3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-2, as applicable~~, for hydrostatic or leak testing, heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS, and operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-2, as applicable,~~ at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-2, as applicable,~~ within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

and embrittlement.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence ~~as~~ a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H. ~~in accordance with the schedule in Table 4.4.6.1.3-1.~~ The results of these fluence determinations shall be used to update the curves ~~A and A'~~ of Figure~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-2.~~

and embrittlement

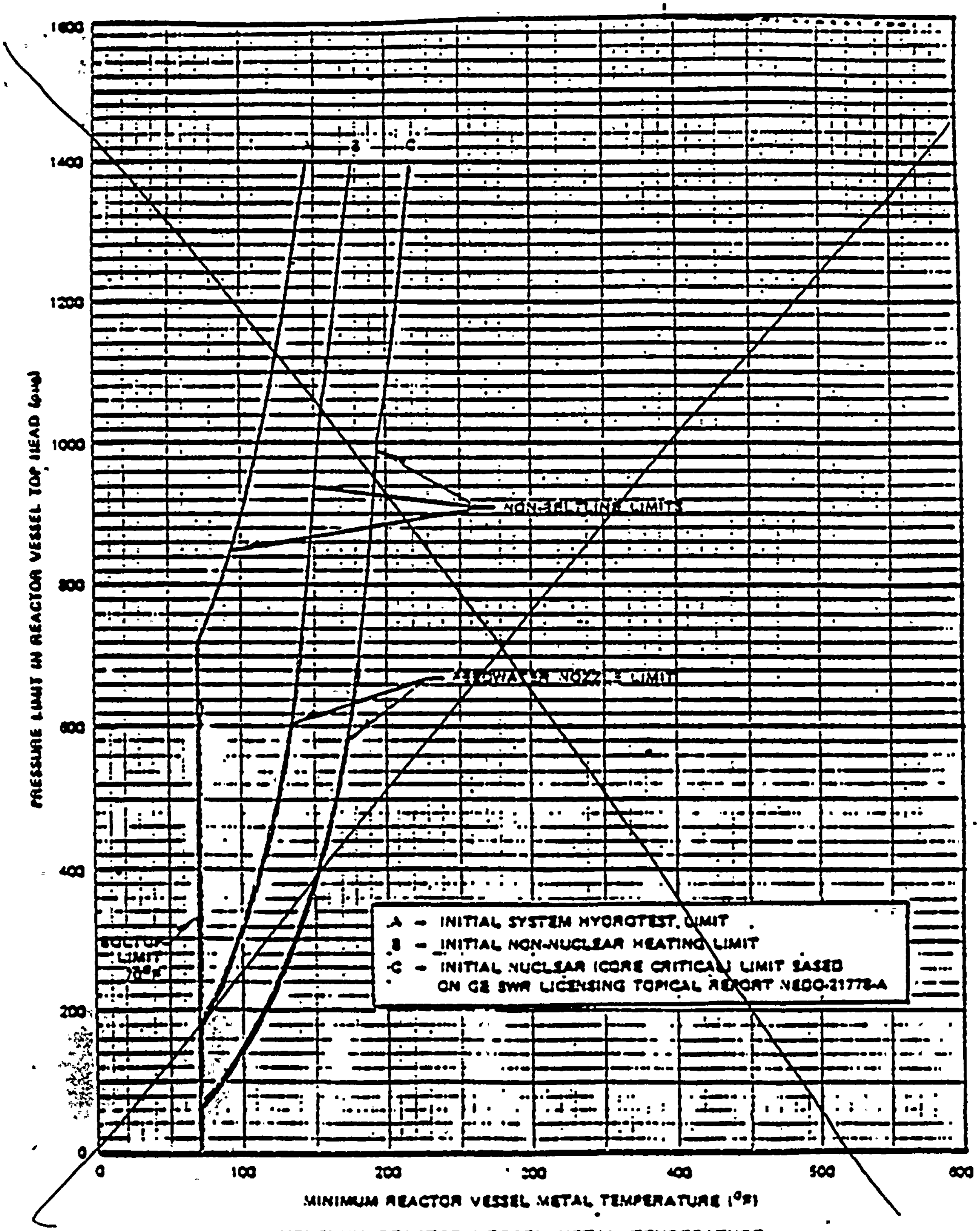
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

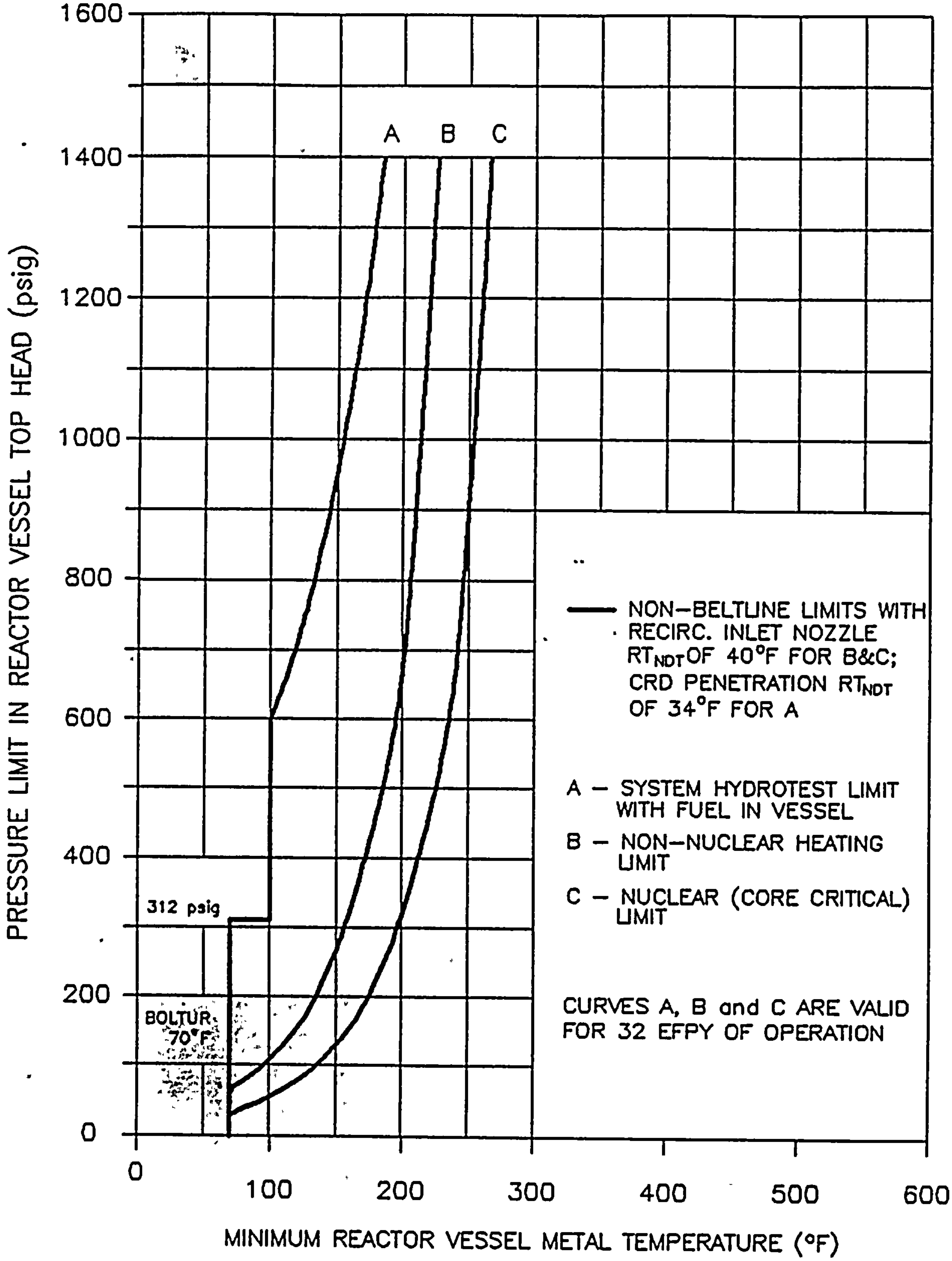
   2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



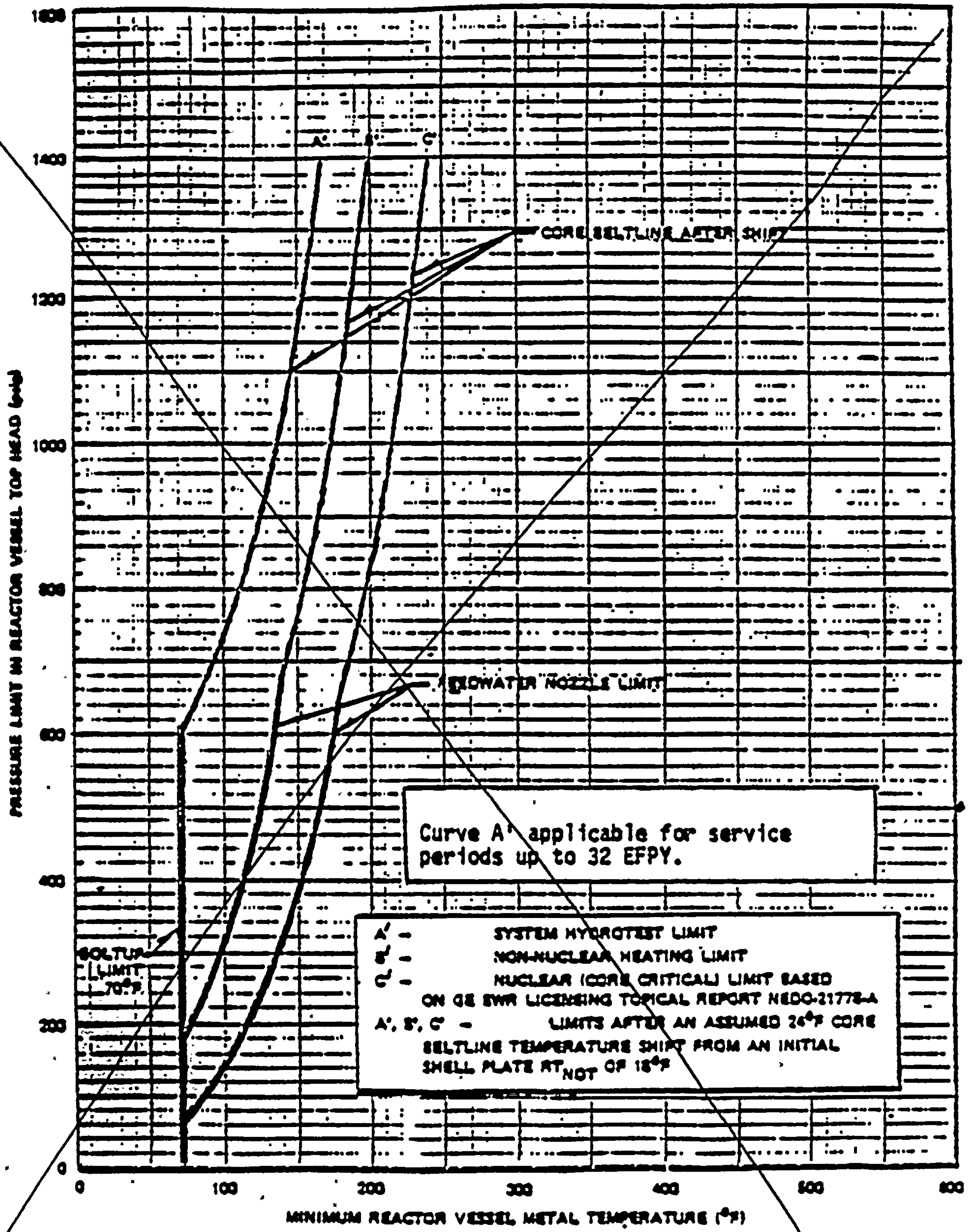
Insert
New Figure

MINIMUM REACTOR VESSEL METAL TEMPERATURE
VS. REACTOR VESSEL PRESSURE
Figure 3.4.6.1-1



REACTOR VESSEL PRESSURE VS. MINIMUM VESSEL TEMPERATURE FOR UNIT 1

Figure 3.4.6.1-1



MINIMUM REACTOR VESSEL METAL TEMPERATURE
VS. REACTOR VESSEL PRESSURE
Figure 3.4.6.1-2

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

SPECIMEN HOLDER	VESSEL LOCATION	LEAD FACTOR*	WITHDRAWAL TIME (EFPY)
131C7717G1	300°	~~1.20~~ 0.936	6
131C7717G2	120°	~~1.20~~ 0.936	15
131C7717G3	30°	~~1.20~~ 0.936	Spare

*At ~~1/4 T~~ I. D. Surface

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistant with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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 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 reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, ~~phosphorus~~ Nickel content and copper content of the material in question, can be predicted using ~~Bases Figure B 3/4.4.6-1 and the recommendations of~~ Regulatory Guide 1.99, Revision ~~1~~ 2, "~~Effects of Residual Elements on Predicted Radiation Damage to~~ Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1~~, curves A', B' and C'~~, includes predicted adjustments for this shift in RT_{NDT} for the ~~end of life fluence.~~ 32 EFPY condition.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting

PRESSURE/TEMPERATURE LIMITS (Continued)

reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision ~~1~~ 2.

The pressure-temperature limit lines shown in Figure~~s~~ 3.4.6.1-1 ~~and 3.4.6.1-2~~, curves C~~, and C'~~, and A ~~and A'~~, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through 1972.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

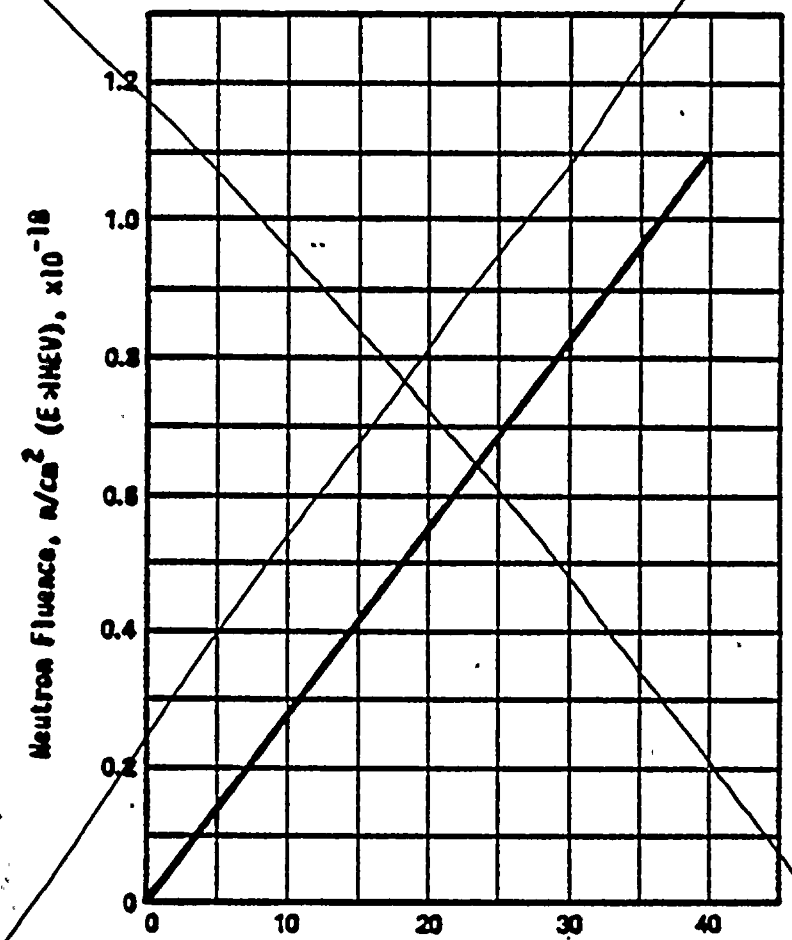
REACTOR VESSEL TOUGHNESS

LIMITING BELTLINE COMPONENT.	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	*Ni* ~~N~~(%)	~~HIGHEST~~ STARTING RT_{NDT} (°F)	ΔRT_{NDT}(°F) ~~MAX.~~ ~~*~~ *	MIN. UPPER SHELF (LFT-LBS)	MAX. RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	C2433-1	0.10	~~0.009~~ *0.63*	+18	~~24~~ *40*	N/A	~~42~~ [illegible]
Weld	N/A	~~1311A27A-1~~ *629616/ L320A27AG*	0.04	~~0.016~~ *0.99*	-50	~~30~~ *33*	N/A	~~20~~ *-17*

NOTE:* These values are given only for the benefit of calculating the ~~end-of-life (EOL)~~ *32 EFPY* RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA-533 GR B CL.1	C1232-2	+20
Bottom Head Dome	"	C9942-2	+34
Bottom Head Torus	"	C9942-2	+34
Top Head Dome	"	C9220-2	+10
Top Head Torus	"	C9355-1	+10
Top Head Flange	SA-508, CL.2	N/A	+10
Vessel Flange	"	N/A	+10
Feedwater Nozzle	"	Q2Q49W	-16
Recirculation Inlet Nozzle	*"*	*Q2Q49W*	*+40*
Weld	Per GE Purch. Spec.	No CMVS Available	0 (Per Purch. Spec. Requirements
Closure Studs	SA-540 GR B24	82552	+70

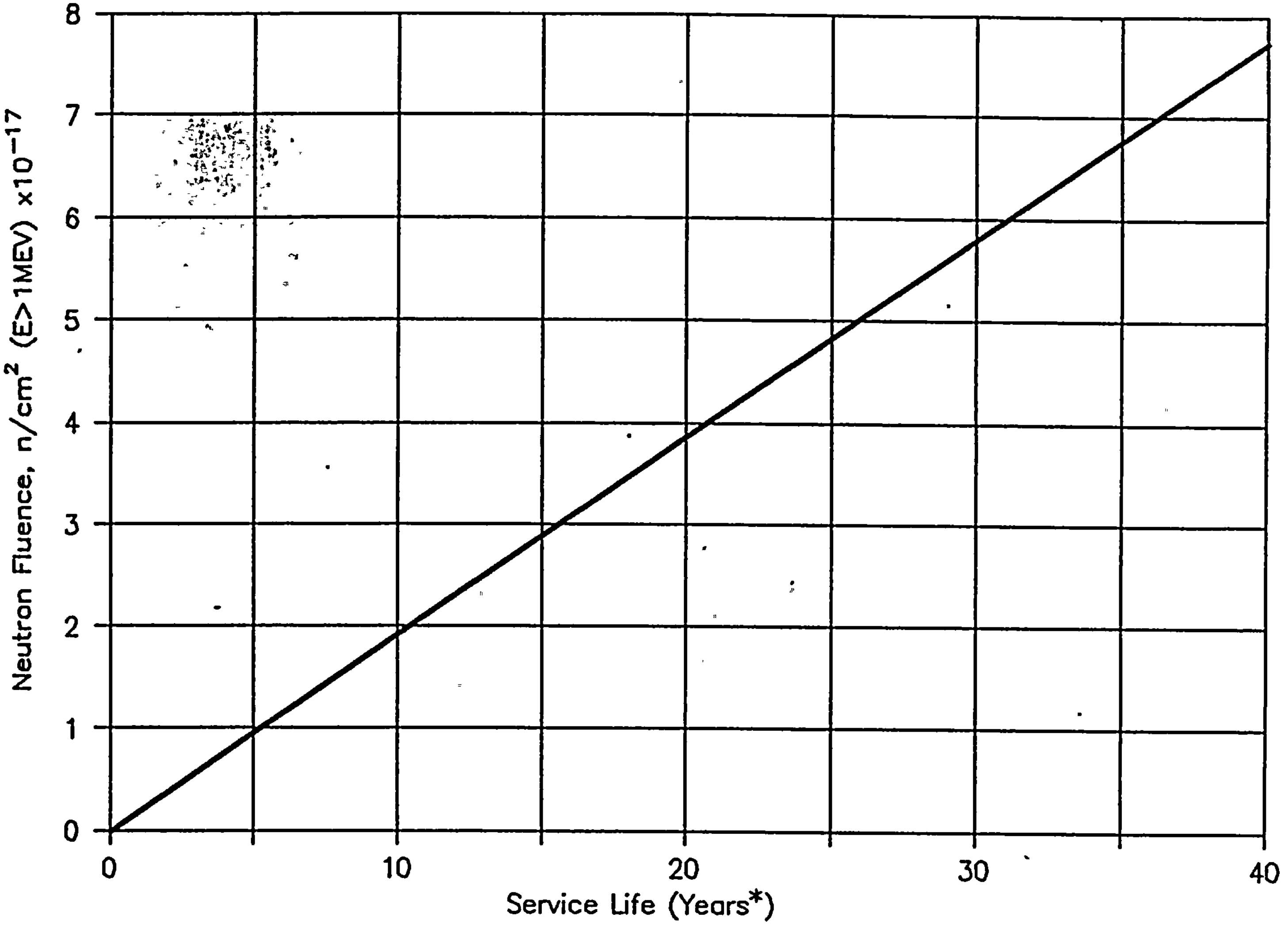
Insert New Figure



Fast Neutron Fluence (E>1 Mev) at ¼T As a Function of Service Life*

Bases Figure B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% availability



Fast Neutron Fluence (E>1 Mev) at I.D. Surface as a Function of Service Life*

Bases Figure B 3/4.4.6-1

*At 90% of RATED THERMAL POWER and 90% availability

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 for hydrostatic or leak testing, heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS, and operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence and embrittlement as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H. ~~in accordance with the schedule in Table 4.4.6.1.3-1.~~ The results of these fluence and embrittlement determinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

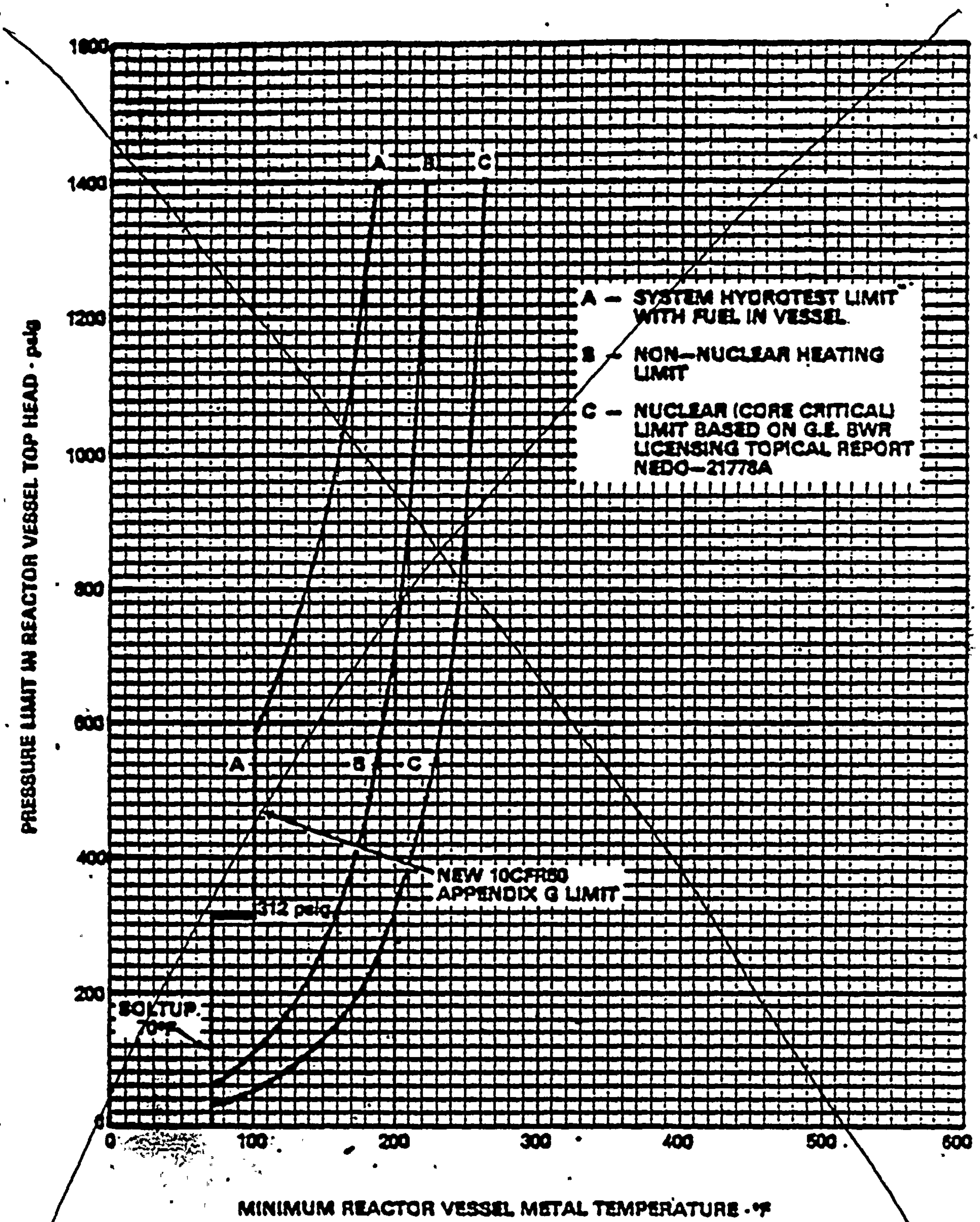
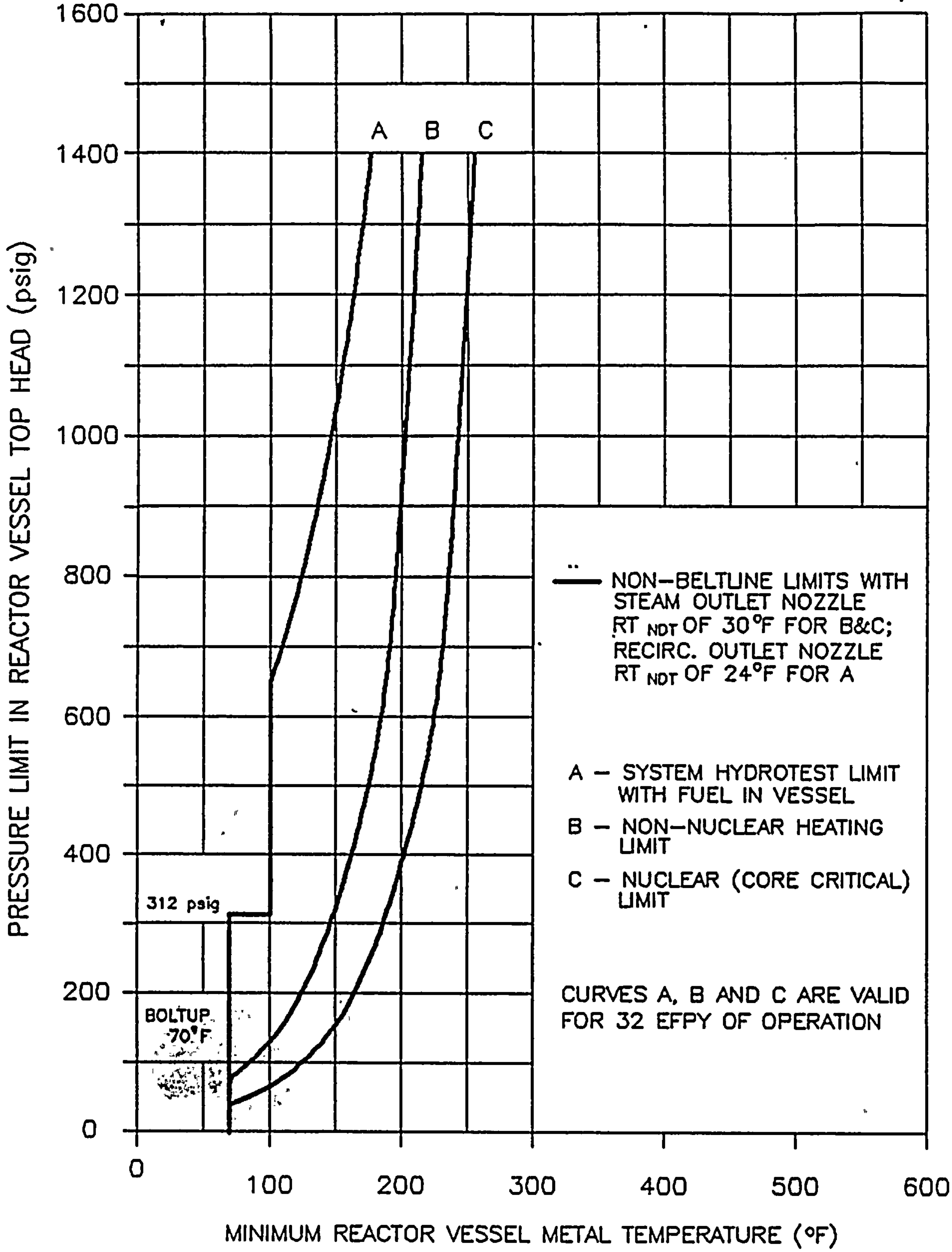


FIGURE 3.4.6.1-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Insert New Figure



REACTOR VESSEL PRESSURE VS. MINIMUM VESSEL TEMPERATURE FOR UNIT 2

Figure 3.4.6.1-1

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

SPECIMEN HOLDER	VESSEL LOCATION	LEAD FACTOR*	WITHDRAWAL TIME (EFPY)
131C7717G1	300°	~~1.20~~ 0.936	6
131C7717G2	120°	~~1.20~~ 0.936	15
131C7717G3	30°	~~1.20~~ 0.936	Spare

*At ~~1/4 T~~. I. D. Surface

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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 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 reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel ~~phosphorus~~ content and copper content of the material in question, can be predicted using ~~Bases Figure B 3/4.4.6-1 and the recommendations of~~ Regulatory Guide 1.99, Revision ~~1~~ 2 "~~Effects of Residual Elements on Predicted~~ Embrittlement of Radiation ~~Damage to~~ Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1 includes predicted adjustments for this shift in RT_{NDT} for the ~~end of life fluence.~~ 32 EFPY condition.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision ~~1~~ 2.

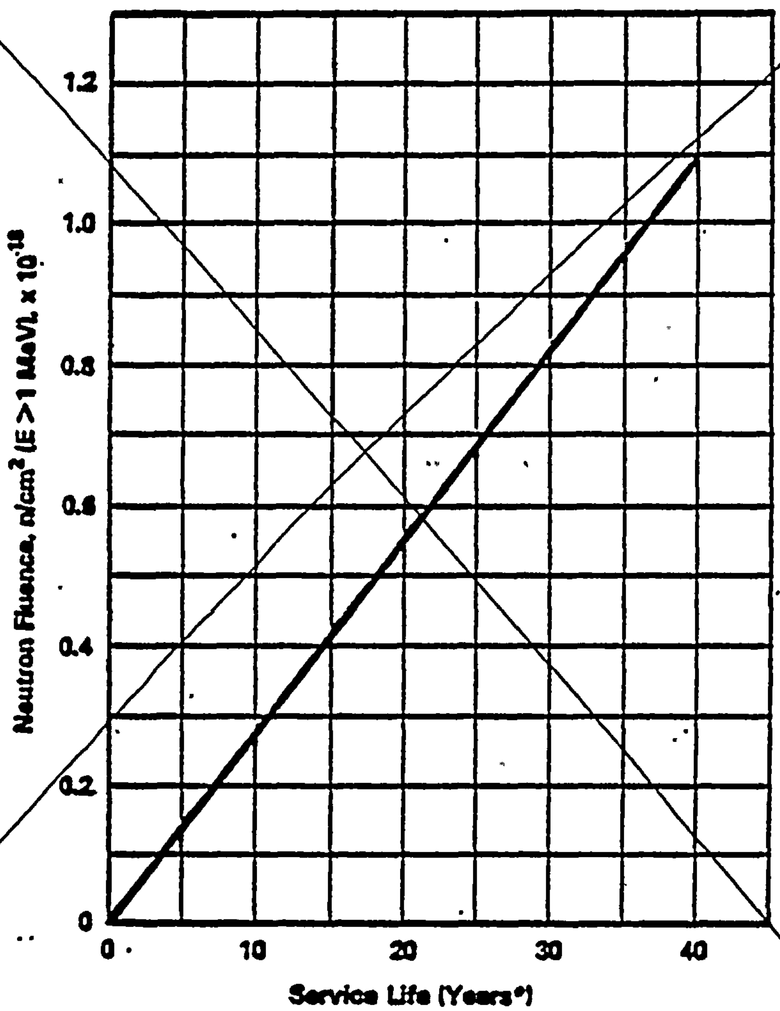
BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

LIMITING BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni ~~X~~(%)	~~HIGHEST~~ STARTING RT_{NDT} (°F)	Δ ~~MAX:~~ RT_{NDT}(°F) *	MIN. UPPER SHELF (LFT-LBS)	MAX. RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	6C1053/1	0.10	~~0.012~~ 0.58	+10	~~27~~ 40	N/A	~~+37~~ +50
Weld	N/A	624263/ E204A27A	0.06	~~0.010~~ 0.89	-20	~~17~~ 50	N/A	~~-3~~ +3[illegible]

NOTE:* These values are given only for the benefit of calculating the ~~end-of-life (EOL) RT_{NDT}~~ 32 EFPY RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring #5	SA-533 GR B Cl.1	All	+10
Bottom Head Dome	"	C0462	+20
Bottom Head Torus	"	C0472	+10
Top Head Side Plates	"	C0473-1	+10
Top Head Flange	SA-508, Cl.2	125H446	+10
Vessel Flange	"	2L2393	+10
Feedwater Nozzle	"	Q2Q62W	-10
Steam outlet Nozzle	"	Q2Q64W	+30
Weld	Bottom Head	All	-20
	Flanges to Shell Top Head	All	-20
	Other Non-Beltline	All	0
Closure Studs	SA-540 GR B24	All	Meet requirements of 45 ft-lbs and 25 mils lateral expansion at +10°F.

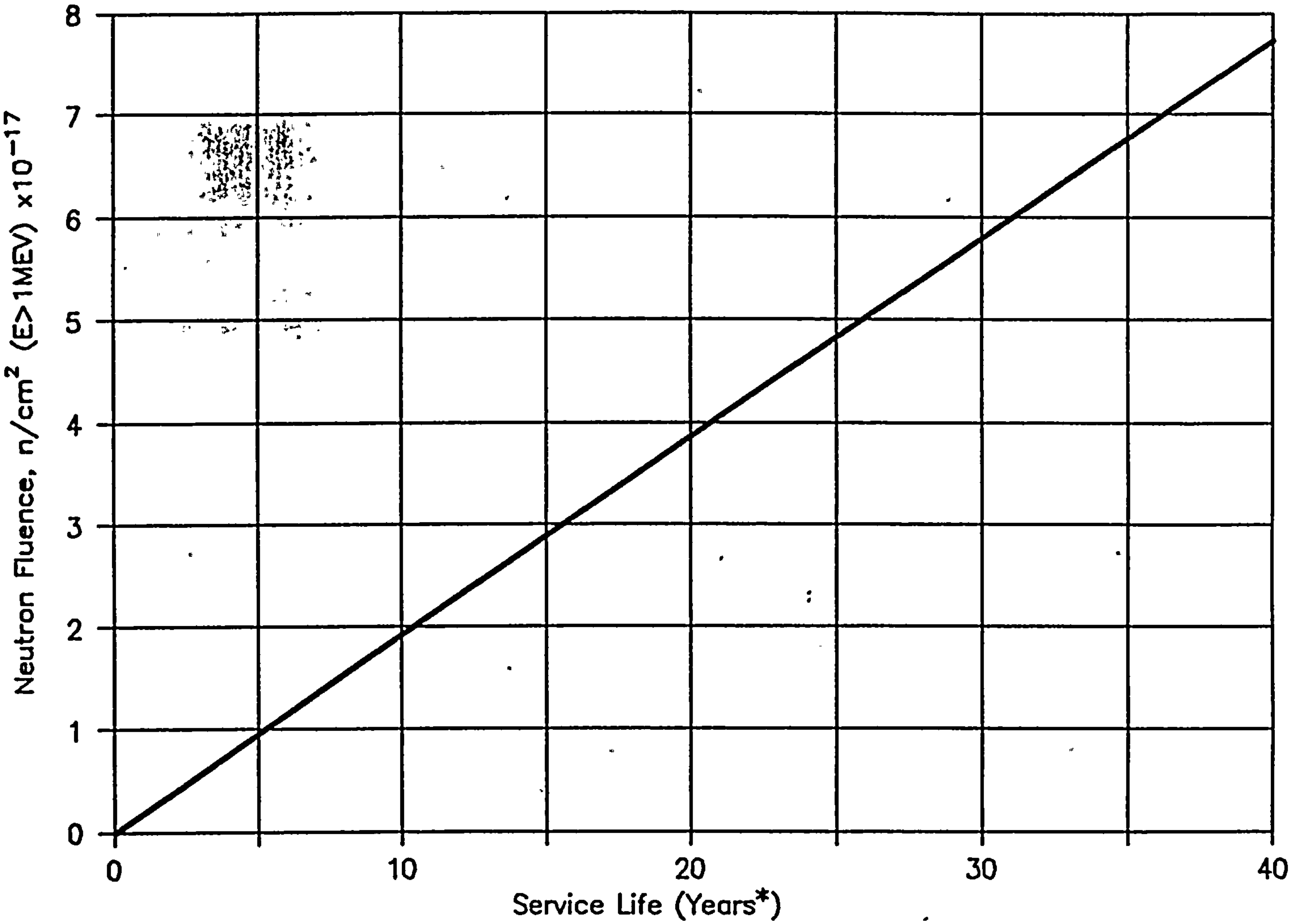


Insert New Figure

BASES FIGURE B 3/4.4.6-1

FAST NEUTRON FLUENCE (E>1 MeV) AT ¼T AS A FUNCTION OF SERVICE LIFE*

*At 90% of RATED THERMAL POWER and 90% availability.



Fast Neutron Fluence (E>1 Mev) at I.D. Surface as a Function of Service Life*

Bases Figure B 3/4.4.6-1

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*At 90% of RATED THERMAL POWER and 90% availability