



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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. NPF-57

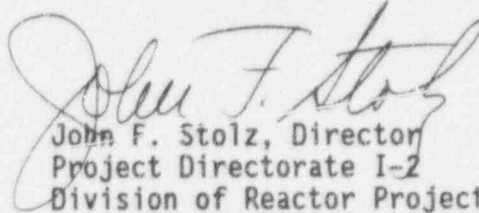
1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
 - A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated July 27, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 88, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance, and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: November 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

<u>Remove</u>	<u>Insert</u>
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
3/4 4-23	3/4 4-23
---	3/4 4-23a
---	3/4 4-23b
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7

REACTOR COOLANT SYSTEM

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 3.4.6.1-1 (hydrostatic or leak testing), and Figure 3.4.6.1-2 (heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS), and Figure 3.4.6.1-3 (operations with a critical core other than low power PHYSICS TESTS), with:

- a. A maximum heatup of 100°F in any one hour period,
- b. A maximum cooldown of 100°F in any one 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 metal temperature shall be maintained greater than or equal to 79°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 Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 as applicable, at least once per 30 minutes.

REACTOR COOLANT SYSTEM

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 3.4.6.1-3 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 70°F:

- a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:
 1. $\leq 100^{\circ}\text{F}$, at least once per 12 hours.
 2. $\leq 80^{\circ}\text{F}$, 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.

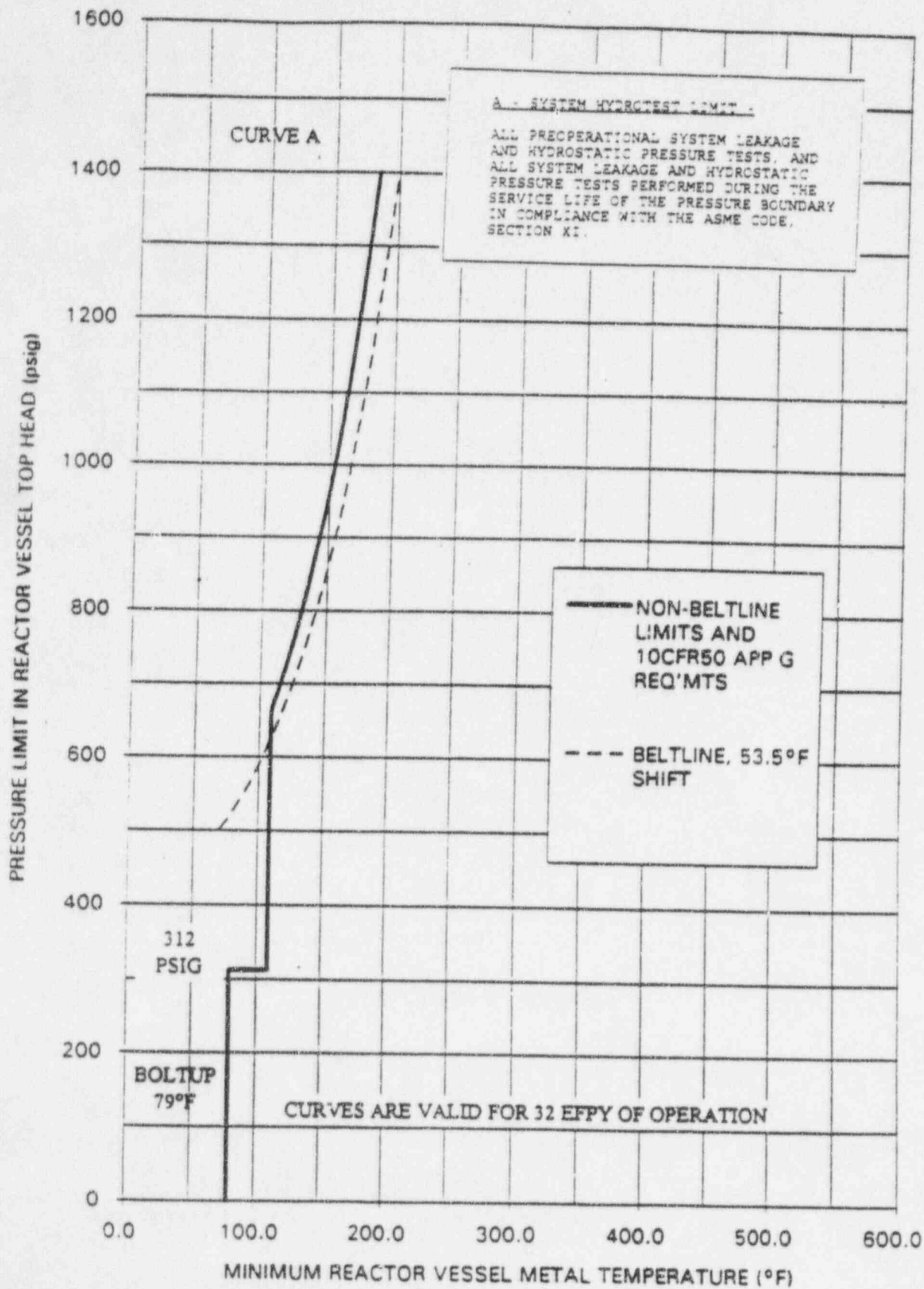
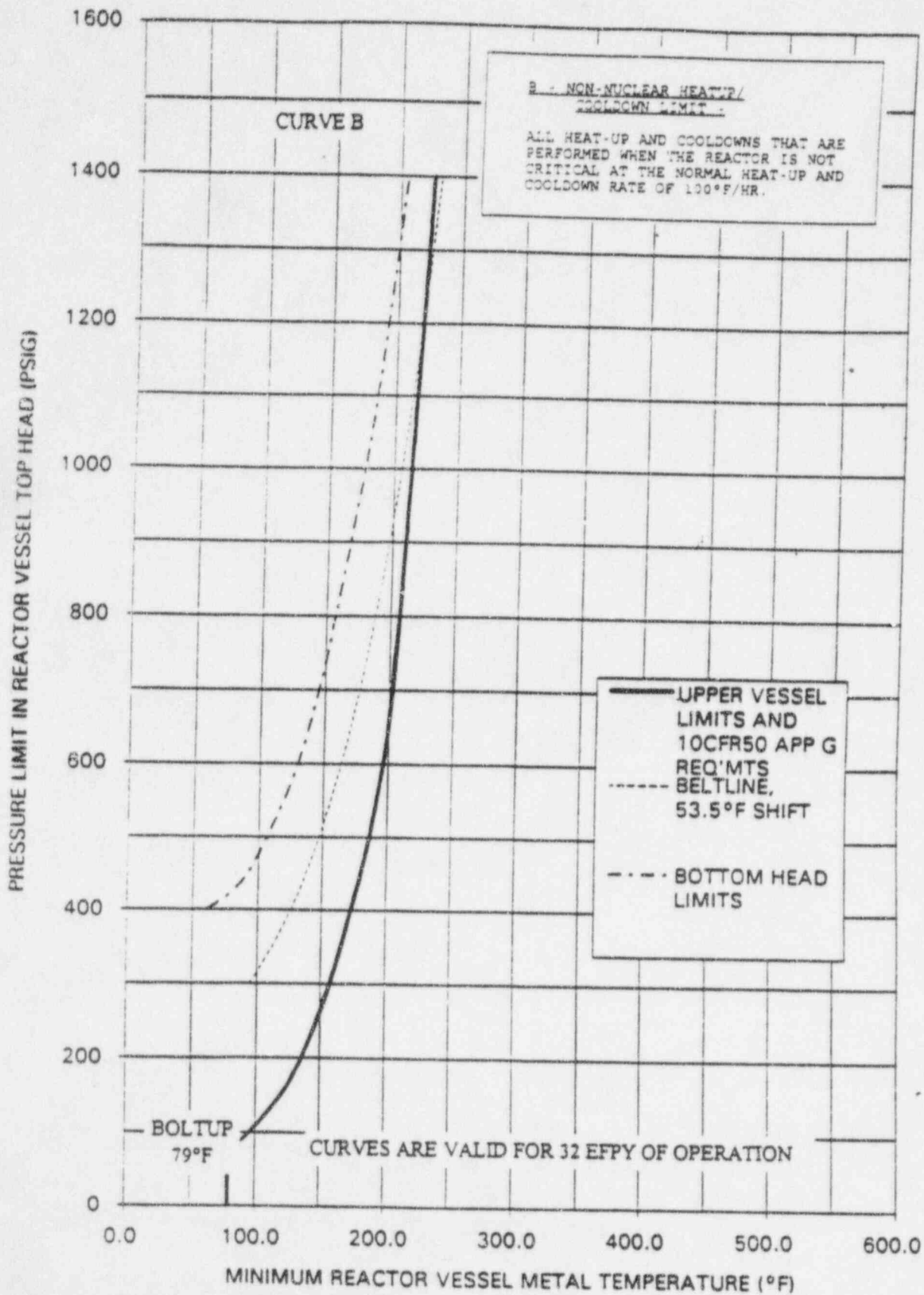
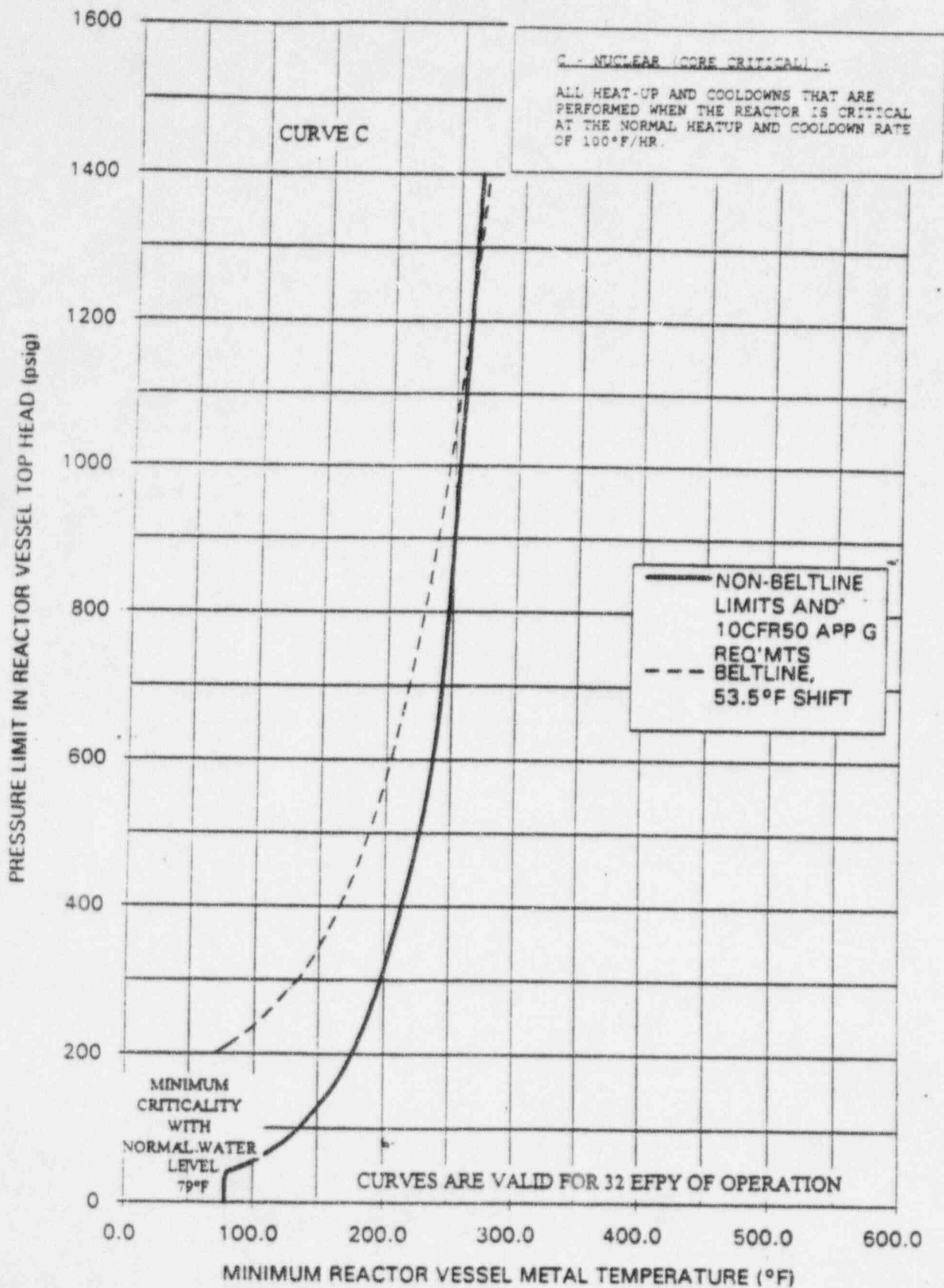


FIGURE 2.1.6.1.1





REACTOR COOLANT SYSTEM

BASES

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 III, Appendix G. 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.

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-3, curves for inservice leak and hydrostatic testing and reactor criticality 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.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in UPSAR Section 5.3 and Appendix 5A.

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).

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

<u>BELTLINE COMPONENT</u>	<u>WELD SEAM I.D. OR MAT'L TYPE</u>	<u>HEAT/SLAB OR HEAT/LOT</u>	<u>CU(%)</u>	<u>Ni(%)</u>	<u>HIGHEST RT_{NDT}(°F)</u>	<u>ΔRT_{NDT}(°F)</u>	<u>PREDICTED EOL UPPER SHELF (FT-LBS)</u>	<u>MAX. EOL RT_{NDT}(°F)</u>
Plate	SA-533 GR B CL.1	5K3025-1	.15	0.71	+19	53.5	67	72.5
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	.08	0.59	-30	2.8	120	32.8

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

<u>NON-BELTLINE COMPONENT</u>	<u>MT'L TYPE OR WELD SEAM I.D.</u>	<u>HEAT/SLAB OR HEAT/LOT</u>	<u>HIGHEST REFERENCE TEMPERATURE RT_{NDT}(°F)</u>
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 2.81×10^{17} n/cm². Therefore, these nozzles are predicted to have an EOL RT_{NDT} of +24°F.