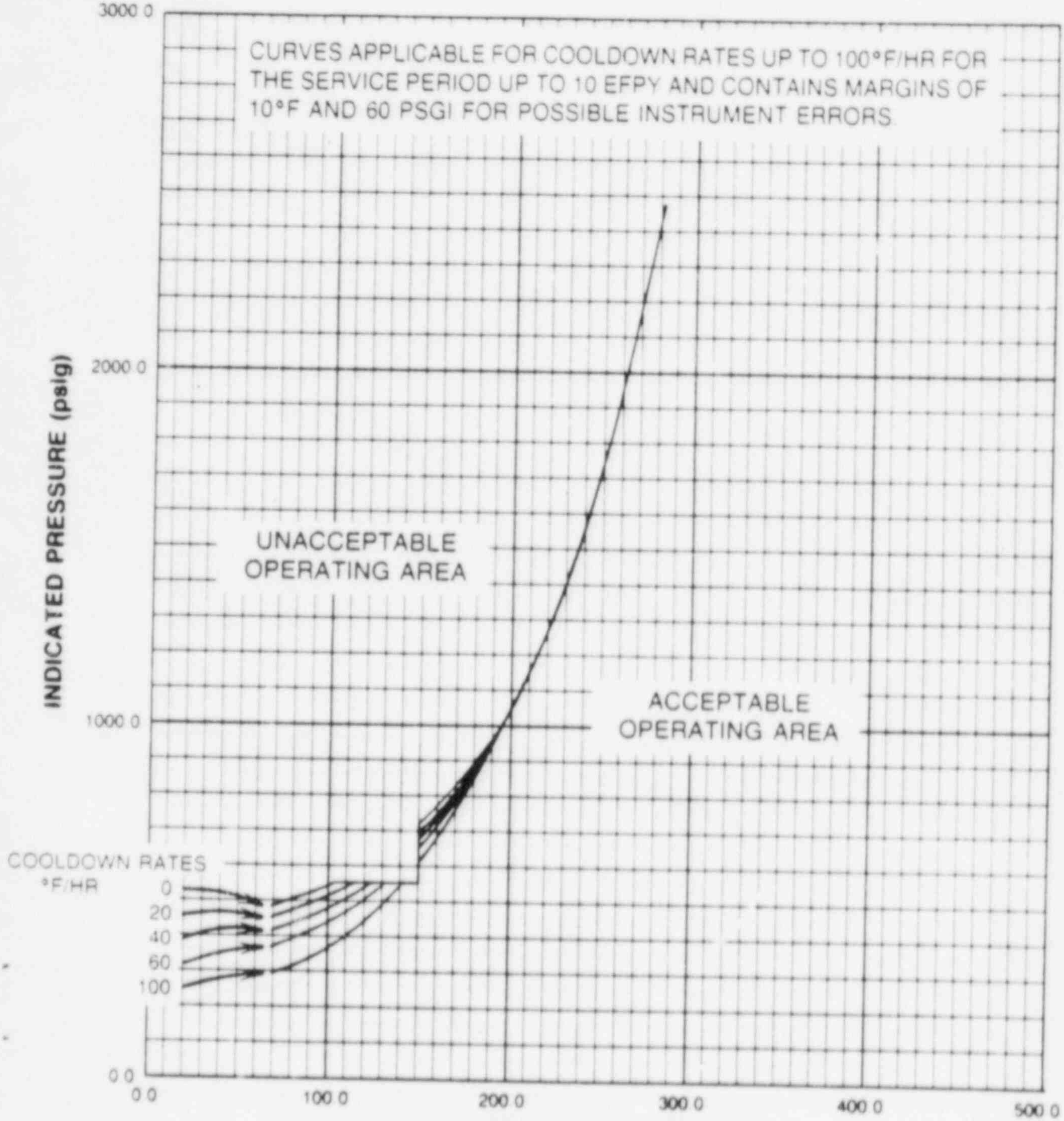

FIGURE A-4. TROJAN REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

APPENDIX A
REFERENCES

1. "Fracture Toughness Requirements," Branch Technical Position MTEB 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.

2. 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.

3. Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U. S. Nuclear Regulatory Commission, Washington, D. C., Amended May 17, 1983 (48 Federal Register 24010).

4. "Pressure-Temperature Limits," Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.