(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3339 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No.   , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*</u>

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.4.6	PRESSURE/TEMPERATURE LIMITS Reactor Coolant System................................	3/4 4-21
	Figure 3.4.6.1-1 Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits - Curve A	3/4 4-23
	Figure 3.4.6.1-2 Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits - Curve B	3/4 4-23a
	Figure 3.4.6.1-3 Core Critical Heatup and Cooldown Pressure/Temperature Limits - Curve C	3/4 4-23b
	Table 4.4.6.1.3-1 (Deleted)..........................	3/4 4-24
	Reactor Steam Dome......................................	3/4 4-25
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES........................	3/4 4-26
3/4.4.8	STRUCTURAL INTEGRITY....................................	3/4 4-27
3/4.4.9	RESIDUAL HEAT REMOVAL	
	Hot Shutdown..	3/4 4-28
	Cold Shutdown...	3/4 4-29
3/4.5 EMERGENCY CORE COOLING SYSTEMS		
3/4.5.1	ECCS - OPERATING..	3/4 5-1
3/4.5.2	ECCS - SHUTDOWN...	3/4 5-6
3/4.5.3	SUPPRESSION CHAMBER.....................................	3/4 5-8
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	
	Primary Containment Integrity..........................	3/4 6-1
	Primary Containment Leakage............................	3/4 6-2
	Primary Containment Air Locks..........................	3/4 6-5
	MSIV Sealing System....................................	3/4 6-7
	Primary Containment Structural Integrity...............	3/4 6-8
	Drywell and Suppression Chamber Internal Pressure.......	3/4 6-9

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3339 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

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 3.4.6.1-3 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 surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

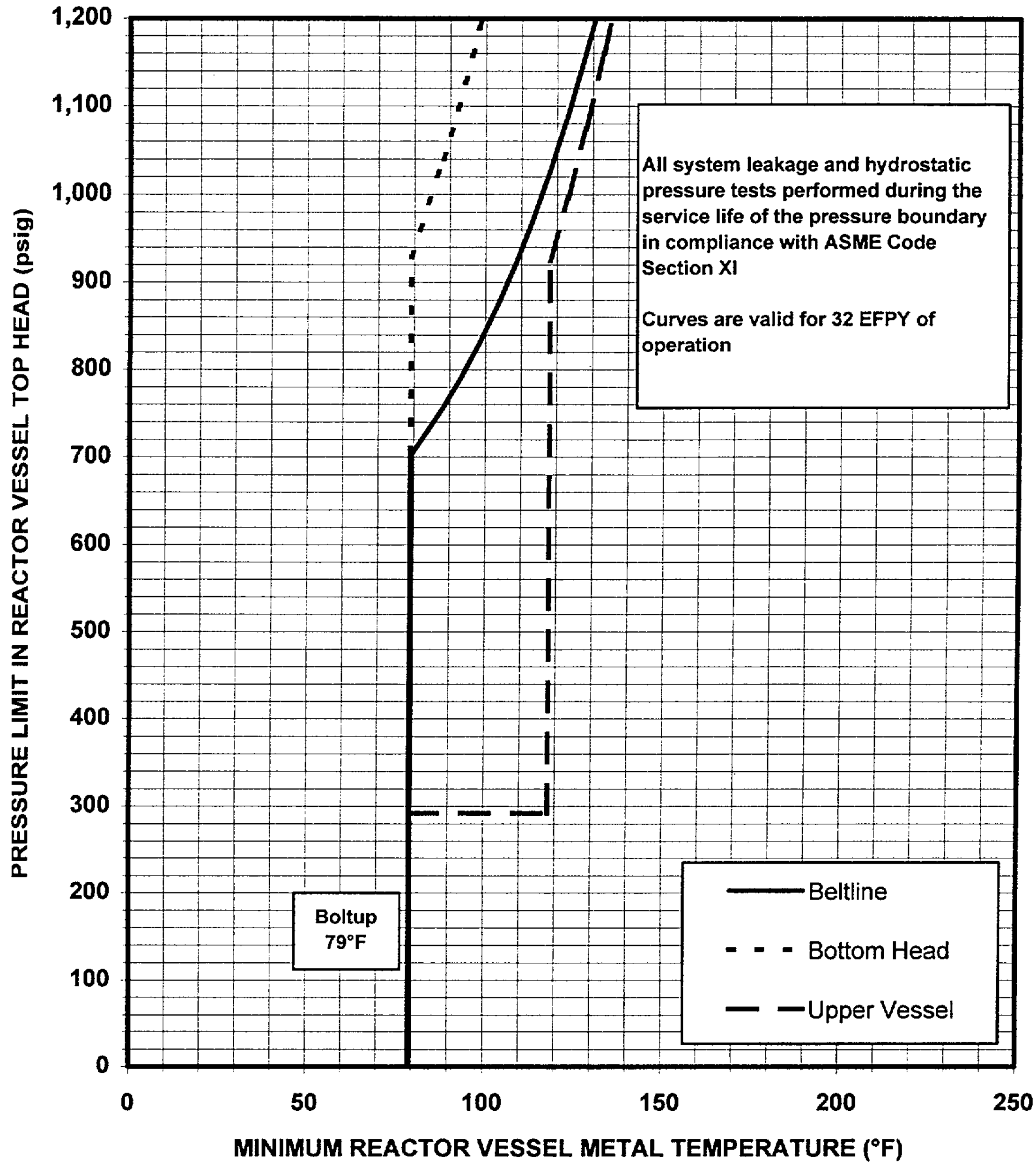
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to the limit specified in 3.4.6.1.d.

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

   1. ≤ 110°F, at least once per 12 hours.
   2. ≤ 90°F, at least once per 30 minutes.

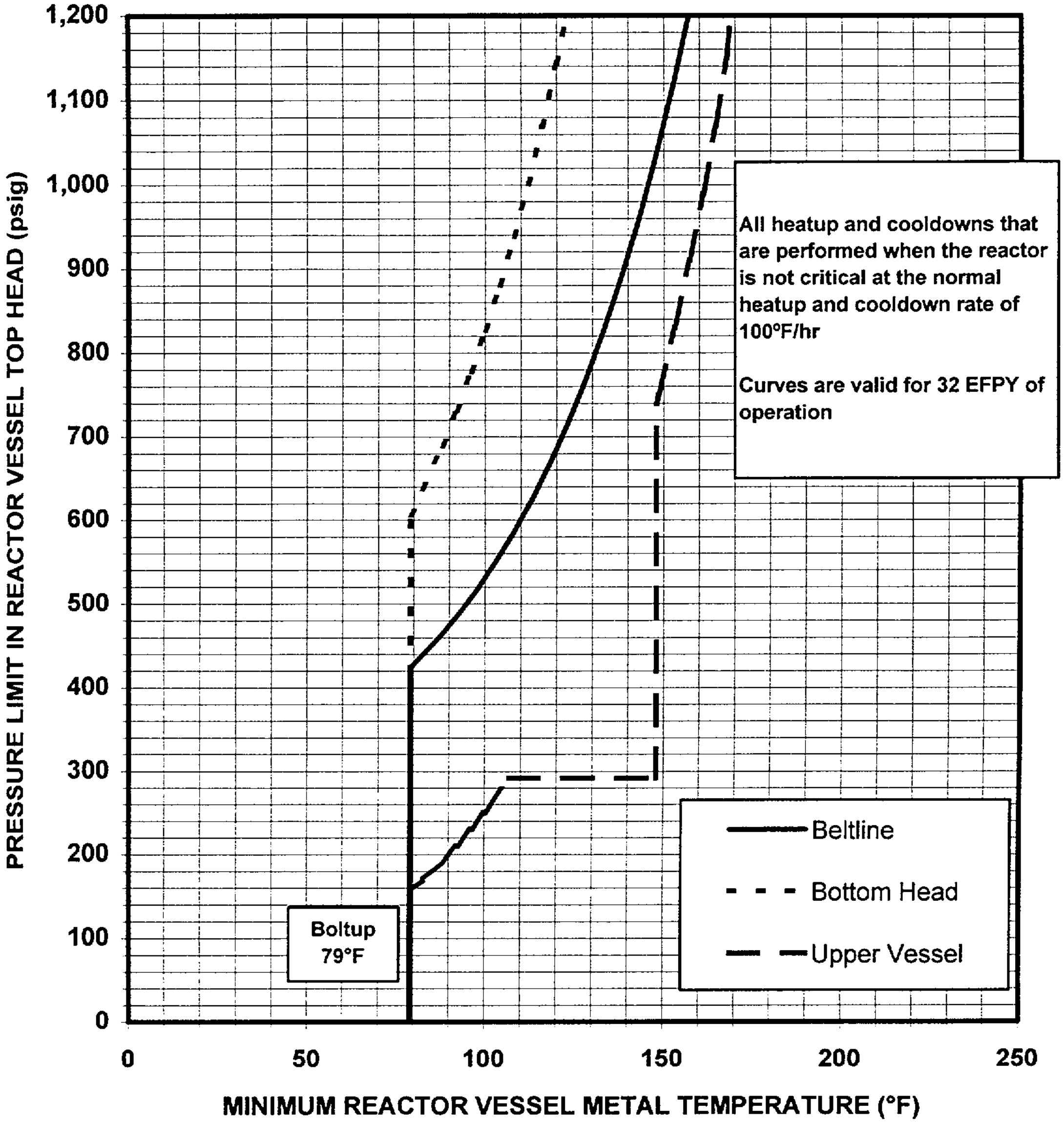
b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

Figure 3.4.6.1-1
Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits - Curve A



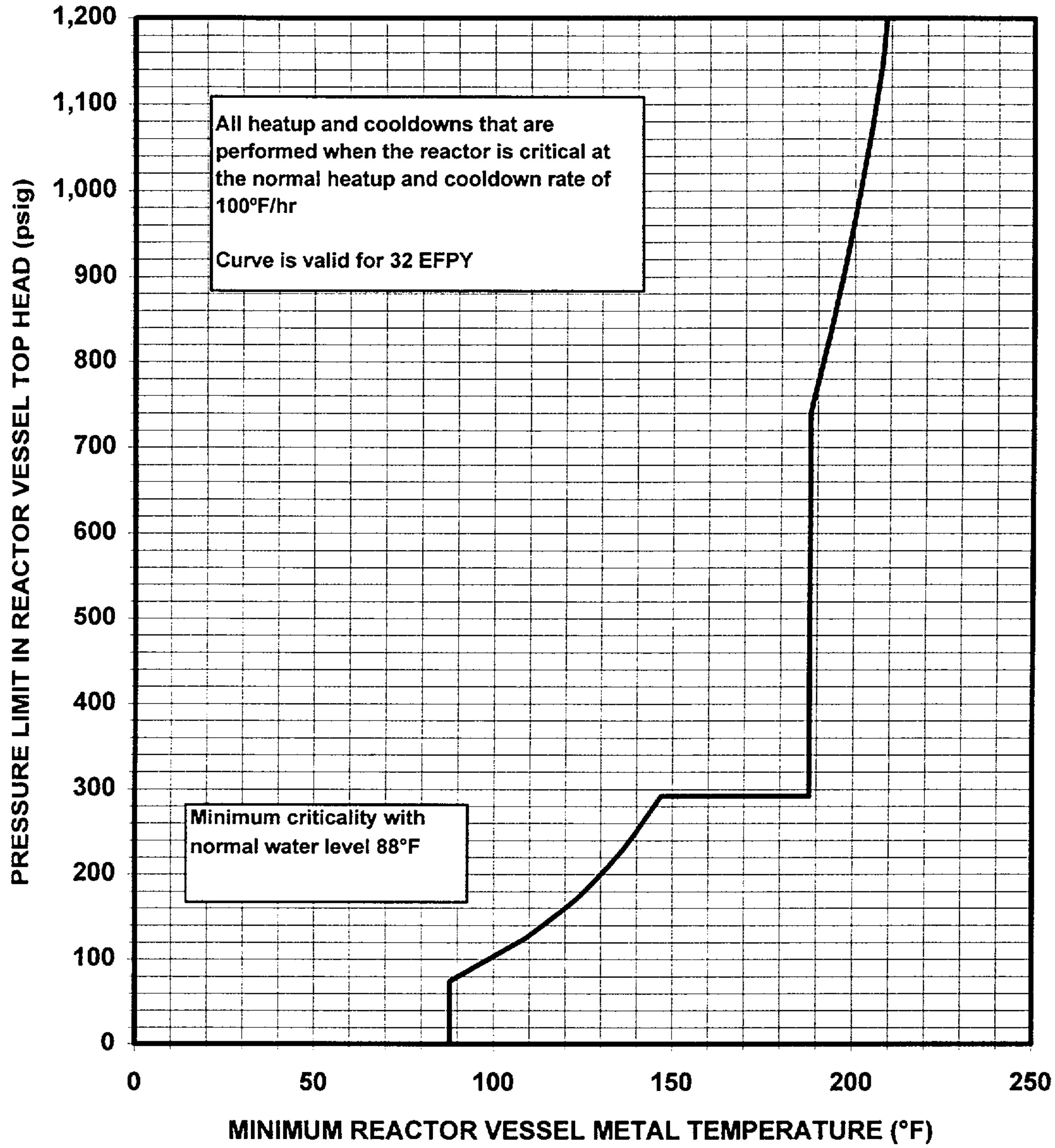
Note: This figure is valid through Cycle 11 Operation in accordance with NRC Safety Evaluation Report supporting Amendment No. 131

Figure 3.4.6.1-2
Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits - Curve B



Note: This figure is valid through Cycle 11 Operation with NRC Safety Evaluation Report supporting Amendment No. 131

Figure 3.4.6.1-3
Core Critical Heatup and Cooldown Pressure/Temperature Limits - Curve C



Note: This figure is valid through Cycle 11 Operation in accordance with NRC Safety Evaluation Report supporting Amendment No. 131

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 UFSAR. 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.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G and ASME Code Cases N-588 and N-640. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some 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 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, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/ temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni(%)	HIGHEST RT_{NDT}(°F)	ART_{NDT}(°F)	PREDICTED EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	0.71	+19	53.8	67	72.8
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	.08	0.59	-30	63.1	120	33.1

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

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, C1.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, C1.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, C1.1	All Heats	+30
LPCI Nozzles(1)	SA 508, C1.2,	All Heats	-20
Top Head Torus	SA 533, GR.B, C1.1	All Heats	+19
Top Head Flange	SA 508, C1.2	All Heats	+10
Vessel Flange	SA 508, C1.2	All Heats	+10
Feedwater Nozzle	SA 508, C1.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

(1) The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of 2.83×10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of +24.6°F.

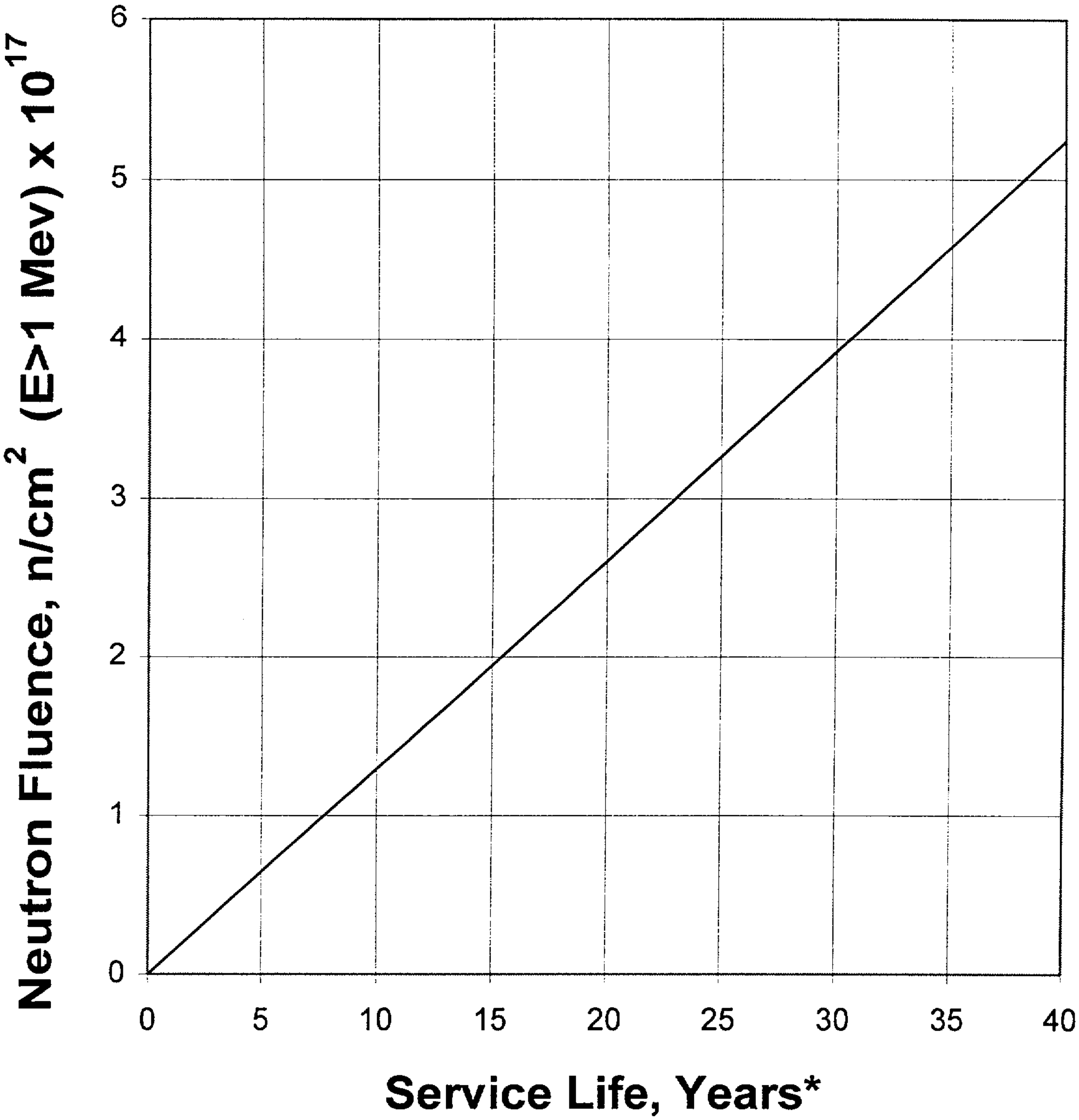


FIGURE B 3/4 4.6-1 LOWER - INTERMEDIATE SHELL FAST NEUTRON FLUENCE (E>1 MeV)
AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

*At 80% capacity factor (40 years = 32 EFPY)

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in References 1, 2 and 3.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

REFERENCES

1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel," (latest approved revision)

2. NEDE-24011-P-A (latest approved revision), "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

3. CENPD-397-P-A, " Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology,'' latest approved revision.