WESTINGHOUSE CLASS 3
CUSTOMER DESIGNATED DISTRIBUTION

WCAP- 11492

ANALYSIS OF CAPSULE V FROM THE VIRGINIA POWER COMPANY SURRY UNIT 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

SURRY UNIT 1 REACTOR VESSEL HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

May 1987

Prepared by: John C. Schmertz
J. C. Schmertz

Verified by: H. Gong
H. Gong

Approved by: [signature]
S. S. Palusamy, Manager
Structural Materials Engineering

Work Performed Under Shop Order No. VCKJ-106

Prepared by Westinghouse for the Virginia Power Company

Although information contained in this report is nonproprietary, no distribution shall be made outside Westinghouse or its licensees without the customer's approval.

WESTINGHOUSE ELECTRIC CORPORATION
Power Systems Division
P.O. Box 2728
Pittsburgh, Pennsylvania 15230-2728

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

1.0 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) for the reactor vessel. 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 fracture toughness 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, 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 Predicted 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[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 metals (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 > 1 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 life in figure 2 for the vessel core region.

2.0 FRACTURE TOUGHNESS PROPERTIES

The preirradiation fracture-toughness properties of the Surry Unit 1 reactor vessel materials are presented in table III. 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[3]. The postirradiation fracture-toughness properties of the reactor vessel beltline material were obtained directly from the Surry Unit 1 Vessel Material Surveillance Program. These results show that the transition temperature shift is less than the shift plus margin as predicted by Rev. 2 of Reg. Guide 1.99, thus validating the conservatism of Reg. Guide 1.99 Rev. 2 for generating the heatup and cooldown curves.

3.0 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[4]. The K_{IR} curve is given by the following equation:

$$K_{IR} = 27.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \qquad (1)$$

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[4] as follows:

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

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 the 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 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 the 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 1/4 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 1/4 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 1/4 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 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 1/4 T defect 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 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 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 1/4 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 the pressure-temperature limitations for the case in which a 1/4 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 therefore 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 3 and 4.

Finally, the 1983 Amendment to 10CFR50[5] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This 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 Surry Unit 1). Table III indicates that the limiting RT_{NDT} of 10°F occurs in the vessel flange of Surry Unit 1, so the minimum allowable temperature of this region is 130°F at pressures greater than 621 psig. These limits are less restrictive than the limits shown on figures 3 and 4.

4.0 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 in section 3, and the procedure is presented in reference 6.

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

The leak limit curve shown in figure 3 represents minimum temperature requirements at the leak test pressure specified by applicable codes[3,4]. The leak test limit curve was determined by methods of references 3 and 5.

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

5.0 ADJUSTED REFERENCE TEMPERATURE

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 metals (plates and forgings). Linear interpolation is permitted. In table 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.

At the vessel inside radius, the calculated neutron fluences for 15 effective full power years (EFPY) are 2.12×10^{19} n/cm^2 at the 0° azimuth (where there are no vertical weld seams) and 3.42×10^{18} n/cm^2 at the 45° azimuth (where the longitudinal weld seams are located). The fluence factors at these two azimuth locations are 1.2043 and 0.7044, respectively.

Applying Regulatory Guide 1.99 Revision 2 procedures to all the beltline region materials, and using tables I and II, it was found that the circumferential weld seam between the vessel intermediate and lower shell course was the limiting material. Credible surveillance data is currently limited to the vessel beltline region vertical weld seam L2, and the chemistry factor for this data is 216, which is less than the chemistry factor of 222 obtained at this same location using table I. Therefore it is conservative to use the Regulatory 1.99 revision 2 procedure and use the chemistry factors from tables I and II.

For the limiting circumferential weld, the chemistry factor is 160.9, based on table I. From equation (4), the ΔRT_{NDT} at the inner surface is equal to 193.8°F (160.9 x 1.2043). Regulatory Guide 1.99 revision 2 provides a formula and rules for establishing margin:

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \qquad (6)$$

Since the initial RT_{NDT} values for the vessel beltline region welds are generic, the temperature σ_I is taken as 17°F. The standard deviation σ_Δ is taken as 19°F[7] for welds, and this is less than 1/2 of [ΔRT_{RDT} surface]. As a result, the margin is $2\sqrt{17^2 + 19^2}$ = 50.99°F. Substituting the obtained values for ΔRT_{NDT} surface and margin, in addition to the initial RT_{NDT} of -6°F[7] into Equation (3) gives the adjusted reference temperature (ART) for the inside surface:

$$\text{ART} = -6 + 193.8 + 50.99 = 238.8°F$$

Using the vessel thickness of 8.05 inches at the beltline, Equation (5) is used to calculate the ART at the 1/4 and 3/4 thickness locations. These are 214.3°F and 174.3°F respectively.

The above analysis was used to develop the Surry unit 1 heatup and cooldown curves shown in figures 3 and 4, respectively.

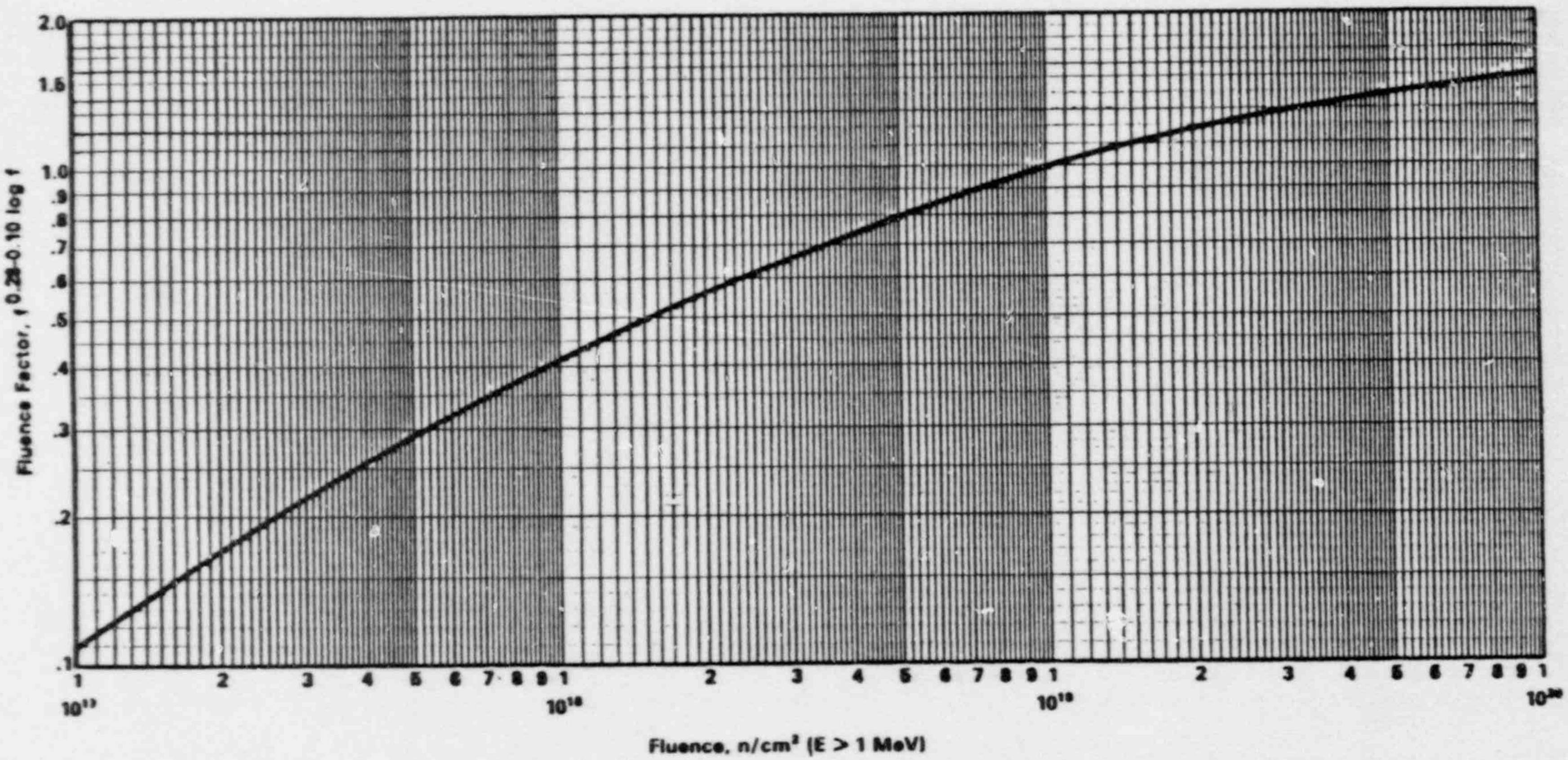

Figure 1. Fluence Factor for Use in the Expression for ΔRT_{NDT}

2369s/042987:10

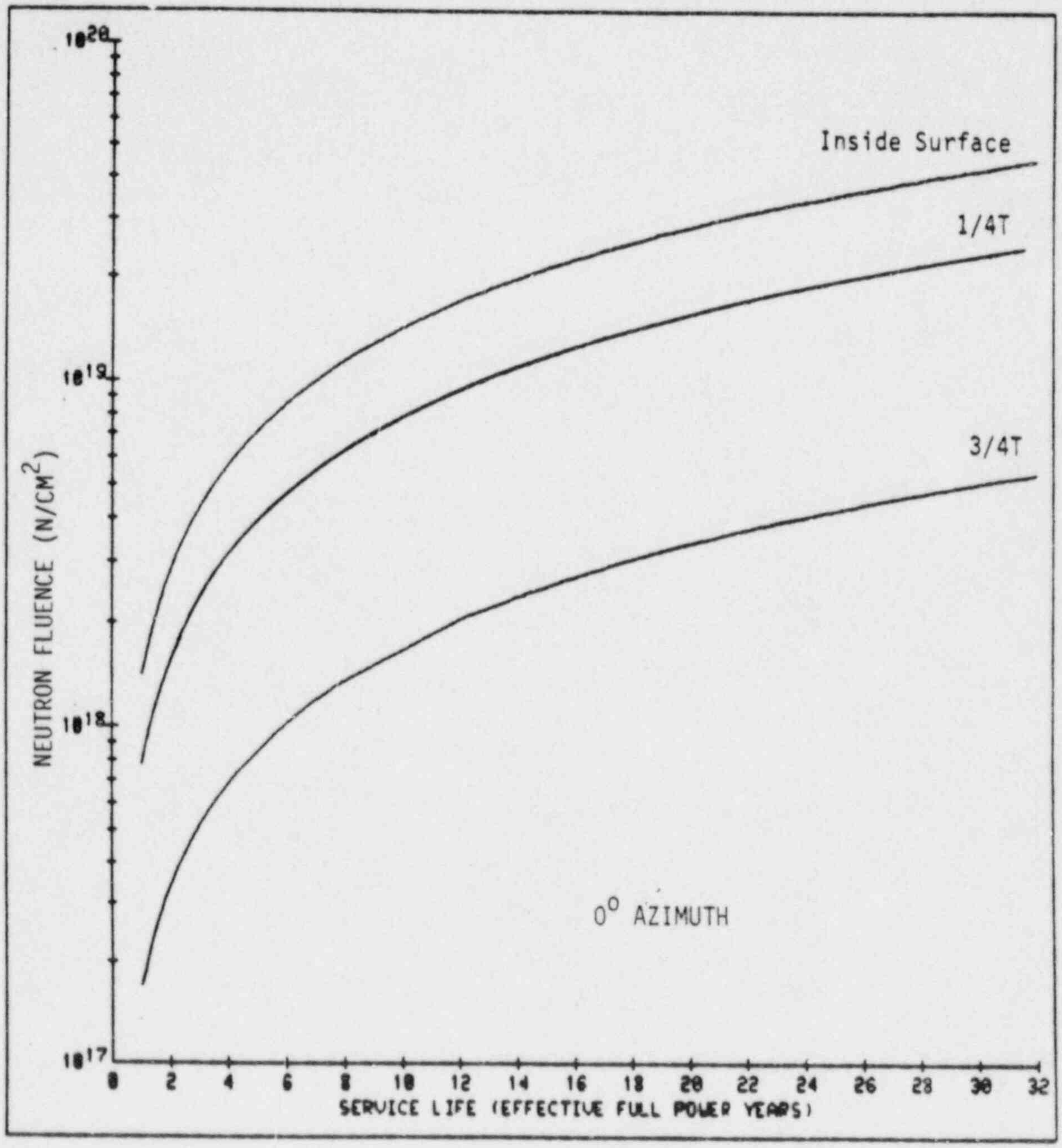

Figure 2 Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life (EFPY) for Surry Unit 1

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
COPPER CONTENT: 0.21 WT%
NICKEL CONTENT: 0.58 WT%
INITIAL RT_{NDT}: $-6^{\circ}F$

RT_{NDT} AFTER 15 EFPY: 1/4T, $214.3^{\circ}F$
3/4T, $174.3^{\circ}F$

CURVES APPLICABLE FOR HEATUP RATES UP TO $60^{\circ}F/HR$ FOR THE SERVICE PERIOD UP TO 15 EFPY AND CONTAINS MARGINS OF $10^{\circ}F$ AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

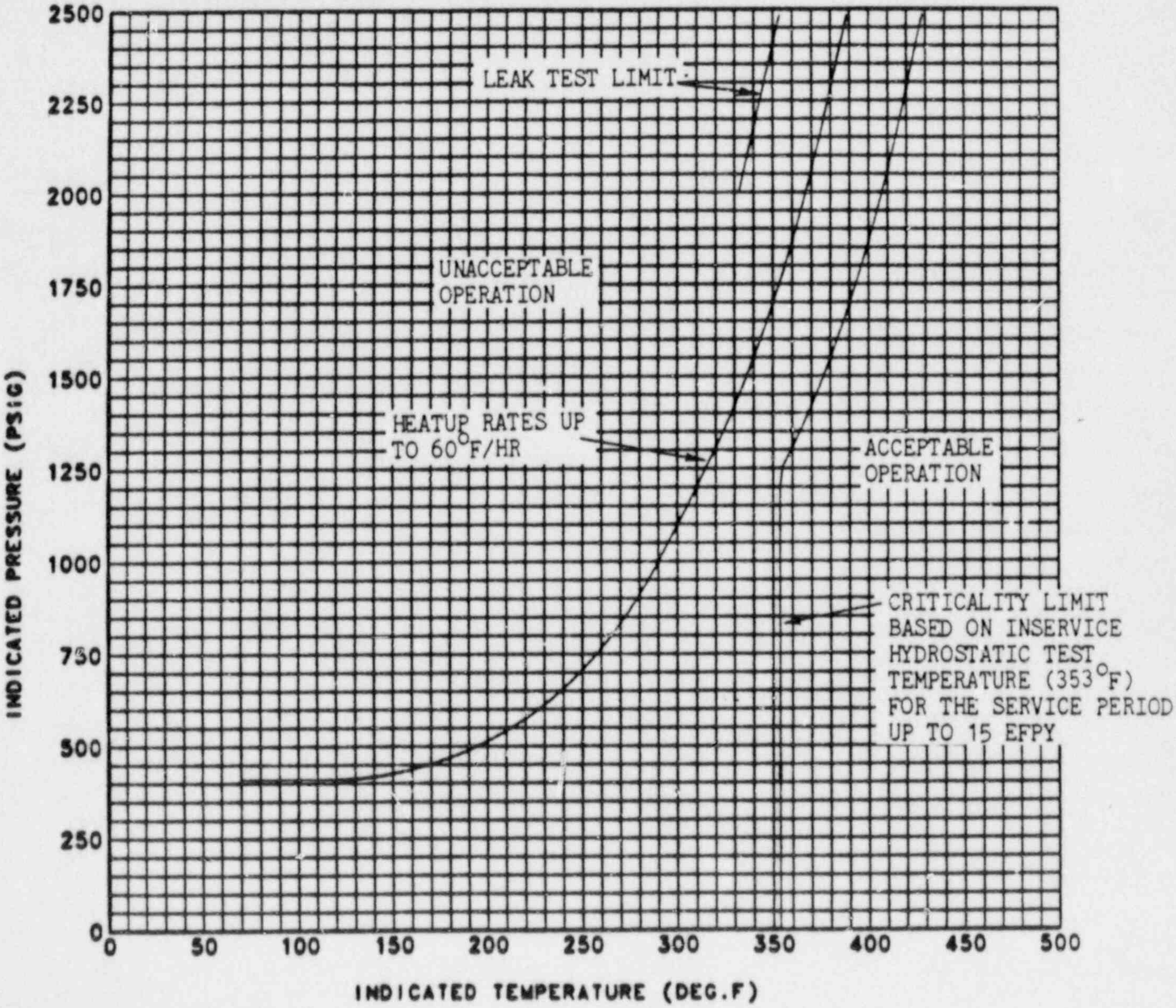

Figure 3. Surry Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 15 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
COPPER CONTENT: 0.21 WT%
NICKEL CONTENT: 0.58 WT%
INITIAL RT_{NDT}: $-6^{o}F$

RT_{NDT} AFTER 15 EFPY: 1/4T, $214.3^{o}F$
3/4T, $174.3^{o}F$

CURVES APPLICABLE FOR COOLDOWN RATES UP TO $100^{o}F/HR$ FOR THE SERVICE PERIOD UP TO 15 EFPY AND CONTAINS MARGINS OF $10^{o}F$ AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

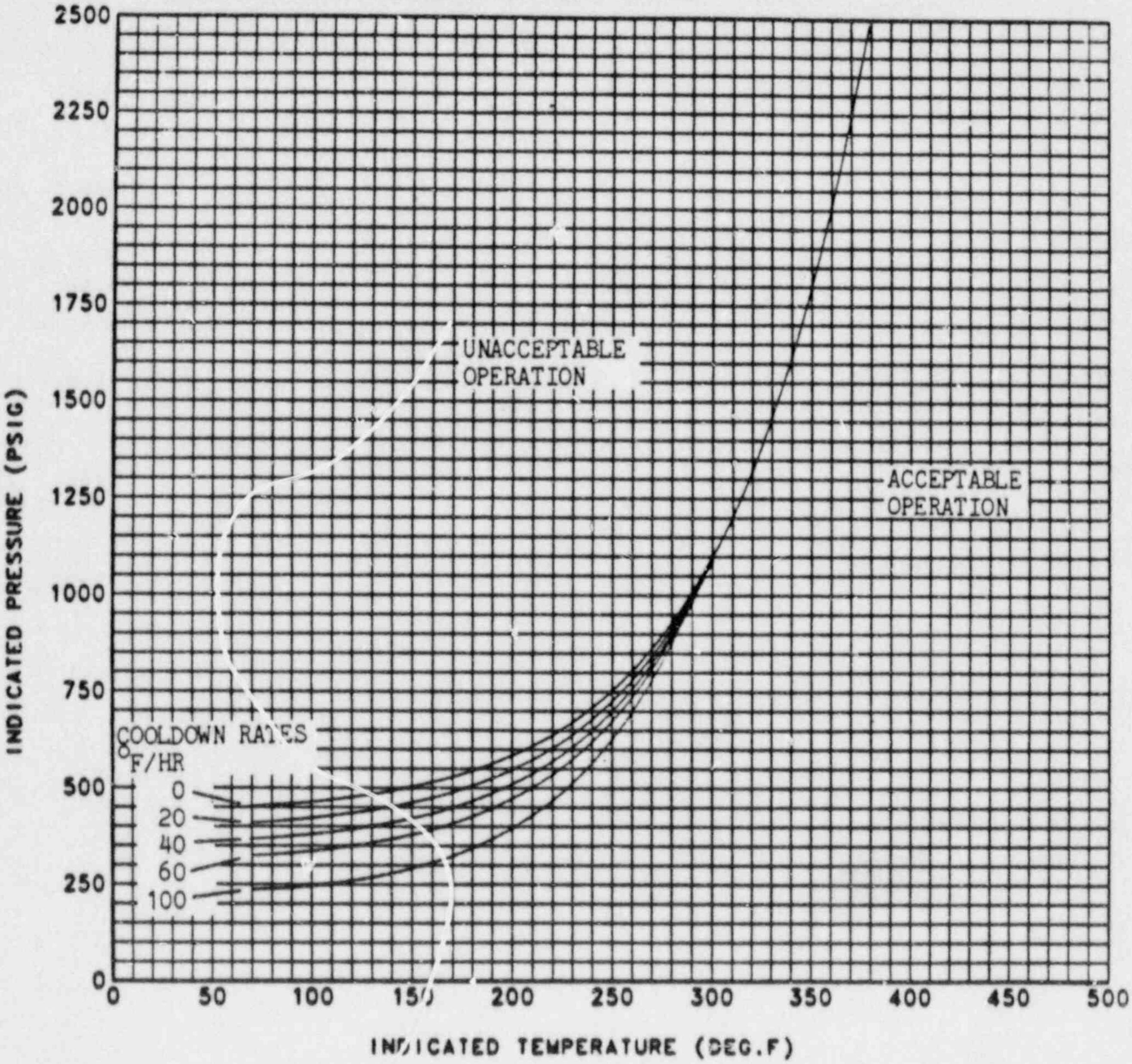

Figure 4. Surry Unit No. 1 Reactor Coolant System Cooldown Limitations Applicable for the First 15 EFPY

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

Table I (Cont'd.)

CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
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	1[illegible]7	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

Table II (Cont'd.)

CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
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

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)[d]

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Closure head dome	C4315-2	A53313 Cl.1	.14	.59	.011	0	0	75
Head flange	FV-1894	A508 Cl. 2	.13	.64	.010	10[a]	10	125
Vessel flange	FV-1870	A508 Cl. 2	.10	.65	.009	10[a]	10	74
Inlet nozzle	9-5078	A508 Cl. 2	-	.87	.007	60[a]	60	64
Inlet nozzle	9-4819	A508 Cl. 2	-	.84	.008	60[a]	60	68
Inlet nozzle	9-4787	A508 Cl. 2	-	.85	.007	60[a]	60	64
Outlet nozzle	9-4762	A508 Cl. 2	-	.83	.007	60[a]	60	85
Outlet nozzle	9-4788	A508 Cl. 2	-	.84	.007	60[a]	60	72
Outlet nozzle	9-4825	A508 Cl. 2	-	.85	.008	60[a]	60	68
Upper shell	122V109	A508 Cl. 2	.07	.74	.010	40	40	83
Intermediate shell	C4326-1	A533B Cl. 1	.11	.55	.008	10	10	115[c]
Intermediate shell	C4326-2	A533B Cl. 1	.11	.55	.008	0	0	93
Lower shell	C4415-1	A533B Cl. 1	.11	.50	.014	20	20	103[c]
Lower shell	C4415-2	A533B Cl. 1	.11	.50	.014	0	0	80
Bottom head ring	123T338	A508 Cl. 2	-	.69	.020	50	50	86
Bottom dome	C4315-3	A533B Cl. 1	.14	.59	.011	0	0	85
Inter. & lower shell vertical weld seam L1, L3, & L4	8T1554 & Linde 80 flux		.18	.63	.014	0[a]	0	N/A

TABLE III (Cont'd.)

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (%)	RT_{NDT} (%)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Lower shell vertical vertical weld seam	299L44 & Linde 80 flux		.35	.67	.014	0[a]	0	70[a]
Inter. to lower shell girth seam	72445 & Linde 80 flux		.21	.58	.016	0[a]	-6	N/A

NOTES:

(a) Estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(b) Normal to major working direction - estimated per NRC standard review plan, NUREG-0800, section MTEB 5-2

(c) Actual values

(d) Reactor Vessel Fabricator Certified Test Reports.

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. "Proposed Revision 2 to Regulatory Guide 1.99, Radiation Damage to Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, February, 1986.

3. "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.

4. ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "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.

5. 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).

6. "Procedure for Generating Heatup and Cooldown Limit Curves," W. T. Kaiser, Westinghouse Report MT-SME-1419, February, 1980.

7. "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for Surry Units 1 and 2," Babcock and Wilcox Report BAW-1909, March 1986.