ENCLOSURE 1

Marked-up Pages of RBS Tech. Specs. for Requested Changes

Replace with revised
Figure 3.4.6.1-1

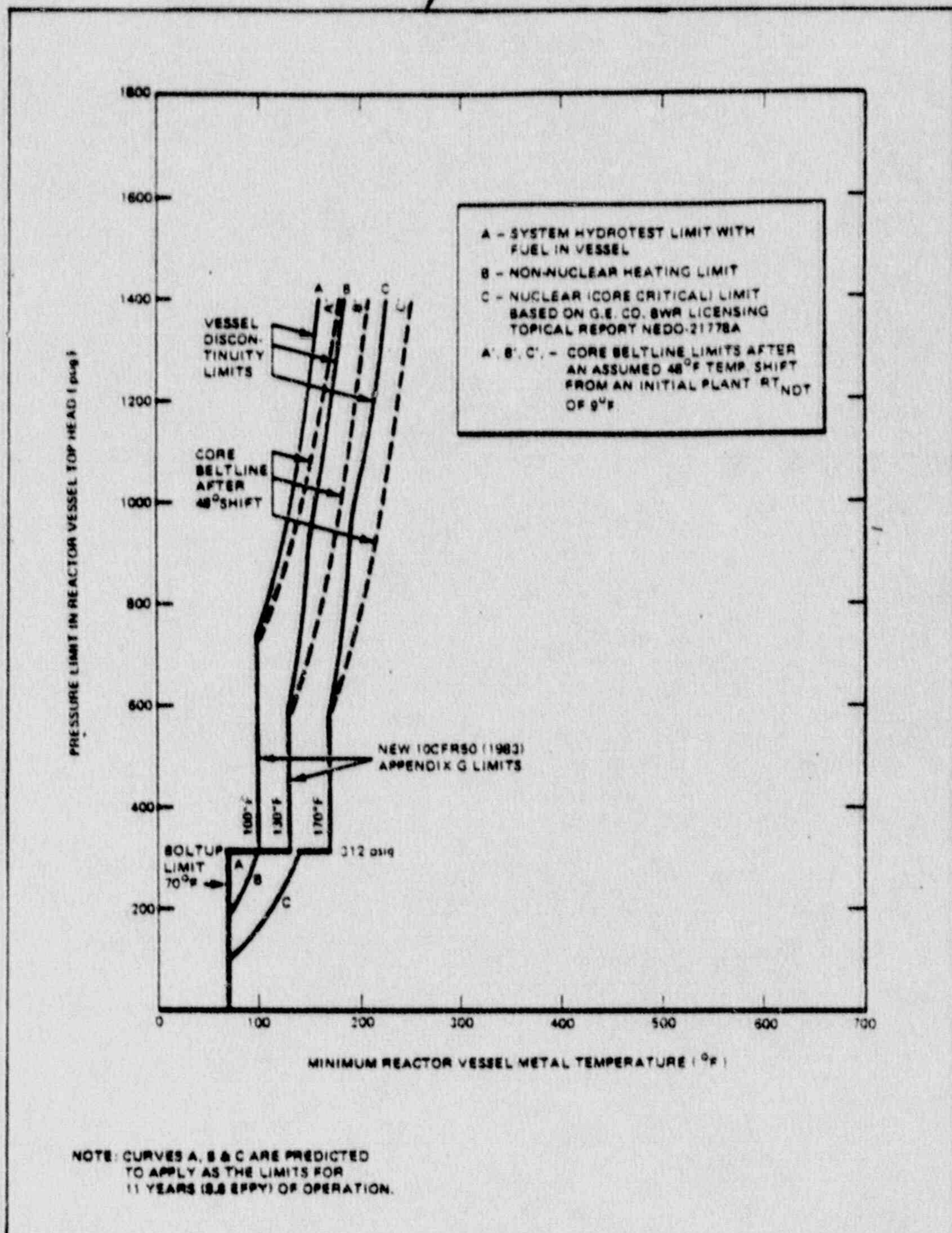


FIGURE 3.4.6.1-1
MINIMUM TEMPERATURE REQUIRED VS REACTOR PRESSURE

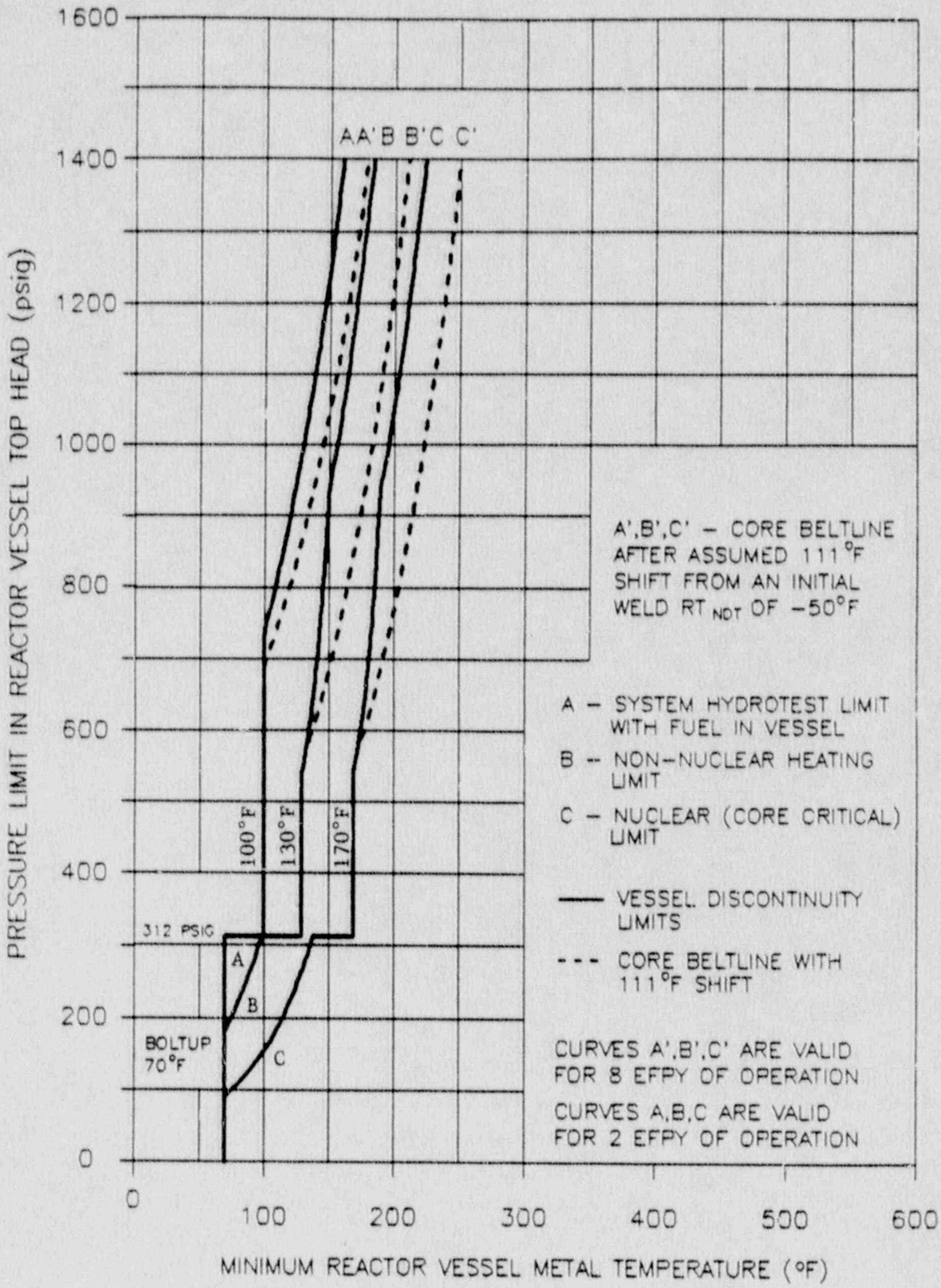


FIGURE 3.4.6.1-1
MINIMUM TEMPERATURE REQUIRED VS REACTOR PRESSURE

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	~~LEAD FACTOR @ ¼T~~	Lead Factor At I.D./¼T	WITHDRAWAL TIME (EFPY)
1	3°	~~0.86~~	0.67/0.89	6
2	177°	~~0.86~~	0.67/0.89	15
3	183°	~~0.86~~	0.67/0.89	Standby

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 thermally 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 thermally 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. Consequently, 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 2, "~~Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials~~ Radiation 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, as well as adjustments for possible errors in the pressure and temperature sensing instruments.~~ conditions at 8 EFPY.

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be determined periodically during operation by removing and evaluating, in accordance with ASTM E185 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 Figure 3.4.6.1-1, 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. However, single failure considerations require that two valves be OPERABLE. 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. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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).

BASES TABLE B 3/4.4.6-1

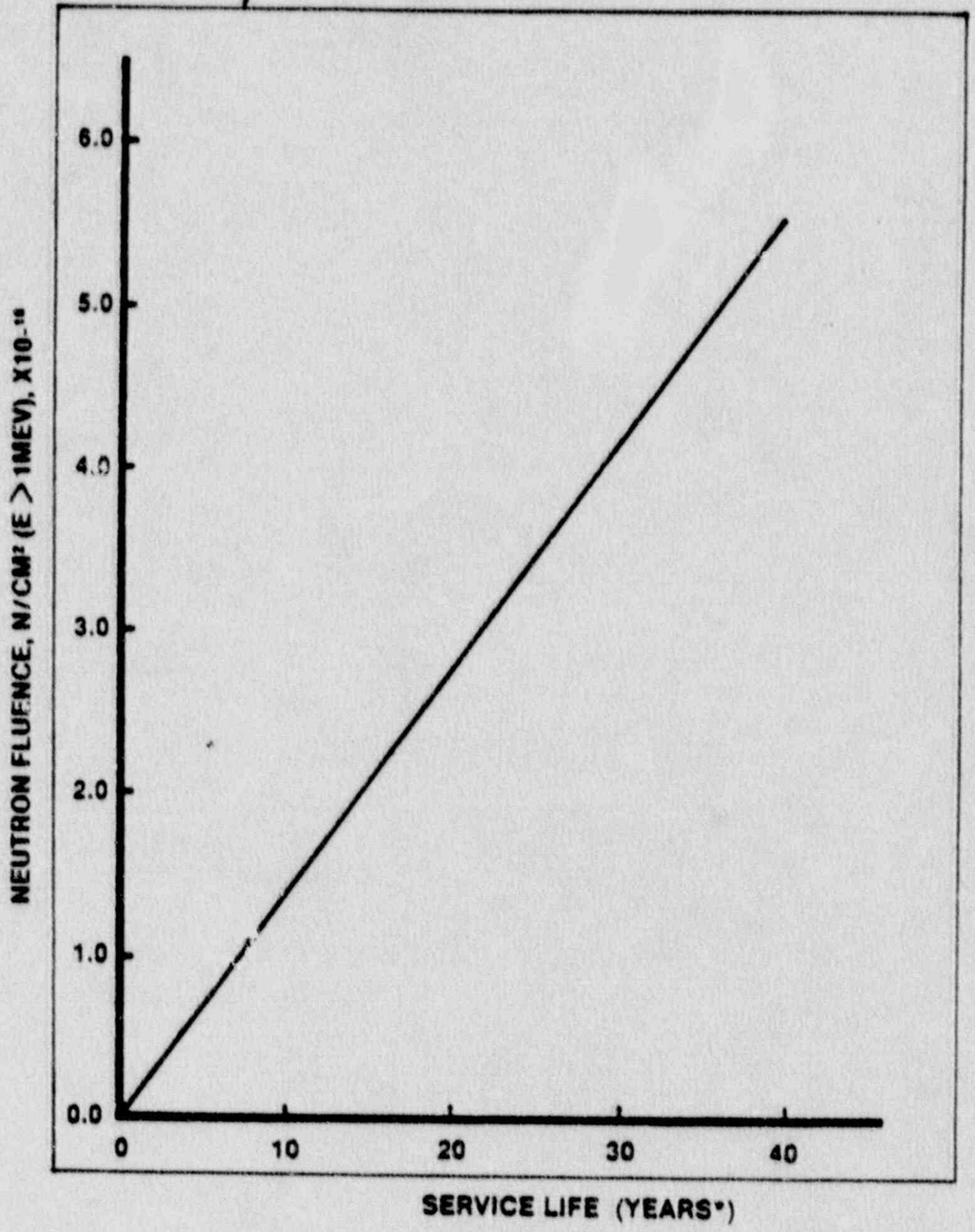
REACTOR VESSEL TOUGHNESS

Limiting BELTLINE COMPONENT	WELD SEAM OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni ~~Ø~~(%)	~~HIGHEST~~ STARTING RT_{NDT}(°F)	ΔRT_{NDT} ~~MAX.~~ * (°F)	AVG. UPPER SHELF (FT-LBS)	Maximum ~~MAX. EOL~~ RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	C3138-2	0.08	~~0.012~~ 0.63	+9	75 ~~48~~	79	84 ~~+57~~
Weld	SHELL COURSE No.2 Vertical Seam 3	~~492L4871/ A421B27AF~~ 5P6756/ Lot 0342	~~0.03~~ 0.09	~~0.020~~ 0.92	-50	153 ~~80~~	97 ~~130~~	103 ~~+30~~

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

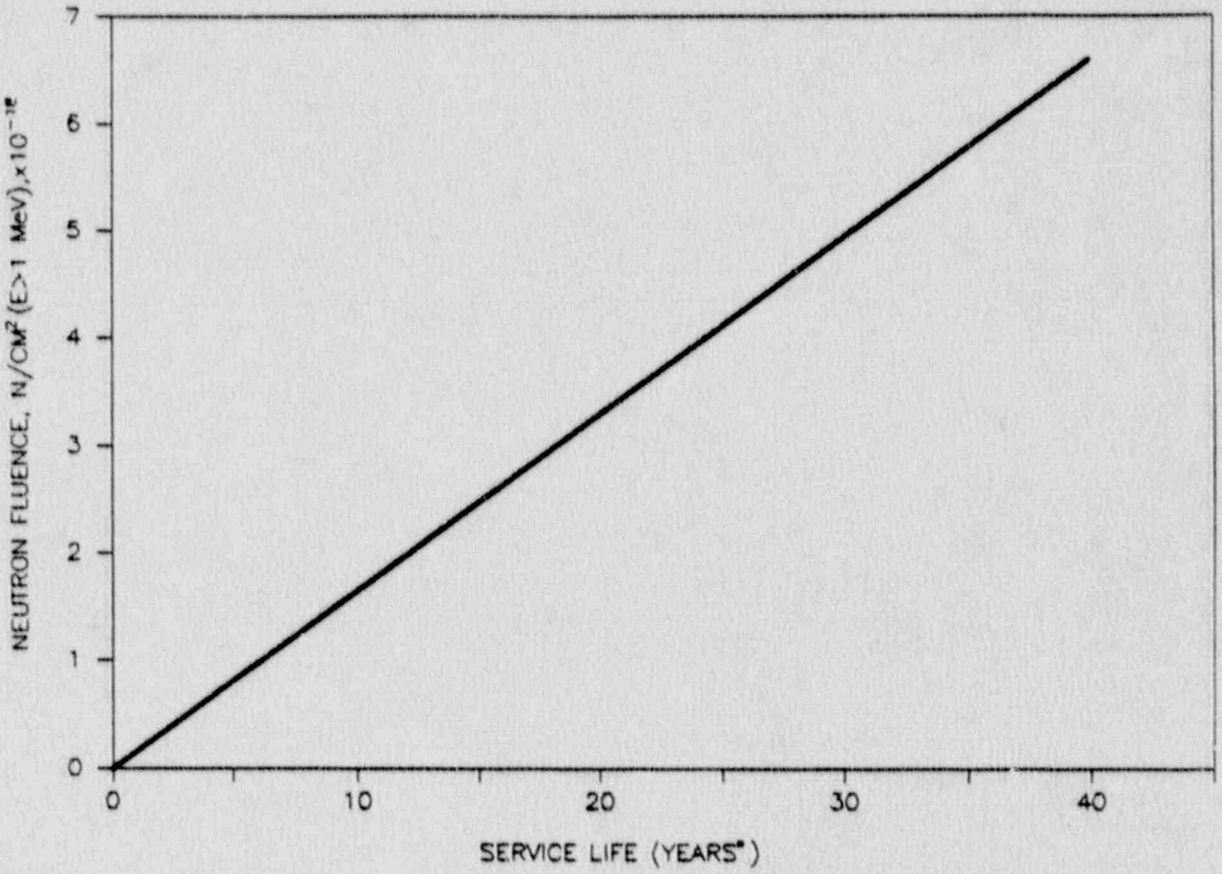
NON-BELTLINE COMPONENT	MT'L TYPE	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT}(°F)
Shell Ring	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Flange	SA 508 Cl.2	ALL HEATS	+10
Vessel Flange	SA 508 Cl.2	ALL HEATS	+10
Feedwater Nozzle	SA 508 Cl.2	ALL HEATS	-20
Weld	LOW ALLOY STEEL	ALL HEATS	-20
Closure Studs	SA 540 GRADE B23 or B24	ALL HEATS	Meets requirement of 45 ft-lbs and 25 mils lateral expansion at + 10°F

Replace with revised
Figure B 3/4.4.6-1



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



BASES FIGURE B 3/4.4.6-1
FAST NEUTRON FLUENCE (E>1 MEV) AT VESSEL I.D.
AS A FUNCTION OF SERVICE LIFE*

* AT 90% OF RATED THERMAL POWER AND 90% AVAILABILITY

Attachment 1

1. GE Letter G-LD-9-160 dated October 6, 1989 "Explanation of Lead Factor change in River Bend Technical Specifications".

2. GE Report SASR 89-20, Rev. 1 "Implementation of Regulatory Guide 1.99, Revision 2 for River Bend Station Unit 1".

GE Nuclear Energy

G-LD-9-160
October 6, 1989

Mr. J.C. Deddens
Senior Vice President - RBNG
Gulf States Utilities
River Bend Station
Post Office Box 220
St. Francisville, Louisiana 70775

Attention: Erwin Zoch

Subject: EXPLANATION OF LEAD FACTOR CHANGE IN RIVER BEND TECHNICAL SPECIFICATIONS

Reference: GE Letter G-LD-9-152 dated September 19, 1989 transmitting GE Report SASR89-20, Implementation of Reg Guide 1.99 Rev. 2

Dear Erwin:

The following is in response to your questions on the lead factor values which you raised after reviewing the Referenced report.

The lead factor currently in the Technical Specifications, 0.86 at the 1/4 T location, was calculated by a combination of one-dimensional and two-dimensional neutron transport finite difference computer codes at the time the FSAR was prepared. As computational capability improved over the years, the one-dimensional calculations were replace by a second two-dimensional calculation, resulting in the revised lead factor of 0.89 at the 1/4 T location.

The combination of two two-dimensional computations is more accurate than the earlier method involving one-dimensional computations. The small reduction in conservatism associated with the revised lead factor is the result of improved calculation accuracy, and is therefore justified.

If you have any questions or comments, please contact Tom Caine at (408) 925-4047 or myself.

Sincerely,

John E. Dale
Nuclear Services Manager
(504) 295-8670

JED10062

cc: GE Nuclear Energy
T.A. Caine
C.E. McGee

GE Nuclear Energy

March 12, 1990

Mr. Erwin Zoch
Gulf States Utilities Company
Highway 61, North Access Road
St. Francisville, LA 70775

Subject: **REVISION OF RIVER BEND P-T CURVES REPORT SASR 89-20**

References:

1. Caine, T. A., "Implementation of Regulatory Guide 1.99, Revision 2 for River Bend Station Unit 1," GE Report SASR 89-20, Revision 1, March 1990.

2. Caine, T. A., "Implementation of Regulatory Guide 1.99, Revision 2 for River Bend Station Unit 1," GE Report SASR 89-20, Revision 0, September 1989.

Dear Erwin,

Enclosed are the following: one bound copy and one loose copy of Reference 1, a set of the pages changed in Reference 1, with changes highlighted, and the vessel outline drawing which shows the vessel diameter and thickness used in the pressure-temperature (P-T) curve calculations.

As we have discussed by phone, the thickness 5.81 inches used in calculating previous P-T curves, including those in Reference 2 and those currently in the Tech Specs, was incorrect. Reference 1 is based on the correct value of 5.41 inches. The most significant change in Reference 1 compared to Reference 2 is that the P-T curves A′, B′ and C′ have increased by about 7°F.

If you have any questions on the enclosed information, please give me a call at the number below.

Regards,

Tom Caine

T. A. Caine, Senior Engineer
Materials Monitoring & Structural Analysis Services
(408) 925-4047, Mail Code 747

cc: (w/o enclosures)
J. Dale, GE NSM