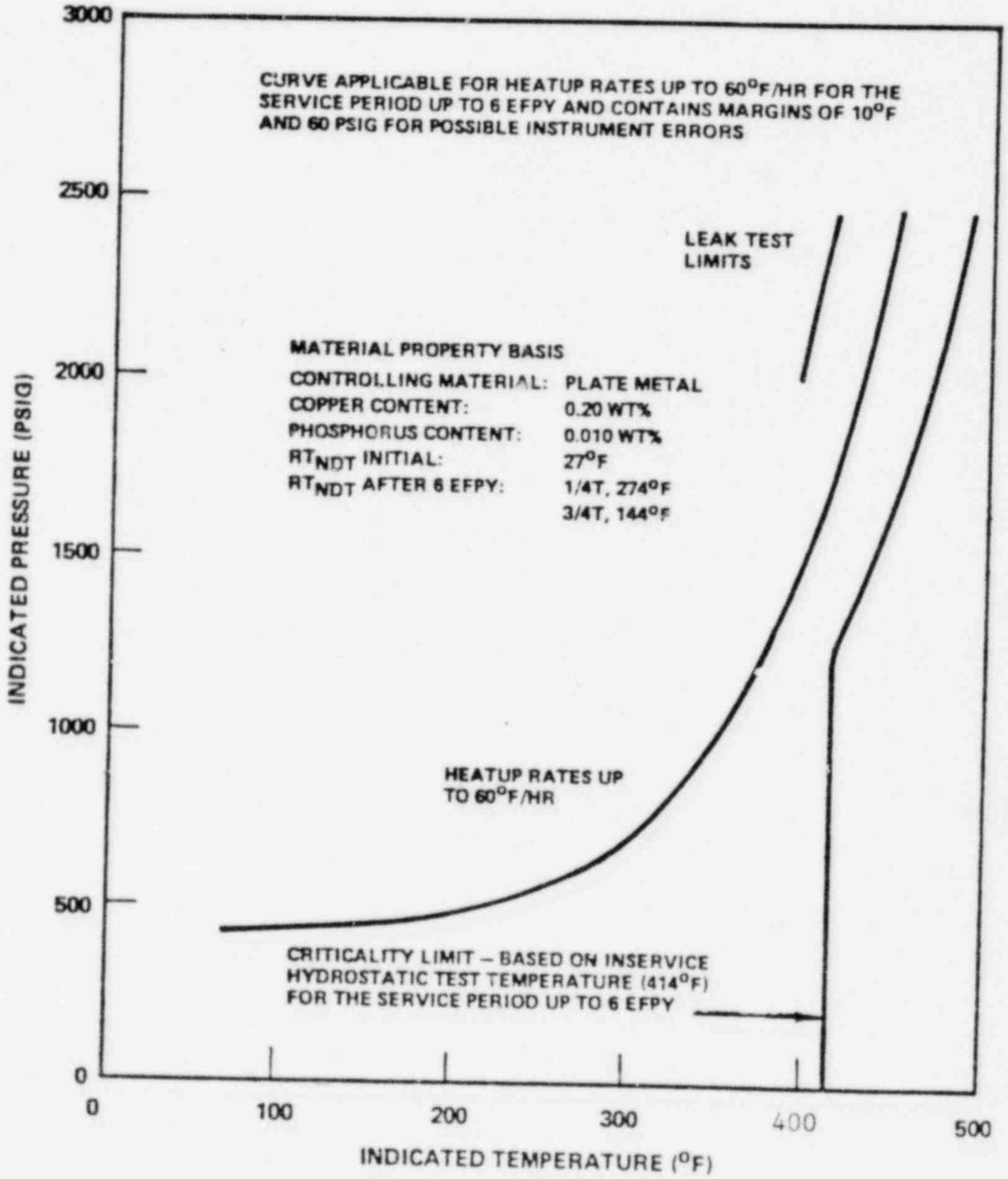

Figure 3.4-2 Beaver Valley Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First 6 EFPY

PROPOSED WORDING

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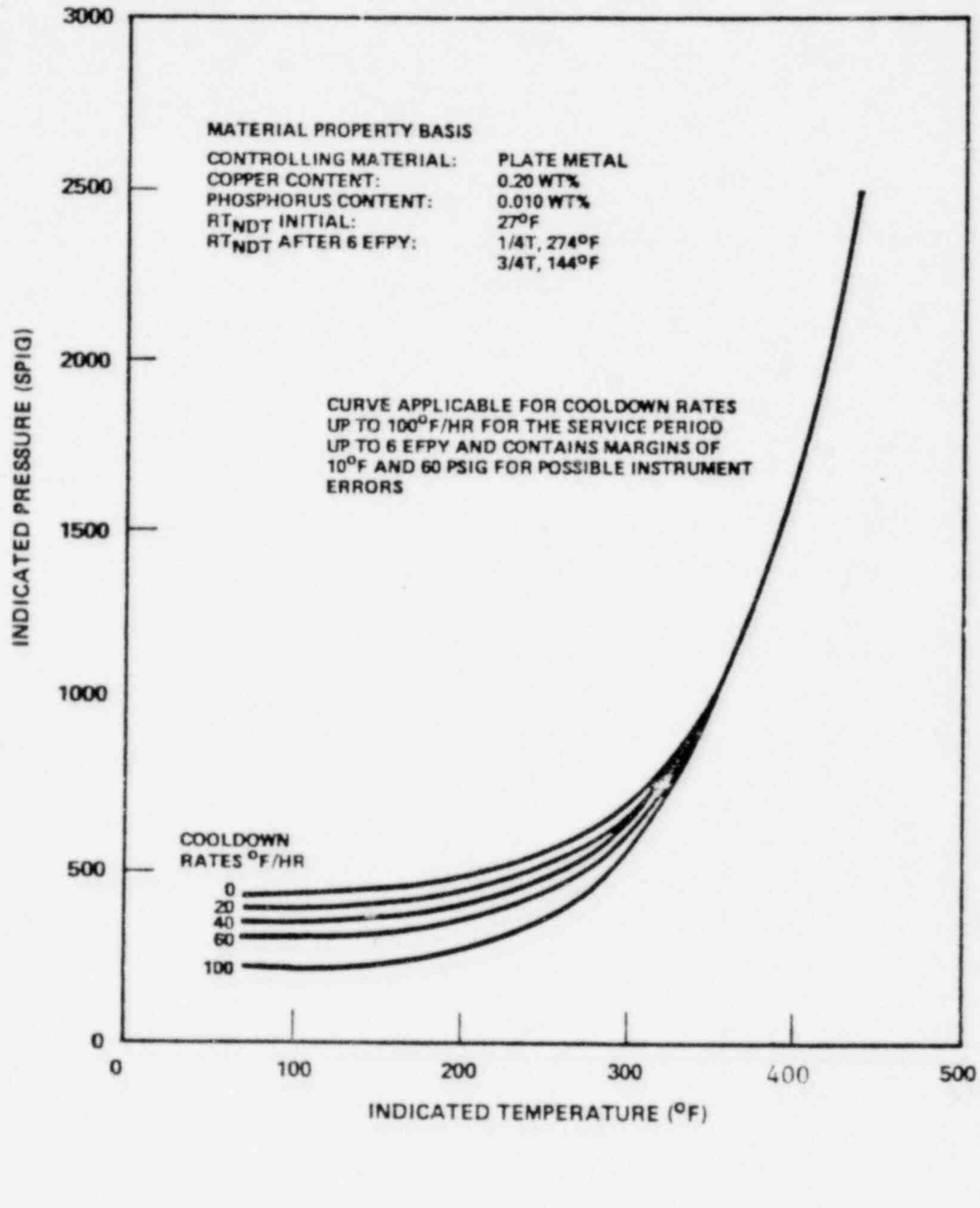


Figure 3.4-3 Beaver Valley Unit No. 1 Reactor Coolant System Cooldown Limitations Applicable for the First 6 EFPY

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying that on a Containment Pressure--High-High signal, the recirculation spray pumps start automatically as follows:

RS-P-1A and RS-P-2B	210 ± 5 second delay
RS-P-2A and RS-P-1B	225 ± 5 second delay

c. At least once per 18 months, during shutdown, by verifying, that on recirculation flow, each outside recirculation spray pump develops a discharge pressure of $\geq$ 115 psig at a flow of $\geq$ 2000 gpm.

d. At least once per 18 months during shutdown, by:

1. Cycling each power operated (excluding automatic) valve in the flow path not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

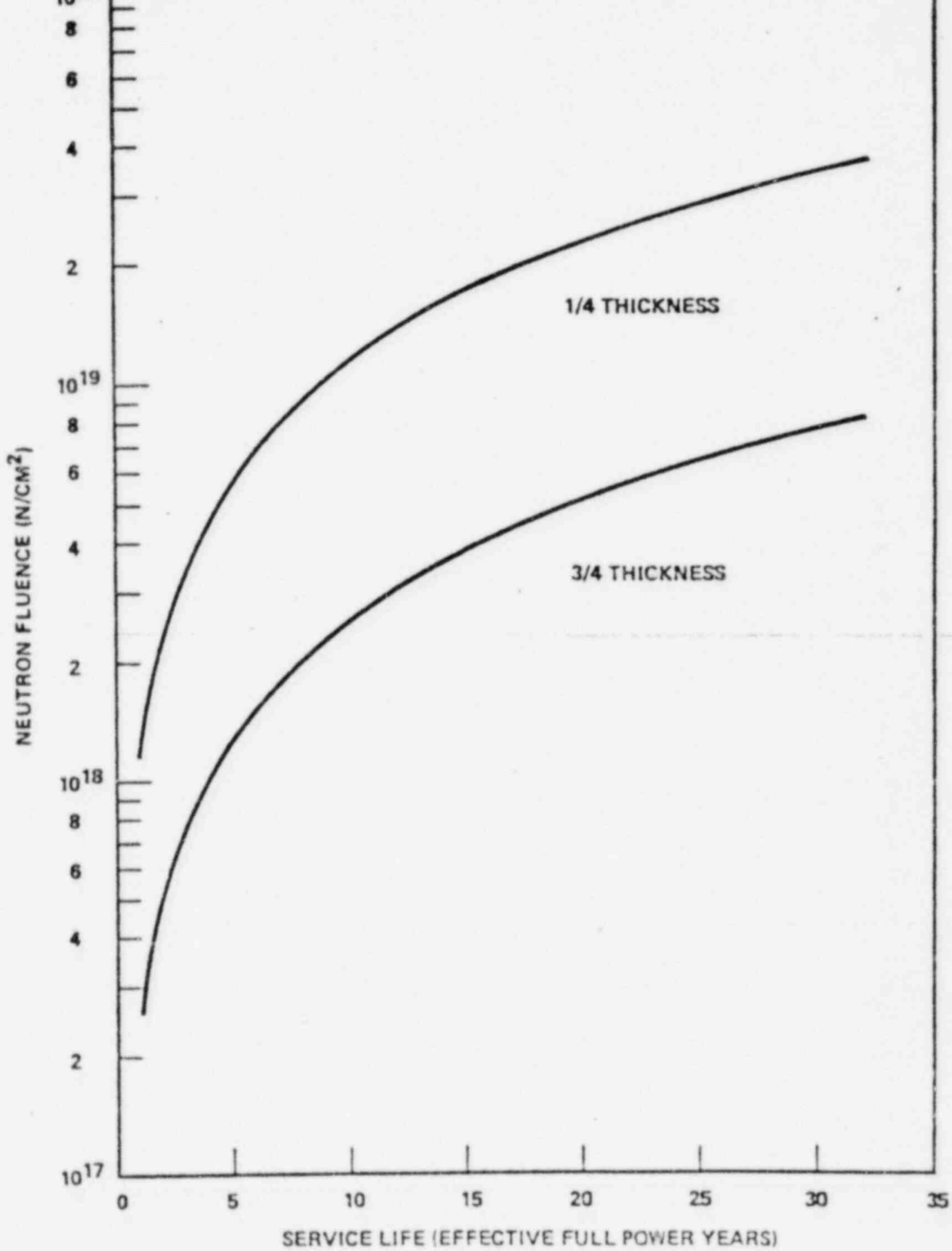
PROPOSED WORDING

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 6 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} as well as adjustments for possible errors in the pressure and temperature sensing instruments.

PROPOSED WORDING



Fast Neutron Fluence (E > 1 Mev) as a Function of Full Power Service Life

Figure B 3/4.4-1

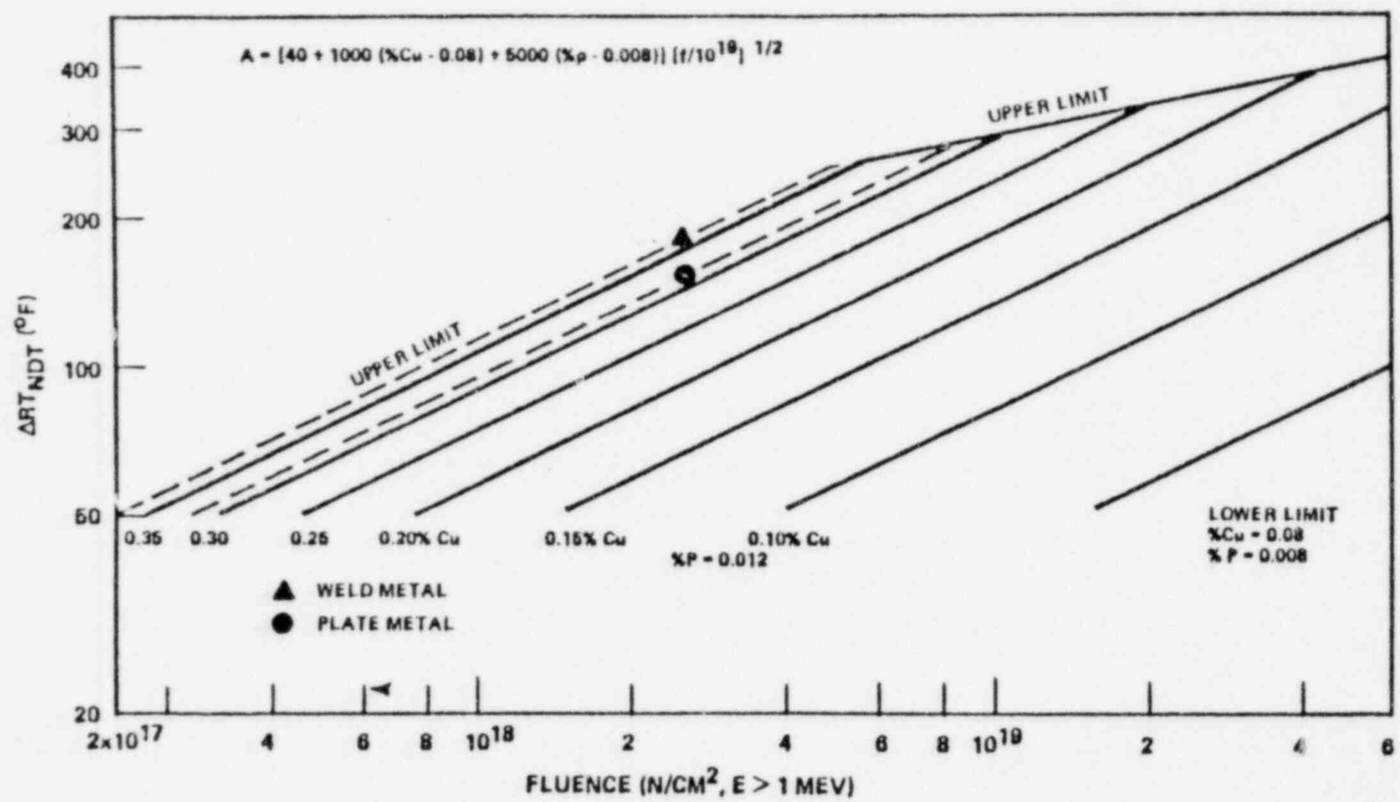


Figure B 3/4.4-2

Effect of Fluence, Copper Content, and Phosphorus Content on ΔRT_{NDT} for Reactor Vessel Steels per Regulatory Guide 1.99

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Heat No.	Code No.	Material Type	Cu (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shelf Energy (Ft-lb) MWD	NMWD
Closure Head Dome	C6213-1B	B6610	A533B CL. 1	.15	.010	-40	0*	121	-
Closure Head Seg.	A5518-2	B6611	A533B CL. 1	.14	.015	-20	-20*	131	-
Closure Head Flange	ZV3758	-	A508 CL. 2	.08	.007	60*	60*	> 100	-
Vessel Flange	ZV3661	-	A508 Cl. 2	.12	.010	60*	60*	166	-
Inlet Nozzle	9-5443	-	A508 Cl. 2	.10	.008	60*	60*	82.5	-
Inlet Nozzle	9-5460	-	A508 Cl. 2	.10	.010	60*	60*	94	-
Inlet Nozzle	9-5712	-	A508 Cl. 2	.08	.007	60*	60*	97	-
Outlet Nozzle	9-5415	-	A508 Cl. 2	-	.008	60*	60*	97	-
Outlet Nozzle	9-5415	-	A508 Cl. 2	-	.007	60*	60*	112.5	-
Outlet Nozzle	9-5444	-	A508 Cl. 2	.09	.007	60*	60*	103	-
Upper Shell	123V339	-	A508 Cl. 2	-	.010	40	40*	155	-
Inter. Shell	C4381-2	B6607-2	A533B Cl. 1	.14	.015	-10	73	123	82.5
Inter. Shell	C4381-1	B6607-1	A533B Cl. 1	.14	.015	-10	43	128.5	90
Lower Shell	C6317-1	B6903-1	A533B Cl. 1	.20	.010	-50	27	134	80
Lower Shell	C6293-2	B7203-2	A533B Cl. 1	.14	.015	-20	20	129.5	83.5
Trans. Ring	123V223	-	A508 Cl. 2	-	-	30	30*	143	-
Bottom Hd. Seg.	C4423-3	B6618	A533B Cl. 1	.13	.008	-30	-29*	124	-
Bottom Hd. Dome	C4482-1	B6619	A533B Cl. 1	.13	.015	-50	-3[illegible]*	125.5	-
Core Region Welds				.30-.37	.013	-	0*	-	> 100
Weld HAZ				-	-	-40	-40	-	136.5

*Estimated Per NRC Standard Review Plan Branch Technical Position MTEB 5-2

MWD - Major Working Direction

NMWD - Normal to Major Working Direction

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nilductility 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'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. The Regulatory Guide 1.99 trend curves which show the effect of fluence and copper and phosphorus contents on ΔRT_{NDT} for reactor vessel steels are shown in Figure B 3/4.4-2.

Given the copper and phosphorus contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Figure B 3/4.4-2. Fast-neutron fluence ($E > 1$ Mev) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations are given as a function of full-power service life in Figure B 3/4.4-1. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.

The preirradiation fracture-toughness properties of the Beaver Valley Unit 1 reactor vessel materials are presented in Table B 3/4.4-1. 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 Beaver Valley Unit 1 Vessel Material Surveillance Program.

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

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \qquad (4-1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nilductility temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G to the ASME Code [2] as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (4-2)$$

1. "Fracture Toughness Requirements," Branch Technical Position MTEB No. 5-2, Section 5.3.2-14 in Standard Review Plan, NUREG-75/087, 1975.

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. 461-469, 1980 Edition, American Society of Mechanical Engineers, New York, 1980.

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 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 (4-2), 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 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 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 insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

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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.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and

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