	ENCLOSURE 1
3.1.3 COMBINED	HEATUP, COOLDOWN AND PRESSURE LIMITATIONS
Applicability:	Applies to heatup and cooldown of the reactor coolant system.
<u>Objective</u> :	To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.
<pre>Specification:</pre>	A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 16 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 16 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly.
	B. Figures 3.1.3a and 3.1.3b shall be updated in accord- ance with the following criteria and procedures:
	(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature rela- tionships for the reactor coolant system.
	(2) The curves in Figure 3.1.3c shall be used in predicting the reference nil-ductility tempera- ture increase, ART _{NDT} , unless measurements on the irradiation specimens show ART _{NDT} s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.
	C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 195°F/hour shall not be exceeded.
	D. The reactor shall not be brought to a critical condi- tion until the pressure-temperature state is to the right of the criticality limit line as shown in Figures 3.1.3a.
<u>Basis</u> :	The initial Reference Nil Ductility Temperature (RT_{NDT}) for all reactor vessel material based on Charpy V-notch ^{NDT} data, drop weight tests, and conservative estimates* is $82^{\circ}F$ or less. The RT _{NDT} at the 1/4 thickness location (location of Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 240°F for the core region of the reactor vessel after 30 years of operation.
■ NKU Standard Re	eview Plan Branch Technical Position MTEB 5-2.

A sixteen (16) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting RT_{NDT} of the reactor vessel at the 1/4 thickness location is approximately 217 F in the core region. This RT_{NDT} is at least 50°F above the RT_{NDT} of all other regions in the primary reactor coolant system.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in the Table 3.1.3.1. The fast neutron (E > 1Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the ΔRT_{NDT} is obtained from Figure 3.1.3c.

Values of ΔRT_{NDT} may continue to be determined in this manner unless measurements on the irradiation specimens show $\Delta RT_{NDT}s$ greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from non-mandatory Appendix G in Section III of the ASNE Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.

The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The straight line portion of the criticality limit given in Figures 3.1.3a is at the minimum permissable temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10CFR Part 50. The curved portion of the criticality limit is shifted 40° F to the right of the heatup curve as required by Appendix G to 10CFR Part 50.

References:

- "Pressure Temperature Limits" Section 5.3.2 of Standard Review Plan, NUREG-751087, 1975.
- (2) S. E. Yanichko, et al, "Analysis of Capsule F from the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program", WCAP 9520, May 1979.

	and a second
2500	
2400	
2300	
2200	LEAK TEST LIMIT
2100	
	MATERIAL PROPERTY BASIS AT 1/4T LOCATION
2000	CONTROLLING MATERIAL: INTERMEDIATE SHELL
1900	PHOSPHORUS CONTENT: 0.014 WT%
1800	RT _{NOT} AFTER 16 EFPY: 217°F
1700	MATERIAL PROPERTY BASIS AT 3/4T LOCATION CONTROLLING MATERIAL: LOWER SHELL
	COPPER CONTENT: 0.14 WT% PHOSPHORUS CONTENT: 0.014 WT%
1600	RT _{NDT} INITIAL: 82°F RT _{NDT} AFTER 16 EFPY: 163°F
1500	
1400	CURVE APPLICABLE FOR HEATUP RATES UP
1300	TO 60°F/HR FOR THE SERVICE PERIOD
1200	OF 10°F AND 60 PSIG FOR POSSIBLE .
1100 .	
1000	BASED ON INSERVICE
900	TEMPERATURE (338°F) FOR THE SERVICE
800	PERICO UP TO 16 EFPY
700	HEATUP RATES OF / HR
- 6 00	
	0
500	60
400	
300	
200	Figure 3.1.3a San Onofre Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First
100	16 EFPY
0	100 200 300 400 500 60
	INDICATED TEMPERATURE (°F)

INDICATED PRESSURE (PSIG)

	an an an an Anna an Ann
2500	MATERIAL PROPERTY BASIS
2400	CONTROLLING MATERIAL: INTERMEDIATE SHELL
et ge	COPPER CONTENT: 0.18 WT PHOSPHORUS CONTENT: 0.014 WT
2300	RT _{NDT} INITIAL: 55°F
	RT.NDT AFTER 16 EFPY: 1/4T. 217°F
2200	CURVE APPLICABLE FOR COOLDOWN RATES UP
2100	TO 100°F/HR FOR THE SERVICE PERIOD UP TO
	16 EFPY AND CONTAINS MARGINS OR 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.
2000	
1000	
1900	
1800	
LOUU	
1700	
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1400	
1300	In the formation of the first second state in the second state of the second state is a second state of the second state of
- 1200	
· 1200	
1100	
1000	
900	
200	
800	
800 700	
700	COOLDOWN RATE ^O F/HR
	COOLDOWN RATE ^O F/HR
700	0
700 600 500	COOLDOWN RATE ^O F/HR 0. 20
700 600	0
700 600 500 400	0. 20 40
700 600 500	20
700 600 500 400 300	
700 600 500 400 300 200	20 40 60 Figure 3.1.3b San Crofre Unit No. 1 Reactor Coclard
700 600 500 400 300	20 40 60 Figure 3.1.3b San Crofre Unit No. 1 Reactor Coclard
700 600 500 400 300 200	20 20 40 60 Figure 3.1.3b San Crofre Unit No. 1 Reactor Coclart

INDICATED PRESSURE (PSIG)

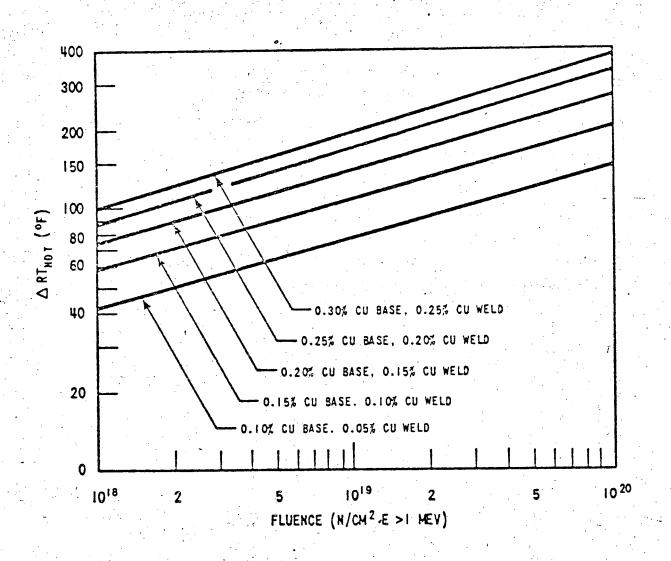


Figure 3.1.3c Effect of Fluence and Copper Content on $\triangle RT_{NDT}$ for Reactor Vessel Steels Exposed to Irradiation at 550°F

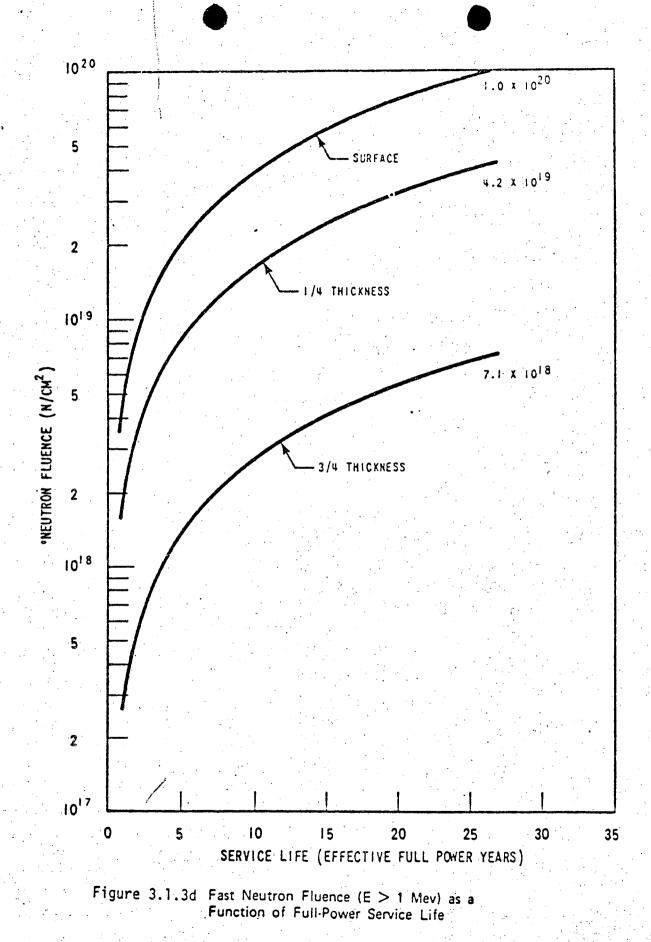


TABLE 3.1.3.1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Materiał Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-1b/35 mil Temp (°F)		RTNDT	Average Upper Shelf Energy (ft-lb)	
						Long.	Trans.	(°F)	Long.	Trans.
Cl. Hd. Dome	W7604	A302B			60 ^[a]	112	132	72	72.5	
Peel Segment	W7605-1	A302B			-10	114	134	74	70.5	
Peel Segment	W7605-2	A302B	cirrent		-10	90	110	50	122	
Peel Segment	W7605-3	A302B	·	-	-10	108	128	68	85	
Peel Segment	W7605-4	A302B	·		-10	120	140	80	74	
Peel Segment	W7605-5	A302B			-10	26	48	-10	109	
Peel Segment	W7605-8	A302B		,	-10	102	122	62	88	
Hd. Flange	W7602	A336 mod			60 ^[a]	[b]		60		
Ves. Flange	W7603	A336 mod		:	60 ^[a]	[b]		60		
Iniet Nozzie	W7811-1	A336 mod	-		60 ^[a]	(b)		60		
Inlet Nozzle	W7611-2	A338 mod	·		60 ^[n]	[b]		60		
Inlet Nozzie	W7611-3	A336 mod		ş. 	60 ^[a] .	[b]	·	60	:	
Outlet Nozzie	W7610-1	A338 mod			60 ^[a]	[b]		60		
Outlet Nozzie	W7610-2	A338 mod			- 80 ^[a]	(b)		60		
Outlet Nozzle	W7810-3	A336 mod			60 ^[a]	[b]		60	<u> </u>	i internet i i
Upper Shell	W7601-3	A302B	0.15	0.014	-10	48	68	8	98.5	
Upper Shell	W7601-6	A302B	0.16	0.012	-30	64	84	24	104	
Upper Shell	W7601-7	A302B	0.15	0.014	-20	52	72	12	95.5	

a. Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2.

b. Only 10°F Charpy V-notch data available. Conservative astimates for NDTT and RTNDT were used.

TABLE 3.1.3.1 (CONT.)

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Meterial Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/36 mil Temp (°F)		RTNDT	Average Upper Shelf Energy (ft-lb)	
						Long.	Trans.	(°F)	Long.	Trane.
Inter. Shell	W7601-1	A302B	0.17	0,013	0	57	120 ^[a]	60	94	76
Inter. Shell	W7601-8	A302B	0.18	0.012	10	93	100 ^[a]	.40	97	79
Inter. Shell	W7601-9	A302B	0.18	0.014	0	84	115[a]	55	102	72
Lower Shell	W7601-2	A302B	0.17	0.013	-20	74	94	34	97	· · · ·
Lower Shell	W7601-4	A302B	0.14	0.014	-10	91	111	51	-94	
Lower Shell	W7601-5	A302B	0.14	0.014	10	122	142	82	87.5	
Bot. Hd. Peel	W7607	A302B			-20	62	82	22	91	
Bot. Hd. Dome	W7606	A302B		·	60 ^[b]	99	119	60	86 🔍	
Weld			0.19	0.017	0 ^[b]		29 ^[a]	0		90
HAZ				·	0 ^(b)		-14 ^[a]	0		101

a. Actual not estimated

b. Estimated per NRC Standard Review Plan Branch Tachnical Position MTEB 5-2.