July 19, 1994

Docket Nos. 50-387
and 50-388

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light
Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Byram:

SUBJECT: CHANGES TO THE SUSQUEHANNA STEAM ELECTRIC STATION UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATION BASES FOR TECHNICAL SPECIFICATION 3/4.4.6 (TAC NOS. M89733 AND M89734)

By letter dated June 13, 1994, Pennsylvania Power & Light Company (PP&L) submitted a proposed change to the Bases of the Susquehanna Unit 1 and Unit 2 Technical Specifications (TS) for TS 3/4.4.6, "Pressure/ Temperature Limits". In that letter you indicated that PP&L had previously submitted the findings of a review of the design bases for the subject TS in a letter dated December 21, 1989, and the staff had found the results to be acceptable. Subsequently, PP&L had committed to revise these TS Bases based upon the results of the design bases review indicating that the reactor vessel heatup/cooldown rates as required by TS 3/4.4.6 should be monitored using saturation temperature.

The staff has reviewed the proposed modifications to the TS Bases and finds that they are a more accurate reflection of the relevant design bases than those in the previous version in the TS for each unit. The staff offers no objection to your proposal to modify the bases. Enclosed is a copy of the revised Bases pages B 3/4 4-4 through 4-9 for Susquehanna Unit 1 and pages B 3/4 4-4 through 4-9 for Susquehanna Unit 2. All staff activities related to TAC Nos. 89733 and 89734 are considered complete.

Sincerely,
/S/
Chester Poslusny, Jr., Sr. Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Revised TS Pages
cc w/enclosure:
See next page

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DATE	7/ /94	7/6/94	7/18/94	7/19/94	/ /

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Chester Poslusny Jr.

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

Background

The basis for this Technical Specification is brittle fracture mitigation of the reactor coolant pressure boundary (RCPB). Fracture mitigation assures that the RCPB meets its principal design criteria of retaining integrity as a radioactive material barrier. Fracture mitigation is achieved by controlling fracture toughness properties of RCPB materials during the design phase and by imposing the pressure/temperature limits and material surveillance requirements of this specification during operation.

The pressure/temperature (P/T) limits are a requirement of 10CFR50 Appendix G. This Appendix requires a fracture mechanics analysis of the reactor vessel using plant specific fracture toughness properties to derive the P/T limits. Beltline and non-beltline regions of the vessel are considered. The reactor vessel is analyzed in detail because it is the RCPB component most susceptible to brittle fracture based on its geometry, material properties, and applied stresses. The analysis technique is specified by ASME Section III Appendix G and assures that P/T limits with a wide margin to brittle fracture will be provided for all operating conditions.

Fracture toughness properties of the vessel beltline materials are affected by neutron fluence. Neutron fluence increases a material's reference temperature nil-ductility transition temperature (RT_{NDT}) which increases the temperature at which brittle fracture can occur. Consequently, P/T limits for the beltline require adjustment to account for this neutron embrittlement in order to maintain the same safety margins for brittle fracture mitigation as the non-beltline regions.

Table B 3/4.4.6-1 contains the starting RT_{NDT} information and other relevant data for the vessel materials. Figure B 3/4.4.6-1 provides the expected vessel fluence as a function of service life. Using this information and the method of Regulatory Guide 1.99 Revision 2, the predicted shift in RT_{NDT} for the beltline materials is determined and incorporated into the P/T limit curves. The actual shift in RT_{NDT} will be established periodically during operation via the material surveillance program. This program is a requirement of 10CFR50 Appendix H and requires the removal and testing of vessel material specimens installed near the inside wall of the vessel in the core mid-plane region. Evaluation of the specimens is in accordance with the requirements of ASTM E 185-82.

The P/T limit curves shall be adjusted, if required, on the basis of specimen evaluation data and the recommendations of Regulatory Guide 1.99 Revision 2.

The P/T curves are composite curves established by superimposing limits derived from the analysis of those regions of the vessel that are most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the vessel will dictate the most restrictive limit. Non-beltline locations may be controlling even when RT_{NDT} adjustments for radiation embrittlement are considered for the beltline region. Across the entire pressure-temperature span of the limit curves, some locations are more restrictive, and thus the curves are composites of the most restrictive regions.

PRESSURE/TEMPERATURE LIMITS (Continued)

LCO

Compliance with the LCO requirements of 3.4.6.1 assures that the reactor vessel remains within the analyzed envelope for fracture mitigation. The basis for the LCO limits is clarified as follows:

Normal heatup/cooldown requirements -

The vessel beltline fracture analysis assumes a 100°F/hr heating or cooling rate for beltline coolant. Therefore, the 100°F in a 1-hour period limit applies to the beltline region coolant. This limit takes into account the thermal inertia of the vessel wall. The best indicator of the beltline region coolant temperature is normally saturation temperature, T_{SAT}.

Hydrostatic and leak testing requirements -

This limit requires that temperatures of vessel metal subject to stress from a given pressure during testing must be to the right of limit Curve A. Changes in coolant temperature must not exceed 20°F in a 1-hour period.

During pressure testing, the BWR is not operating at predictable saturation conditions. Additionally, coolant temperatures must not exceed 200°F which would cause a change in operational condition. Therefore, a tighter limit on changes in coolant temperature is used.

Flange temperature requirements -

10CFR50 Appendix G sets several minimum requirements based on the closure flange RT_{NDT}. The vessel analysis mandated by this Appendix requires that flange region metal temperature be at its RT_{NDT} or greater whenever it is stressed by the vessel head bolting preload. GE practice is to require this temperature to be at (RT_{NDT} + 60°F) hence the stated minimum temperature of 70°F is required.

Applicability

The potential for violating the P/T limits exists at all times when the reactor coolant system can be pressurized. Flange limits and heatup/cooldown limits can be potentially violated any time the reactor vessel material is a different temperature than its cooling source.

PRESSURE/TEMPERATURE LIMITS (Continued)

Action

Restoration to within the limits is appropriate because the reactor vessel will be placed back within the fracture analysis envelope. This action is in the proper direction to reduce RCPB stress. Additionally, an evaluation is required to determine if reactor coolant system operation may proceed. The purpose of this evaluation is to determine if RCPB integrity is acceptable and must be accomplished before the event is reconciled.

Surveillance Requirements

The bases for the surveillance requirements of this specification are clarified as follows:

4.4.6.1.1 -

Verification that operation is within limits is an appropriate surveillance when- RCPB temperature and pressure conditions are undergoing changes. Compliance with the heatup/cooldown limits assures that assumptions used in the vessel analysis remain valid. Compliance with the appropriate limit curve assures that the vessel remains within the analyzed brittle fracture envelope during the change. Curve A applies during inservice leak and hydrostatic testing. Curve B applies when heating or cooling while the core is not critical. Curve C applies whenever the core is critical.

4.4.6.1.2 -

A separate limit (Curve C) is used when the reactor is critical. Consequently, it is appropriate to verify that the pressure and temperature are within the appropriate limit prior to the withdrawal of control rods that might make the reactor critical.

4.4.6.1.3 -

The reactor vessel material surveillance program is a requirement of 10CFR50 Appendix H. This program assures that the P/T limits are being adjusted as required to account for actual neutron embrittlement of the beltline materials.

4.4.6.1.4 -

While in OPERATIONAL CONDITION 4, with coolant temperature less than or equal to 100°F, surveillance of the flange temperatures is required to ensure the 70°F limit is not violated. With coolant temperature less than or equal to 80°F, a more frequent check of the flange temperatures is required because of the reduced margin to the limit. The flange temperatures must also be verified to be above the limit prior to and during the tensioning of vessel head bolting studs to ensure that once the head is tensioned the limit is satisfied.

 Revised by NRC Letter dated July 19, 1994

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figure 3.4.6.1-1 curves C 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

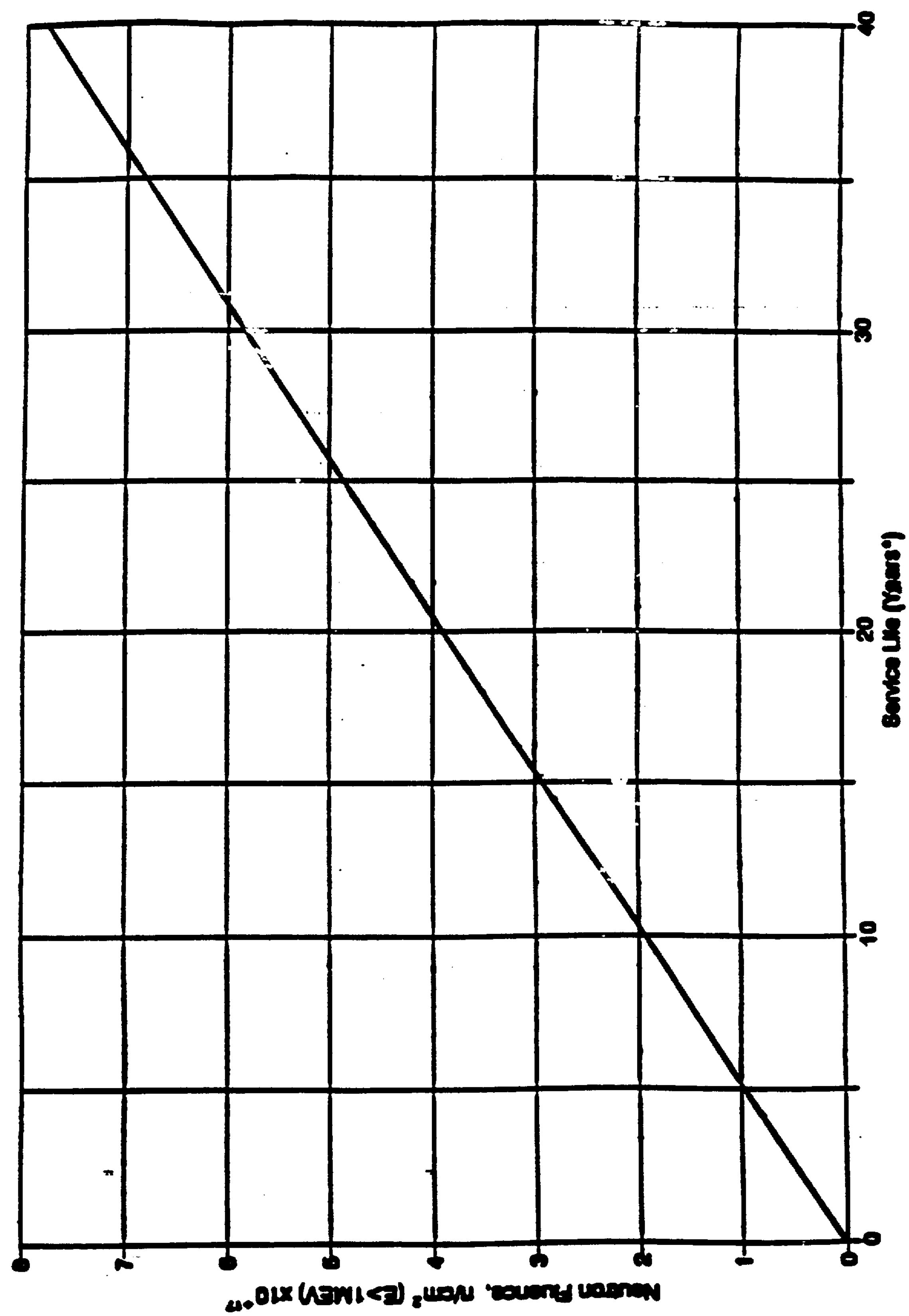
REACTOR VESSEL TOUGHNESS

Limiting Beltline Component	Weld Seam I.D. or Mat'l. Type	Heat/Slab or Heat/Lot	CU(%)	NI(%)	Starting RT_{NDT} (°F)	ΔRT_{NDT} (°F)*	Min. Upper Shelf (Lft-Lbs)	Max. RT_{NDT} (°F)
Plate	SA-533 GR B CL.1	C2433-1	0.10	0.63	+18	40	N/A	58
Weld	N/A	6296161 L320A27AG	0.04	0.99	-50	33	N/A	-17

NOTE: * These values are given only for the benefit of calculating the 32 EFPY RT_{NDT}

NON-BELTLINE COMPONENT	MATERIAL 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		No CNVS Available	0
Closure Studs	SA-540 GR B24	82552	+70

Revised by NRC Letter dated July 19, 1994



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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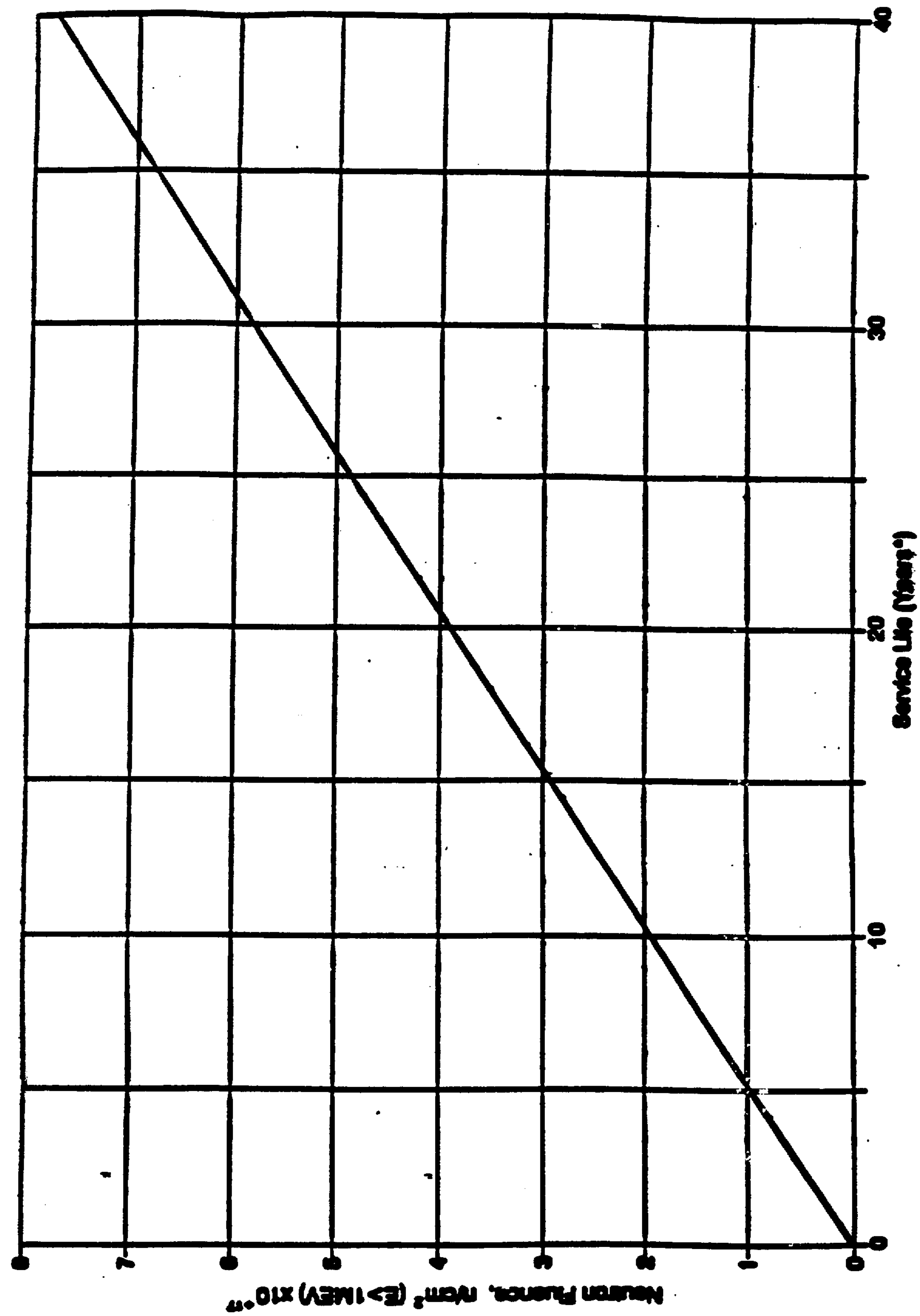
Plate	SA-533 GR B CL.1	C2421-3	0.13	0.68	-10	56.7	N/A	46.7
Weld	N/A	624263/ E204A27A	0.06	0.89	-20	50	N/A	+30

NOTE: * These values are given only for the benefit of calculating the 32 EFPY RT_{NDT} per R.G. 1.99 Rev. 2

NON-BELTLINE COMPONENT	MATERIAL 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 Flanges to Shell Top Head Other Non-Beltline	All All All	-20 -20 0
Closure Studs	SA-540 GR B24	All	Meet requirements of 45 ft-lbs and 25 mils lateral expansion at +10°F



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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