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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system pressure and reactor vessel metal temperature shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum reactor coolant heatup of 100°F in any one hour period,

b. A maximum reactor coolant cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 10°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 the reactor coolant system pressure and reactor vessel metal temperature shall be determined to be to the right of the limit lines of Figure 3.4.6.1-1 curves A or B ~~and B~~, as applicable, at least once per 30 minutes.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system pressure and reactor vessel metal temperature shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve~~s~~ C ~~and C'~~ 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 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.

4.4.6.1.4 The reactor vessel material specimens shall be removed and examined 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.

4.4.6.1.5 ~~The reactor flux wire specimens shall be removed at the first refueling outage and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4.4.6-1. The results of the fluence determinations, in conjunction with Figure B 3/4.4.6-1, shall be used to adjust the curves of Figure 3.4.6.1-1, as required.~~

replace with:

The Pressure-Temperature limit Figure 3.4.6.1-1 is valid through 10 EFPYs and shall be re-evaluated prior to exceeding 10EFPYs.

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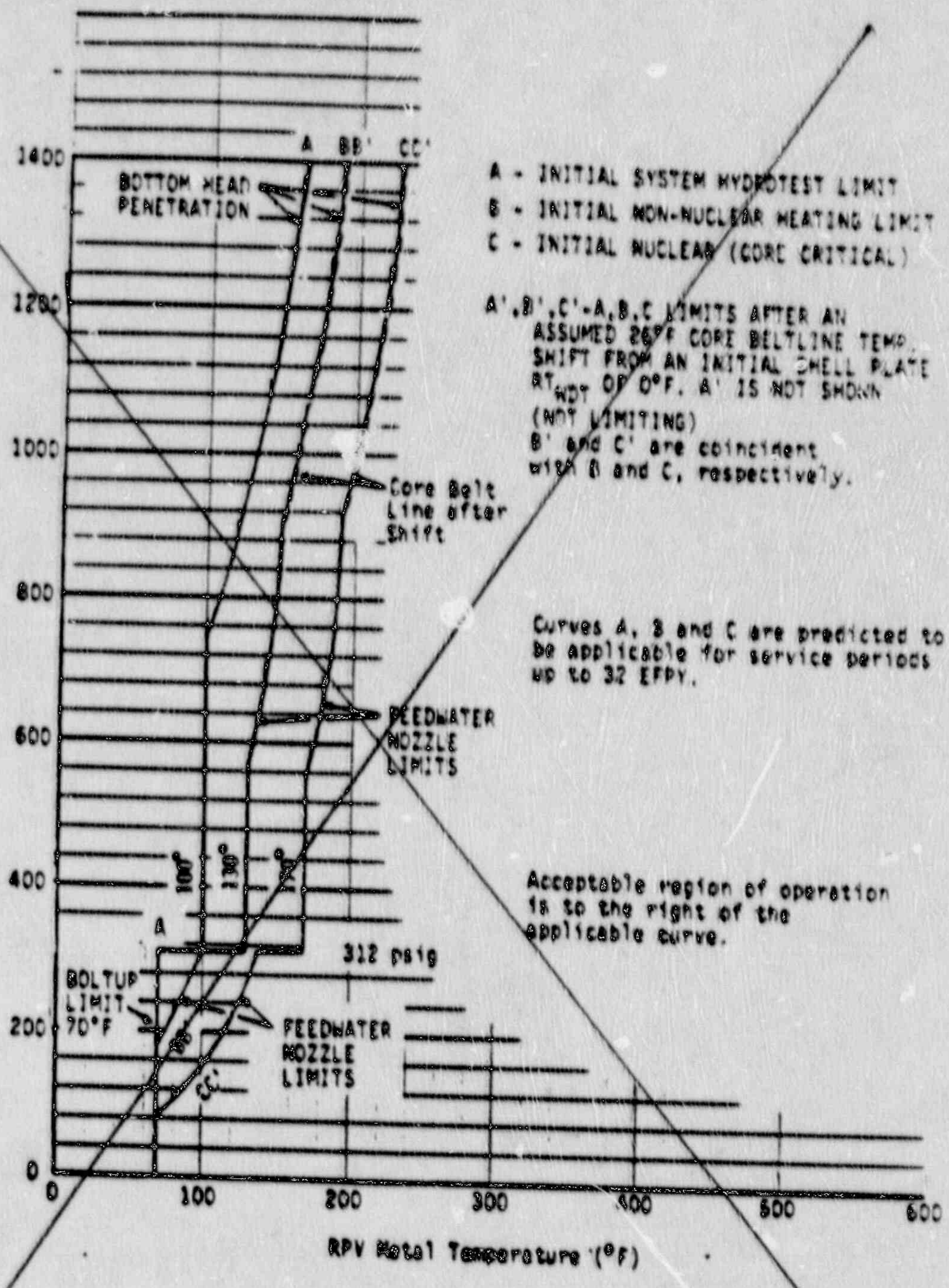
Delete this figure.
Replace with revised Figure 3.4.6.1-1



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

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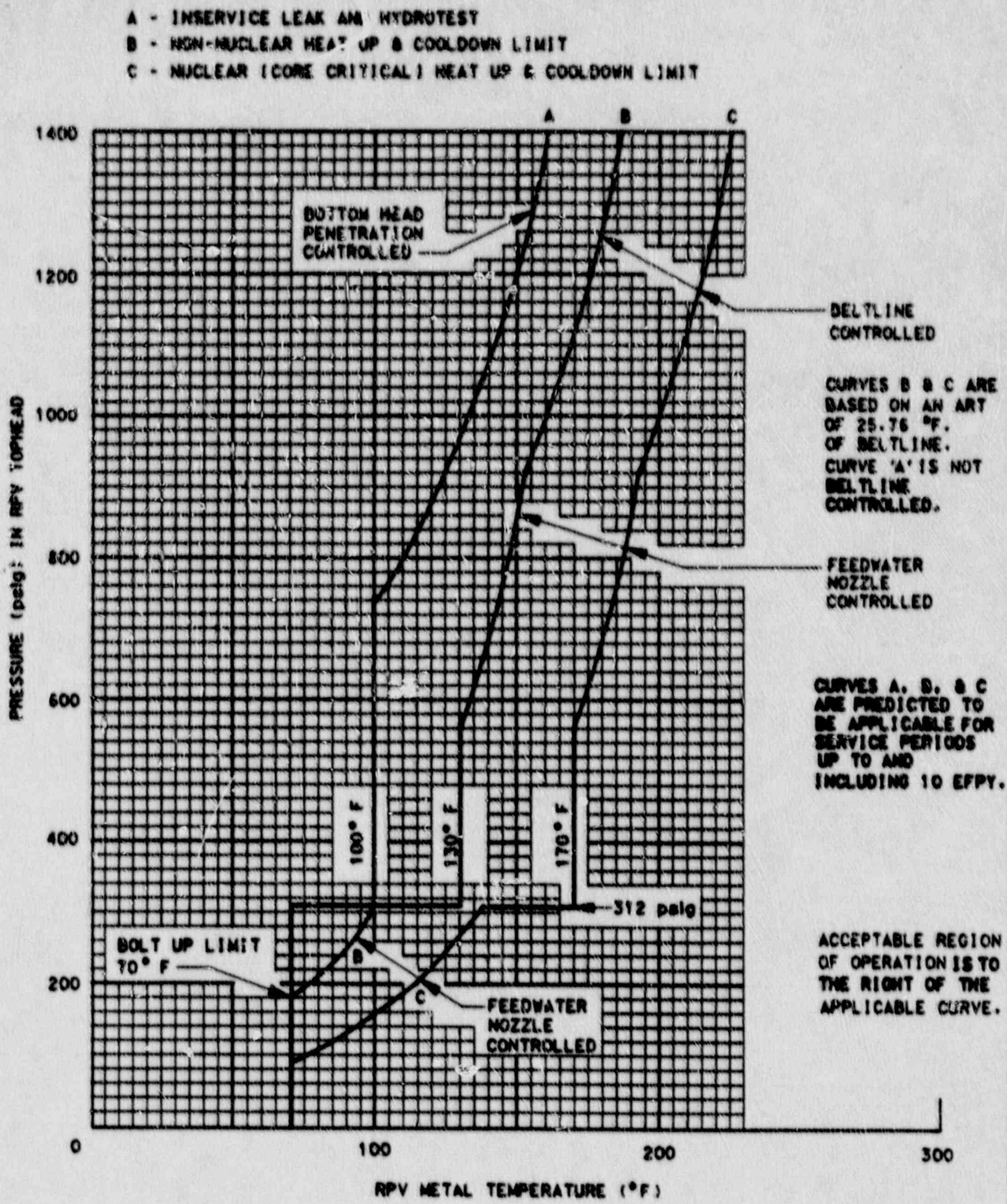

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

FIGURE 3.4.6.1-1

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REACTOR COOLANT SYSTEM

BASES

3/4.4.5 SPECIFIC ACTIVITY (Continued)

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

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 RT_{NDT} for welds and base material in the closure flange region is $\leq 10°F$. The initial hydrostatic test pressure was 1563 psig. 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 Embrittlement of ~~Damage to~~ 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~~. ~~Curves B' and C' are coincident with curves B and C, respectively.~~

Adjusted Reference Temperature (ART)

revised based on results from the wire capsule analysis performed after irradiation for one cycle.

10 Effective Full Power Years (EFPY) of exposure.

Bases Figure B 3/4.4.6-1 has been revised to reflect the analysis of the flux wire dosimeter which was removed during the first refueling outage. The upper bound curve B 3/4.4.6-1 was used in determining the Pressure-Temperature limit curves in Figure 3.4.6.1-1.

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PRESSURE/TEMPERATURE LIMITS (Continued)

for higher Service Periods

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 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 Figures 3.4.6.1-1, curves C, ~~B~~, 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, 1977 Edition, and Addenda through Summer 1978.

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

Beltline Component	Weld Seam I.D. or Material Type	Heat No.-Slab No. or Heat No./Lot No.	Cu %	Ni ~~*~~ (%)	Starting RT_{NDT} (°F)	* Maximum ~~**~~ ΔRT_{NDT} (°F)	Minimum Upper Shelf (ft-lb)	**Maximum ~~EOL RT_{NDT} (°F)~~ ART (°F) at 10EFPY
Plate	SA-533 Gr.B, CL.1 SA-533 Gr.B, CL.1	C2594-2	0.04	~~0.012~~ 0.63	0	~~+26~~ 8.84	96 (C2594-2)	~~+26*~~ 17.7
Weld	#2 Shell Long. Seams	627260/B322A27AE	0.06	~~0.020~~ 1.08	-30	~~44~~ 27.88	~~N/A~~ 88	~~+14~~ 25.76 (limiting)

Non-Beltline Component	Material Type or Weld Seam I.D.	Heat No.-Slab No. or Heat No./Lot No.	Highest Starting RT_{NDT} (°F)
Shell Ring	SA-533 Gr.B, CL.1	C2815-2, C2779-2, C2779-1, C2780-2, C2780-1, C2741-1	+10
Bottom Head Dollar Plate	SA-533 Gr.B, CL.1	A1113-1 C2630-2	0
Bottom Head Radial Plates	SA-533 Gr.B, CL.1	C2539-2, A1145-1	+10
Top Head Dollar Plate	SA-533 Gr.B, CL.1	C2440-3	-30
Top Head Side Plates	SA-533 Gr.B, CL.1	C2944-1	+10
Top Head Flange	SA-508 CL.2	4801682	-30
Vessel Flange	SA-508 CL.2	4801141	-30
Feedwater Nozzle	SA-508 CL.2	Forging No. 249n-1, 2, 3, 4, 5, & 6, Q2Q65W	-20
Weld	N/A	N/A	-20***
Closure Stud	SA-540 Gr.B24	84025, 84299	+10

~~*Combination of the highest starting RT_{NDT} plate and the highest ΔRT_{NDT} plate.~~

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

***Based on purchase spec. requirements.

* ΔRT_{NDT} computed based on Regulatory Guide 1.99 Revision 2

** ΔART for 10EFPY calculated based on Regulatory Guide 1.99 Revision 2

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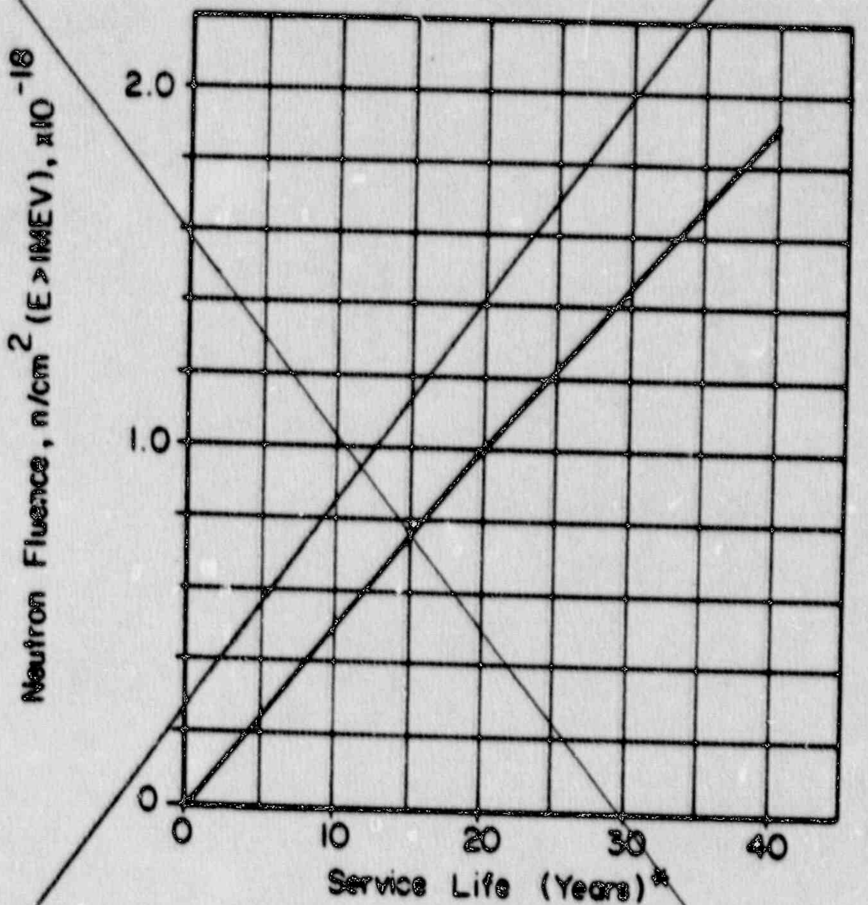
Replace with revised
bases Figure B 3/4 4.6-1



Fast Neutron Fluence (E>1 MeV) at 1/4 T As a Function of Service Life *

BASES FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 MeV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

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

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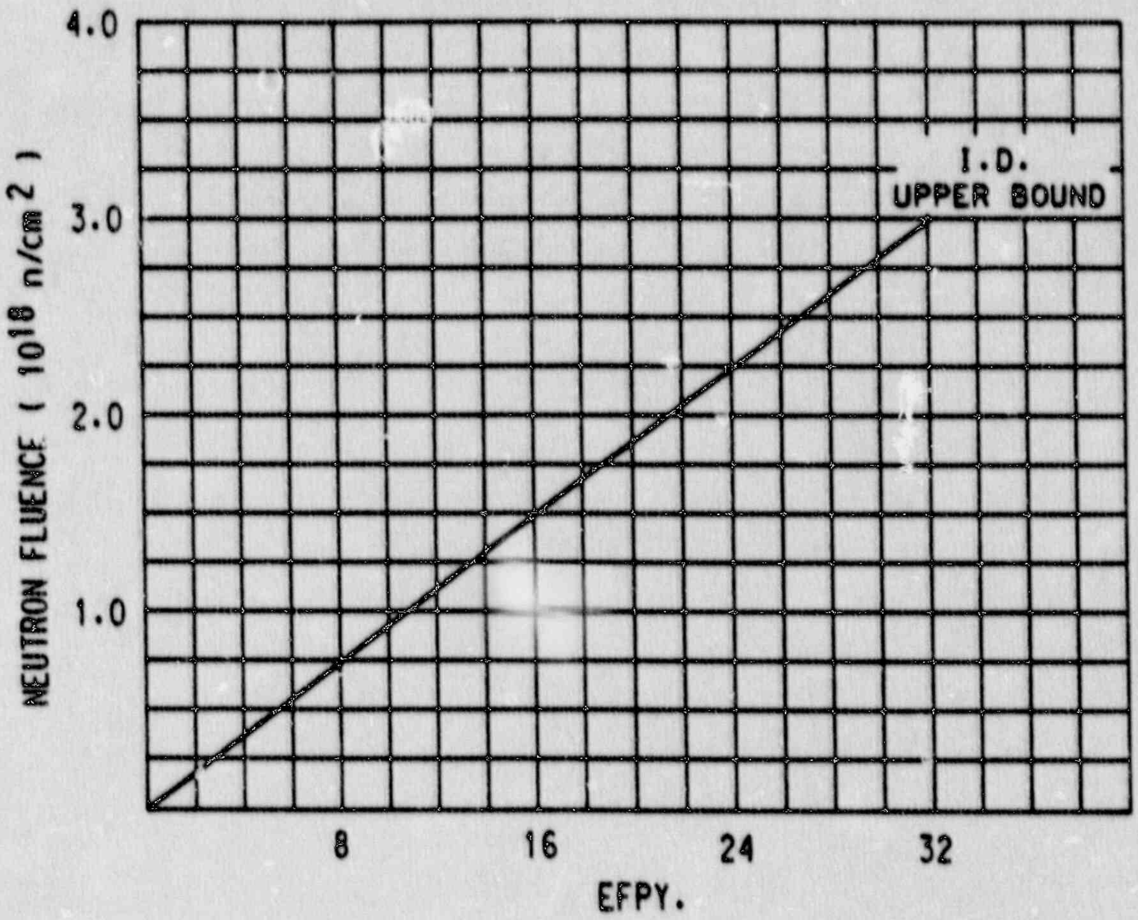

BASES FIGURE B 3/4.4.6.1 FAST NEUTRON FLUENCE (E>1MeV) AT VESSEL I.D. AS A FUNCTION OF EFPY.