Attachment 1

Technical Specification Changed Pages
in Support of the Pressure/Temperature Limits Report.

REMOVE	INSERT
ii	ii
iii	iii
xx(a)	xx(a)
1-5	1-5
----	1-5a
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	---------
3/4 4-21	----------
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
6-19	6-19

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DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

add 1.44 from previous page

INDEX

LIST OF FIGURES

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OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

insert from following page

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or
2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position. except as provided in Table 3.6.3-1 of Specification 3.6.3.

move 1.32 to page 5a

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

Insert to page 1-5

PRESSURE/TEMPERATURE LIMITS REPORT

1.31A The PRESSURE/TEMPERATURE LIMITS REPORT (PTLR) is the WNP-2 specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.12. Plant operation with these operating limits is addressed in LCO 3.4.6.1 (REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS).

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REACTOR COOLANT SYSTEM

provided in the PRESSURE/TEMPERATURE LIMITS REPORT during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

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 limits ~~lines shown on Figure 3.4.6.1 -- (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 heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-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 80°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

limits as specified in the PRESSURE/TEMPERATURE LIMITS REPORT

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 Figure 3.4.6.1 curves A, B or C, as applicable,~~ at least once per 30 minutes.

WNP-2 Pressure/Temperature Limits curve C of the PRESSURE/TEMPERATURE LIMITS REPORT

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 curve 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.

Deletion requested in Supply System Letter 602-92-087 dated April 10, 1992

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 Part 50, Appendix H in accordance with the NRC approved schedule ~~in Table 4.4.6.1.3-1~~. The results of these examinations shall be used to update the curves of Figure ~~3.4.6.1.~~ 1 of the PRESSURE/TEMPERATURE LIMITS REPORT.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

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

1. $\leq$ 100°F, at least once per 12 hours.

2. $\leq$ 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.

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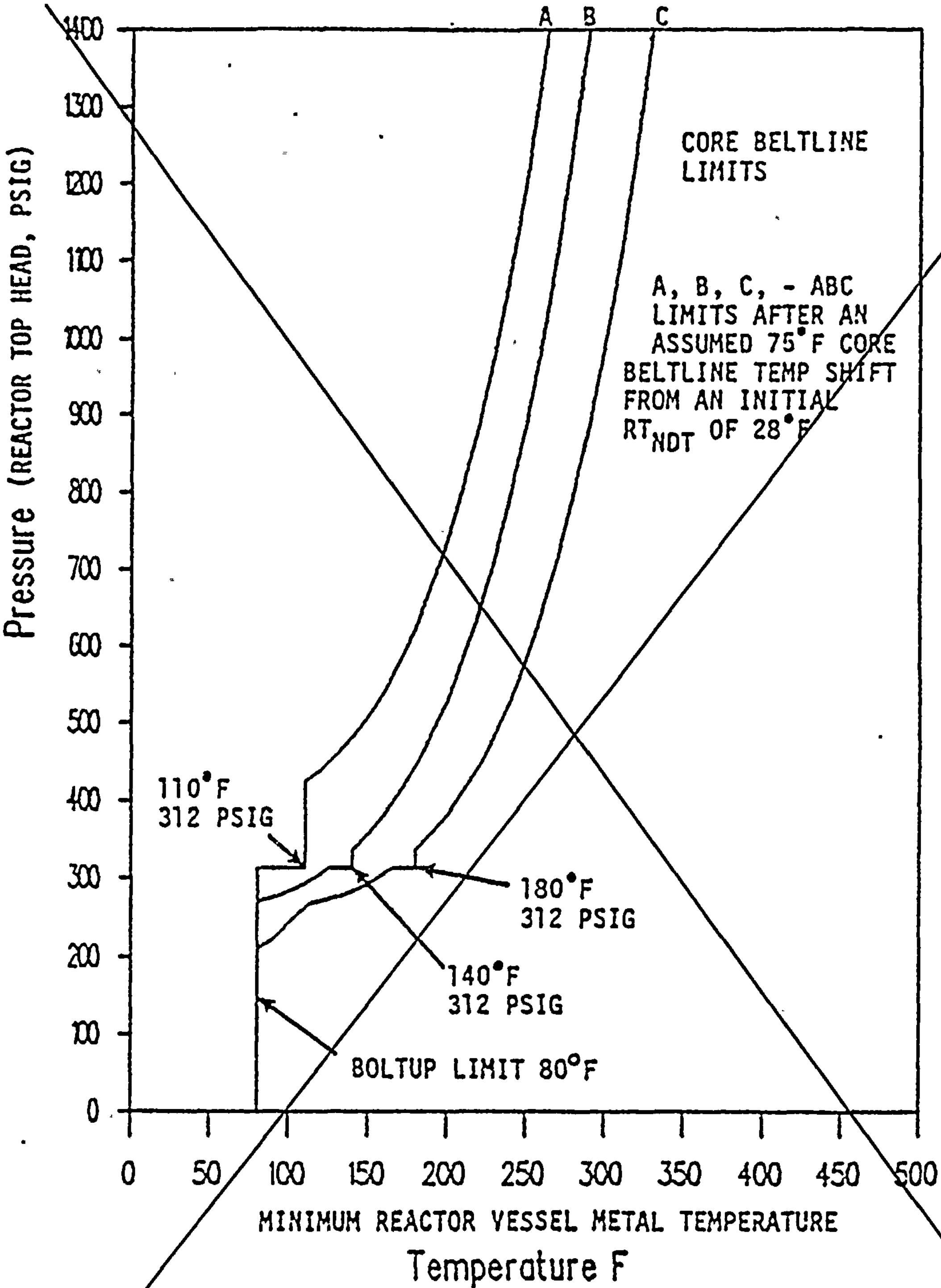

Relocated to PRESSURE/TEMPERATURE LIMITS REPORT

FIGURE 3.4.6.1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VERSUS REACTOR VESSEL PRESSURE

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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 4.9 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 results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron irradiation, E greater than 1 MeV, 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, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/ temperature limit curve, ~~Figure 3.4.6.1~~ includes predicted adjustments for this shift in RT_{NDT}. ~~for the end of life fluence and is effective for 10 EFPY.~~ provided in the PRESSURE/TEMPERATURE LIMITS REPORT

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 Part 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~~ the PRESSURE/TEMPERATURE LIMITS REPORT shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

the PRESSURE/TEMPERATURE LIMITS REPORT

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in ~~Figure 3.4.6.1~~ 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.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

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 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 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.

RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.11 The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a(a)(2).

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

Insert from following page.

Insert to page 6-19:

PRESSURE/TEMPERATURE LIMITS REPORT

6.9.1.12 The PRESSURE/TEMPERATURE LIMITS REPORT is the unit-specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates for the current reactor vessel fluence period. Plant operation within these operating limits is addressed in Specification 3.4.6.1 (Pressure/Temperature Limits).

Pressure and temperature limits shall be established and documented in the PRESSURE/TEMPERATURE LIMITS REPORT before each reactor vessel fluence period or any remaining part of a reactor vessel fluence period for Specification 3.4.6.1, Pressure/Temperature Limits. The analytical methods used to determine the pressure and temperature limits shall be those previously reviewed and approved by the NRC. The pressure/temperature limits shall be determined so that all applicable limits of the safety analysis are met. The PRESSURE/TEMPERATURE LIMITS REPORT, shall be provided upon issuance, for each reactor vessel fluence period to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Changes to this report shall become effective after review and acceptance by the Plant Operating Committee and the approval of the Plant Manager.

Attachment 2

Example of the Pressure/Temperature Limits Report

Controlled Copy No.________

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WNP-2

PRESSURE/TEMPERATURE LIMITS REPORT

REVISION 0

APRIL 1992

PTLR Implementation

Revision 0 of this report has been reviewed and adopted by the WNP-2 Plant Operating Committee in POC meeting 92-14 held April 1, 1992.

T. M. Erwin, Sr Engineer Materials & Welding	Date
S. L. Scammon, Supervisor Plant Technical	Date
S. L. McKay, Manager Operations	Date
A. G. Hosler, Manager WNP-2 Licensing	Date
J. W. Baker, WNP-2 Plant Manager	Date

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1.0 Pressure/Temperature Limits

This Pressure/Temperature Limits Report for WNP-2 has been prepared in accordance with the requirements of Technical Specification Section 6.9.1.12. The pressure/temperature (P/T) limits have been developed using the methodology provided in the identified references.

The following pressure-temperature limits are included in this report.

1) Allowable plant heatup and cooldown rates.

2) Curve A for hydrostatic or leak testing.

3) Curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS.

4) Curve C for operation with a critical core other than low power PHYSICS TESTS.

2.0 Allowable Heatup and Cooldown Rates

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 below required heatup and cooldown limits. The pressure and temperature shall be determined at least once every 30 minute interval. The determined temperature and pressure shall be to the right of the appropriate curves in Figure 1.

The following limits shall apply:

1. A maximum heatup of 100^{o} F in any 1-hour period.

2. A maximum cooldown of 100^{o} in any 1-hour period.

3. A maximum temperature change of less than or equal to 20^{o} F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

4. The reactor vessel flange and head temperature must be greater than or equal to 80^{o} F when reactor vessel head studs are under tension.

3.0 References

1. Supply System calculation ME-02-89-58, Dated 4/2/90.

2. 10 CFR 50 Appendix G requirements.

3. Reg. Guide 1.70 Section 5.3.2/Pressure-Temperature limits

[illegible]

WNP-2 Pressure/Temperature Limits

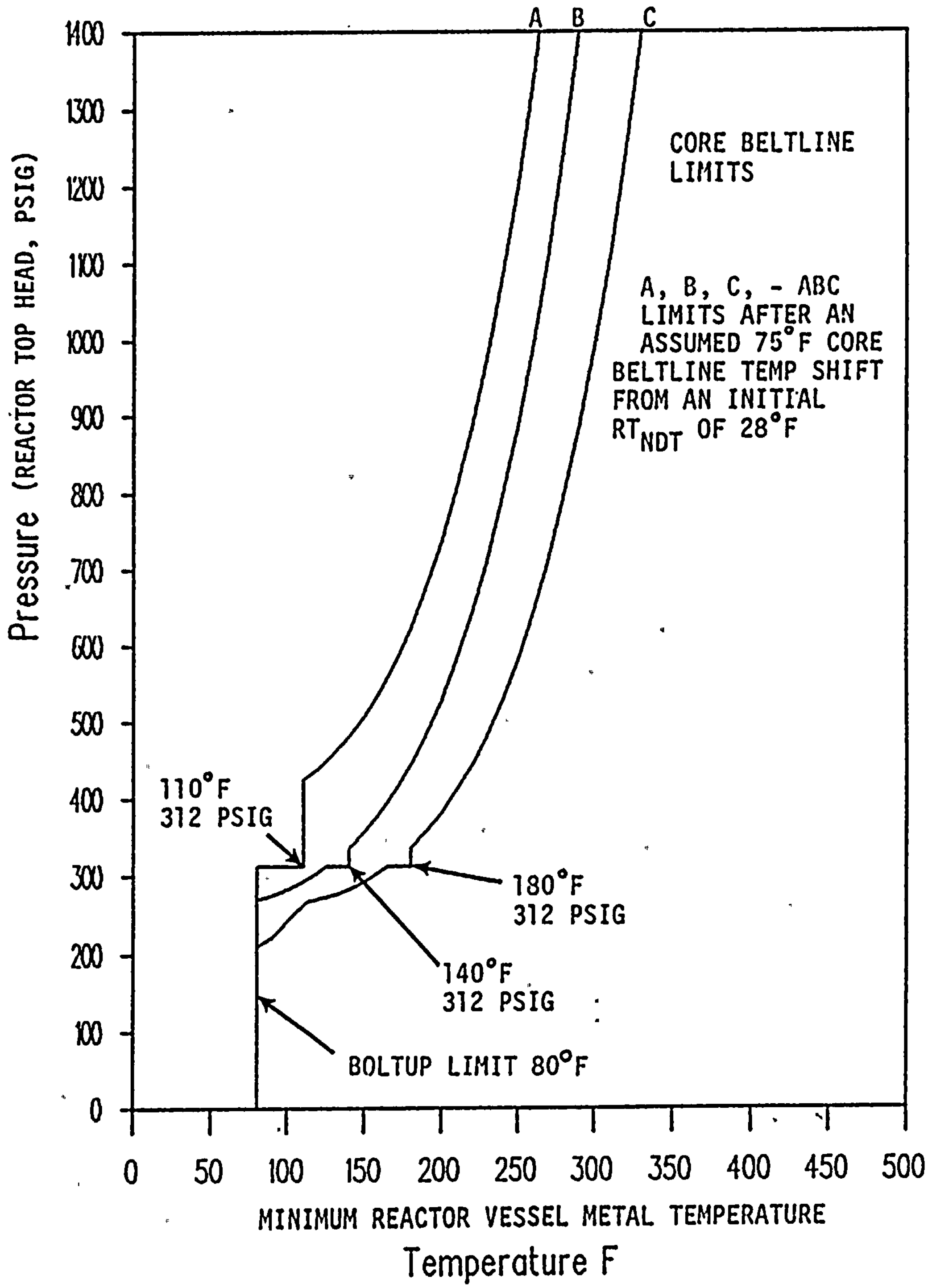

Figure 1