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THE B&W OWNERS GROUP

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Method Of Compliance With Fracture Toughness And Operational Requirements Of 10 CFR 50, Appendix G

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TECHNOLOGIES

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METHOD OF COMPLIANCE WITH FRACTURE TOUGHNESS AND OPERATIONAL REQUIREMENTS OF 10 CFR 50, APPENDIX G

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EXECUTIVE SUMMARY

This document is a new revision to an existing topical report, BAW-10046, Rev. 2 which was approved by the NRC in June 1986. Revision 3 of this report was submitted in April 1990, however, the B&W Owners Group requested the NRC to suspend its review of this report; pending a new revision (revision 4) to this report. Both revisions 2 and 3 had a section for elastic-plastic fracture mechanics analytical procedure to address the low upper-shelf toughness issue. Since the low upper-shelf issue was resolved through separate submittals, this section is removed from BAW-10046, Rev. 4. In addition, this revision contains changes prompted by the revisions of Appendix G to Section XI, ASME Boiler and Pressure Vessel Code.

In comparison with the last approved version of this report, Revision 2, the current report differs in the following major areas;

1. Complete removal of all the reference to the low upper-shelf toughness issue since this issue has been disposed by separate submittals of topical reports.
2. Stress intensity factor solutions by Raju and Newman and those found in the 1995 editions of Appendix G and Code Cases N-588 and N-640.
3. Code Case N-588 allows the postulated defect orientation coincident to the weld seam direction for circumferential weld consideration.
4. A new section on Low Temperature Over-pressurization (LTOP) introduced by Code Case N-514 and subsequent Code change shown in 1995 Code edition.
5. Reference to the reactor material properties to the B&W Owners Group GL-92-01 Response.
6. Code Case N-640 that allows the use of the K_{Ic} curve in place of the K_{Ia} curve and a revised LTOP setpoint.

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1. INTRODUCTION

BAW-10046, Rev. 2 [1] was approved by the Nuclear Regulatory Commission (NRC) in June 1986. Revision 3 of this report was submitted in April 1990, however, the B&W Owners Group requested the NRC to suspend its review of this report; pending another revision to this report. Both revisions 2 and 3 had a section for elastic-plastic fracture mechanics analytical procedure to address the low upper-shelf toughness issue. Since the low upper-shelf issue was resolved through separate submittals, this section is removed from BAW-10046, Rev. 4. In addition, this revision contains changes prompted by the revisions to Appendix G to Section XI, ASME Boiler and Pressure Vessel Code (hereafter ASME Code).

This revision completely eliminates the low upper-shelf toughness issue. This was prompted by the B&W Owners Group submittal and subsequent approval by the NRC of the low upper-shelf toughness topical reports, BAW-2178PA [2] and BAW-2192PA [3]. The acceptance criteria and the methodology to address the low upper-shelf toughness is now available in Appendix K of Section XI, ASME Code [4].

A summary of the changes made to this topical report revision is;

1. Complete removal of all the reference to the low upper-shelf toughness issue since this issue has been disposed by separate submittals of topical reports.

2. Stress intensity factor solutions by Raju and Newman and those found in the 1996 editions of Appendix G and Code Cases N-588 [5] and N-640 [6].

3. Code Case N-588 allows the postulated defect orientation coincident to the weld seam direction for the weld metal consideration.

4. A new section on Low Temperature Over-pressurization (LTOP) introduced by Code Case N-514 [7] and subsequent Code change shown in 1995 Code edition [8].

5. Reference to the reactor material properties to the B&W Owners Group GL-92-01 Response [9].

6. Code Case N-640 [11] that allows the use of the K_{Ic} curve in place of the K_{Ia} curve and a revised LTOP set-point.

2. REQUIREMENTS OF APPENDIX G TO 10 CFR PART 50

2.1. Background

In 1973, the Nuclear Regulatory Commission issued a new appendix to 10CFR Part 50, entitled "Appendix G - Fracture Toughness Requirements." Subsequently this Appendix G was revised in 1983. This revised appendix specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors and provides specific guidelines for determining pressure-temperature operational limitations on the RCPB. The toughness and operational requirements are specified to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the RCPB may be subjected over its service lifetime. In 1987, Appendix G of the ASME Code Section III was adopted by Section XI and also called it Appendix G. This appendix was identical with Section III Appendix G except for K_{IR}, which is labeled as K_{Ia} to be consistent with Appendix A of Section XI. Later in 1993 addenda, the first revision to Section XI was issued to specify how to generate low temperature over-pressurization (LTOP) limits.

2.2. Appendix G, 10 CFR Part 50

The latest revision of Appendix G to 10 CFR Part 50 was issued in December of 1995 [10]. There have been no major changes in this revision except for some clarifications and presentation of various minimum temperature requirements in Table 1 of Ref. 6. The requirements of the current Appendix G to 10 CFR Part 50 are specified in §IV.A and §IV.B is for annealing requirements.

Excerpts from Appendix G, 10 CFR Part 50

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may continue to be operated only for that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forging, the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

1. *Reactor Vessel Charpy Upper-Shelf Energy Requirements.*

- This part is irrelevant to this report and left blank. –

2. *Pressure-Temperature Limits and Minimum Temperature Requirements*

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In Table 1, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.

b. The pressure-temperature limits identified as "ASME Appendix G limits" in Table 1 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

c. The minimum temperature requirements given in Table 1 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 1, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in Table 1.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

Table 2-1 (Table 1 in Ref. 10) Pressure and Temperature Requirements

Operating Condition	Vessel Pressure[1]	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤ 20%	ASME Appendix G Limits	(2).
1.b Fuel in the vessel	> 20%	ASME Appendix G Limits	(2) + 90°F (6).
1.c No fuel in the vessel (Preservice Hydrotest Only).	ALL	(Not Applicable)	(3) + 60°F.
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤ 20%	ASME Appendix G Limits	(2).
2.b Core not critical	> 20%	ASME Appendix G Limit	(2) + 120°F (6).
2.c Core critical	≤ 20%	ASME Appendix G Limits + 40°F	Larger of [(4)] or [(2) + 40°F].
2.d Core critical	> 20%	ASME Appendix G Limits + 40°F	Larger of [(4)] or [(2)+160°F].
2.e Core critical for BWR(5)	≤ 20%	ASME Appendix G Limits + 40°F	(2) + 60°F.

(1) Percent of the preservice system hydrostatic test pressure.

(2) The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.

(3) The highest reference temperature of the vessel.

(4) The minimum permissible temperature for the inservice system hydrostatic pressure test.

(5) For boiling water reactors (BWR) with water level within the normal range for power operation.

(6) Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

2.3. Pressure-Temperature Limits and Minimum Temperature Requirement

This requirement is in two parts: the first is the "ASME Appendix G Limits," and the second is the ASME Code Section III minimum temperature requirements based on material RT_{NDT}. This is summarized in Table 1 in Appendix G to 10 CFR Part 50.

2.3.1. ASME Appendix G Limits

In 1987, Section XI of ASME Boiler and Pressure Vessels Code adopted a non-mandatory appendix, Appendix G, which is a copy of Appendix G of Section III of the ASME Code. The reasoning behind this is that the nuclear power plant operational requirements belong to Section XI and Section III addresses design and construction of nuclear power plants. The content of the Section XI Appendix G was identical to the Appendix G of Section III until the first revision was made to accommodate the addition of an LTOP limit in the 1993 addenda of the ASME Code. The methods to be used for determining the ASME Appendix G Limits are described in Section 5.

2.3.2. Minimum Temperature Requirements

All the minimum temperature requirements listed in Table 1 of 10 CFR Part 50 are satisfied with the above ASME Appendix G limits.

3. REACTOR OPERATING CONDITIONS

3.1. Pressure Boundary Components

The reactor coolant pressure boundary (RCPB) is defined by 10 CFR 50.2, (v). The reactor coolant (RC) system is typically made up of the following components: reactor vessel, steam generators, pressurizer, reactor coolant pumps, valves and interconnecting piping. The RC system contains and circulates reactor coolant at the pressure and velocity necessary to transfer the heat generated in the reactor core to the secondary fluid in the steam generators.

The other pressure-containing portions of the RCPB are the auxiliary system components. These include the makeup and purification system piping and valves (including RC pump seal injection lines); the emergency core cooling system high- and low-pressure and core flooding injection piping and valves; the vent, drain, and other piping and valves used for maintaining the RC system; and the incore instrumentation piping.

Portions of the RCPB are exempted from the requirements for Class 1 components of ASME Code Section III by footnote 2 to NRC Regulation 10 CFR 50.55a. Components of the RCPB included under this exemption provision are generally designed and fabricated in accordance with the requirements for Class 2 components in ASME Code Section III (see Regulatory Guide 1.29, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants"). None of these components are constructed of ferritic material except the core flood tanks, which are carbon steel in some B&W designed plants. Although the core flood tanks are isolated from the RC system by two valves during normal operation, connecting piping to the tanks (1-inch lines for nitrogen addition fill and drain) does penetrate reactor containment. Therefore, the system is part of the RCPB to the outermost containment isolation valve. Since these tanks are isolated from the RC system during all conditions of normal operation, including anticipated operational occurrences, they need not be considered in developing the RC system pressure-temperature limitations and are not discussed in this report.

3.2. Ferritic Materials and RCPB Operational Parameters

The ferritic materials used in construction of the RCPB for a typical nuclear power plant are listed for each component in Table 3-1. The pressure boundary of the RC system is fabricated primarily from ferritic materials with clad austenitic steel weld metal, while that of the auxiliary systems is fabricated primarily from austenitic stainless steel.

Consequently, only the RC system components require special protection against non-ductile failure and must comply with the fracture toughness requirements of 10CFR Part 50, Appendix G. This protection against non-ductile failure is ensured by imposing pressure-temperature limitations on the operation of the RC system. The margin of safety is controlled by the maximum calculated allowable pressure at any given temperature. The following loading conditions require pressure-temperature limits: 1) Test Conditions, and 2) Normal Operations

The required protection against non-ductile failure and typical operational parameters of the RC system are described in the following paragraphs for each of the loading conditions.

3.3. Test Conditions

3.3.1. Preservice System Hydrostatic Test

Prior to initial operation, the RC system is hydrostatically tested in accordance with ASME Code requirements. During this test, the system is brought up to an internal pressure not less than 1.25 times the system design pressure. This minimum test pressure is in accordance with Article NB-6000 of ASME Section III. Since the system design pressure is 2500 psig, the preservice system hydrostatic test pressure is 3125 psig. Initially, the RC system is heated to a temperature above the calculated minimum test temperature required for adequate fracture toughness. Heatup is accomplished by running the RC pumps. The pressurizer heaters are used to heat the pressurizer to the required temperature. Before the test temperature is reached, the pressure is maintained above the net positive suction head (NPSH) required for RC pump operation but below the maximum allowable pressure for adequate fracture toughness. When the test temperature is reached, the RC pumps are stopped and RC makeup water is added to fill the pressurizer. The test pressure is then reached using either the pressurizer heaters or the hydrostatic pumps connected to the RC system. The test pressure is held for the minimum specified time, and then examination for leakage in accordance with the ASME Code is performed.

3.3.2. Inservice System Leakage and Hydrostatic Tests

When inservice system leakage & hydrostatic tests are required, the system is brought from cold to hot shutdown. The means of heating the system and increasing the pressure are the same as those used during normal heatup. If it is necessary to cool the system down after either test, normal cooldown procedures are used. These tests are conducted in accordance with the requirements of ASME Section XI, Article IWA-5000. The test pressure for the inservice leakage tests is the pressure that, for the component located at the highest elevation in the system, is no less than the system nominal operating pressure at 100% rated reactor power. The requirements for the minimum test pressure versus the test temperature for the inservice hydrostatic tests are given in ASME Section XI Table IWB-5222-1. The test temperatures for both the inservice leakage and hydrostatic tests are determined by the requirements for fracture toughness.

3.4. Normal Operation

3.4.1. Bolt Preload

During bolt preload, the reactor vessel closure studs are tensioned to the specific load. Bolt preloading is not allowed until the reactor coolant temperature and the volumetric average temperature of the closure head region (including the studs) are higher than the specified minimum preload temperature. After the studs are tensioned, system pressure can be increased by the pressurizer until it is above the NPSH required for RC pump operation. The heatup transient begins when the RC pumps are started.

3.4.2. Heatup

During heatup the RC system is brought from cold shutdown to hot shutdown. The heat sources used to increase the temperature of the system are the RC pumps and any residual (decay) heat from the core. During this procedure the coolant temperature is always above the minimum specified bolt preload temperature. Initially, the reactor coolant temperature may be as low as room temperature for initial core loading or as high as 130°F after refueling.

At any given time throughout the heatup transient, the temperature of the reactor coolant is essentially the same throughout the system except in the pressurizer. The system pressure is controlled by the pressurizer heaters and is maintained between the minimum required pressure for RC pump NPSH and the maximum pressure defined by the fracture toughness requirements. The heatup rate is maintained below the maximum rate used to establish the maximum allowable pressure-temperature limit curve.

3.4.3. Cooldown

RC system cooldown brings the system from hot to cold shutdown. The cooldown is normally accomplished in two phases. The first phase reduces the fluid temperature from approximately 550°F to below the design temperature of the decay heat removal system (approximately 300°F). The second phase is further cooling by the decay heat removal system. The actual procedures to reduce temperature vary slightly from plant to plant and are in the plant specific functional specifications.

3.4.4. Reactor Core Operation

The reactor core is not allowed to become critical until the RC system fluid temperature is above a certain temperature as defined in the plant technical specification. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40°F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

Table 3-1. Ferritic Materials Used in a Typical Reactor Coolant Pressure Boundary

Component	Material
Reactor Vessel	
Plates	SA 533, Grade B, Class 1
Forging	SA 508, Class 2; SA 320, Grade L43
Bolting	SA 540, Grade B-23
Welds	SFA 5.5, SFA 5.17
Steam Generator	
Plates	SA 533, Grade B, Class 1; SA 516, Grade 70
Forging	SA 508, Class 1 or Class 2
Bolting	SA 320, Grade L43
Welds	SFA 5.5, SFA 5.17
Pressurizer	
Plates	SA 516, grade 70; SA 212, Grade B
Forging	SA 508, Class 1
Bolting	SA 320, Grade L43
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Piping	
Plates	SA 516, Grade 70
Forging	SA 105, Grade II
Seamless Pipe & Tubing	SA 106, Grade C
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Pump	
Forging	SA 105, Grade II
Bolting	SA 540, Grades B-23, -24; SA 193, Grade B7

4. MATERIAL PROPERTIES

4.1. Determination of Reference Temperature for Nil-Ductility Transition (RT_{NDT})

The key material parameter regarding Appendix G to 10 CFR Part 50 is the nil-ductility transition reference temperature (RT_{NDT}). This parameter is the basis for the required minimum temperatures and the fracture toughness K_{Ia} in the ASME Appendix G limits. The RT_{NDT} values for unirradiated materials are determined by the rules given in the ASME Section III, NB-2331. The RT_{NDT} values for irradiated materials, needed only for beltline materials, are obtained by the rules in the USNRC Regulatory Guide 1.99, Rev. 2 [11]. According to the regulatory guide rules, the RT_{NDT} value of unirradiated materials is defined as the initial RT_{NDT} (IRT_{NDT}).

In Regulatory Guide 1.99, Revision 2, the adjusted RT_{NDT}, ART, is defined as;

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where

IRT_{NDT} = reference temperature for the unirradiated material
ΔRT_{NDT} = mean value of the adjustment in reference temperature caused by irradiation
Margin = value used to obtain conservative upper-bound values

The ART is the RT_{NDT} value for irradiated materials and the RT_{NDT} is input to the following ASME Code reference fracture toughness equation;

$$K_{Ia} = 26.78 + 1.223 \exp\{0.0145(T - RT_{NDT} + 160)\}, \text{ ksi}\sqrt{\text{in}}.$$

Since the fracture toughness of the beltline region materials will change throughout the lifetime of a reactor vessel, periodic adjustments to the pressure-temperature limit curves of the RCPB are required. The magnitude of these adjustments is proportional to the shift in RT_{NDT} caused by neutron irradiation. Therefore, it is essential to determine the radiation-induced change in RT_{NDT} of the beltline region materials.

Since the ΔRT_{NDT} is based on the temperature shift of the Charpy curves measured at the 30 ft-lb level, it is necessary to know, by analysis or from the results of the material surveillance program, the magnitude of the Charpy 30 ft-lb shift.

4.2. Determination of RT_{NDT} of Unirradiated Material

The RT_{NDT} of the ferritic materials are determined in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or later Editions and Addenda. When sufficient material is available, the RT_{NDT} of the beltline region materials are obtained by testing specimens oriented normal to the principal working direction. The test procedure is in accordance with ASME Section III, paragraph NB-2300 (Summer 1972 or later Edition and Addenda).

4.3. RT_{NDT} Values for Pressure Boundary Component Materials

Table 4-1 presents RT_{NDT} values for base metals in typical RCPB materials. All initial RT_{NDT} data and additional data required to calculate adjusted RT_{NDT} for all the B&W Owners Reactor Vessel Working Group plants are contained in topical report BAW-2325, Revision 1 [9].

4.4. Mechanical Properties of RCPB Materials

All mechanical properties related to ferritic materials can be obtained from the ASME Code, Section II.

Table 4-1. RT_{NDT} Data

Material/Type	RT_{NDT}, °F				
	No. of Data	High Measured	Avg. Measured	Estimated Value	Estimated minus Avg. Measured
SA 508, Class 2 forging	24	60	4	60 or T_{NDT}[a]	56
SA 533 Grade B low-alloy plates	13	40	0	40	40
SA 516 Grade 70 carbon steel plates	20	10	-11	10	21
SA 508 Class 2 HAZ	6	30	-25	30	55
SA 533 Grade B HAZ	11	10	-23	10	33
SA 516 Grade 70 HAZ	7	-20	-26	-20	6
SA 106 Grade C piping	11	50	5	50	45

[a]60°F or the drop weight temperature, if known.

5. ANALYTICAL METHODS FOR DETERMINING ASME APPENDIX G LIMITS

5.1. Pressure Boundary Components

All the reactor coolant system (RCS) pressure boundary components are to be considered for the ASME Appendix G limits. However, due to neutron radiation induced embrittlement, the beltline region of a reactor vessel is the most critical component. In addition, the closure head is important in the Appendix G consideration due to pre-operational bolt-up stress and the minimum temperature requirement in Table 2-1 (Table 1 of Appendix G to10 CFR Part 50). Nozzles are also considered with a postulated nozzle corner crack as discussed in Welding Research Council Bulletin (WRCB) 175 [12]. For the ASME Appendix G limits consideration, closure head, outlet nozzle, and beltline region are the three components to be analyzed and other components are considered to be bounded by these three.

5.2. Maximum Postulated Defect

5.2.1. Beltline Region of Reactor Vessel

In Appendix G of the ASME Boiler and Pressure Vessel Code, (Section III and Section XI), it is required to postulate a quarter of reactor vessel thickness surface defect in the direction perpendicular to the maximum stress as shown in Figure 5-1. There is no requirement in Appendix G as to which surface the reference defect should be postulated on. In applications this defect is postulated on both internal and external surfaces of reactor vessels, even though there is no water environment and no known causes for crack initiation or propagation on the external surface of reactor vessels.

In Code Case N-588 issued in December 1997, it is allowed to postulate a quarter thickness defect in circumferential direction in a circumferential weld as shown in Figure 5-2.

Figure 5-1 Postulated Longitudinal Flaw in Reactor Vessel

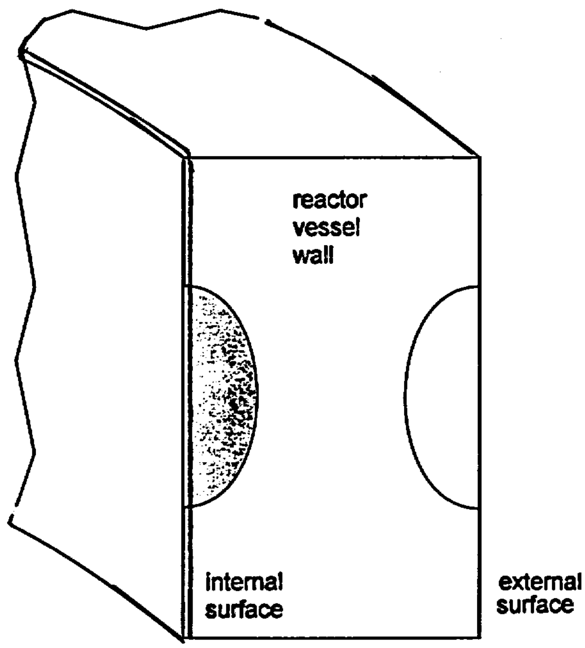
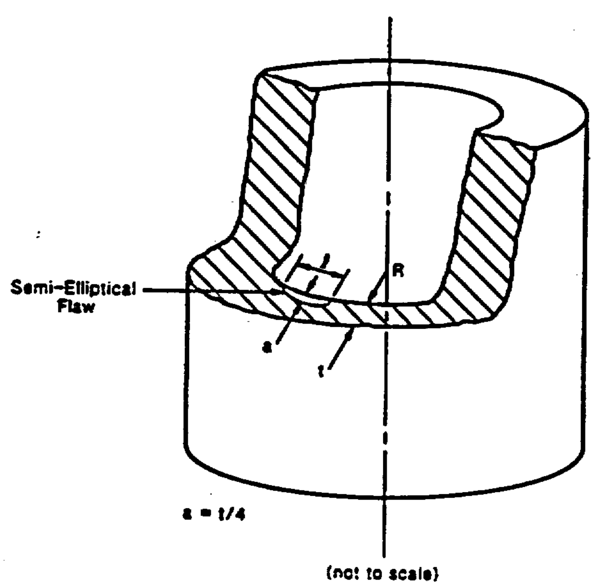


Figure 5-2 Postulated Circumferential Flaw in Reactor Vessel



5.2.2. Closure Head

According to Article G-2220 of Appendix G ASME Section XI, a smaller flaw size may be postulated. Closure head is subjected to high stresses due to bolt preload-induced bending and internal pressure. Since the bolt stresses induce tensile bending stress on the external surface, an external surface defect is postulated. Considering the relatively easy inspection of this region, the maximum defect size was assumed to be one sixth of the thickness. This maximum defect for closure head region was approved in the Revision 1 of this report and has been in use since 1978. The postulated flaw orientation is in the circumferential direction to coincide with the direction of the weld and is perpendicular to the maximum bending stress orientation. Since the 1/6 t location from the outside is rather insensitive to the inside downcomer reactor coolant temperature, the closure head limit is controlled by heatup rates. This limit bounds the cooldown limit of the closure head. In addition, with an increase in fluence on the beltline material, most of the plant closure head limit is bounded by the beltline region material and becomes inconsequential in the final composite limit curve.

5.2.3. Outlet Nozzle

As also allowed by Article G-2223, a smaller flaw size of one tenth of the thickness was postulated in the corner of an outlet nozzle, based on past inspection results which showed no presence of any significance size-defects. And also the reliability of defect detection and sizing was demonstrated for nozzle corner regions earlier under another B&W Owners Group effort (BAW-2208 [13]). The outlet nozzle is entirely made of forging, SA 508 Cl 1 or Cl 2.

Since the nozzle area does not have much exposure to neutron radiation compared with the beltline region, it has been shown that the nozzle corner crack limit is always bounded by the beltline. The pressure-temperature limit derived from the postulated nozzle corner crack is automatically satisfied by the limits based on the beltline region.

5.3. Fracture Toughness of Reactor Vessel Steels

Appendix G of Section XI of the ASME Code requires that the following K_{Ia} curve to be used in pressure-temperature limit calculations (up to and including 1995 edition and 1996 addenda);

$$K_{Ia} = 26.78 + 12.445 \exp[0.0145(T - RT_{NDT})], \ \ \mathrm{ksi}\sqrt{\mathrm{in}}\ .$$

When a licensee elects to use the latest ASME Code Case N-640 through an exemption, the following K_{Ic} curve can be used in place of the K_{Ia}. For this case, the LTOP set point should be at 100% of the Appendix G pressure-temperature limit instead of 110% allowed in N-514.

$$K_{Ic} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})], \ \ \mathrm{ksi}\sqrt{\mathrm{in}}\ .$$

5.4. Transient Temperature and Thermal Stress Calculation

The analytical processes used in the determination of the allowable pressures for the generation of pressure-temperature limit are described in the sections below.

The temperature profile through the reactor vessel wall is determined by solving the one-dimensional axisymmetric heat conduction equation:

$$\rho C_p \frac{\partial T}{\partial t} = k\left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right)$$

The equation is subjected to the following boundary conditions at the inside and the outside walls of the reactor vessel;

at the inside wall where $r = R_i$,

$$-k\frac{\partial T}{\partial r} = h(T_w - T_b)$$

at the outside wall where $r = R_o$,

$$\frac{\partial T}{\partial r} = 0$$

where,

- ρ = density
- C_p = specific heat
- k = thermal conductivity
- T = vessel wall temperature
- r = radius
- t = time
- h = convective heat transfer coefficient
- T_w = wall temperature
- T_b = bulk coolant temperature
- R_i = inside radius of vessel wall
- R_o = outside radius of vessel wall

The above equation is solved numerically using a finite difference or finite element method to determine the wall temperature as a function of radius, time, and thermal transient rate. Once a temperature profile through the reactor vessel wall is determined for a particular time point, the corresponding thermal stresses are determined from the theory of elasticity or a finite element model in the radial direction. This numerical procedure is particularly important for multi-rate heatup and cooldown cases. Most of B&W Owners Reactor Vessel Working Group plants have the multi-rate heatup and cooldown ramps in their technical specifications.

5.5. Stress Intensity Factors, K_I

The stress intensity factors in Code Cases N-514, N-588, N-640 and in Appendix G to Section XI of the 1995 and later editions were obtained using the K_I solutions listed below, without crack face pressure loading.

5.5.1. Longitudinal Semielliptical Surface Flaws

The stress intensity factor equation for a longitudinal semielliptical surface flaw in a cylinder by Raju and Newman [15] has been widely used in the industry since 1982 and is considered well suited for the purpose because this solution is for a cylinder not for a plate. The solution is based on finite element analysis of a cylinder with radius to thickness ratios of 10 and 4 and crack depths of 0.2, 0.5, and 0.8. The Raju-Newman equation can be presented in two different forms: One in a general form that is amenable to a polynomial stress profile through the wall as follows;

$$K_{I_t} = \sqrt{\pi \frac{a}{Q}} \sum_{j=0}^{3} G_j A_j a^j \qquad (5\text{-}1)$$

where

$$Q = 1 + 1.464\left(\frac{a}{c}\right)^{1.65} \qquad (5\text{-}2)$$

and:

G_j	=	influence coefficients
A_j	=	coefficients of cubic fit of through-thickness thermal stress
a	=	crack depth
c	=	½ crack length

and another form in terms of an internal pressure load for internal and external surface semielliptical flaws;

$$K_{I_p} = \frac{p R i}{t}\sqrt{\pi \frac{a}{Q}}\, F \qquad F = F_i \text{ or } F_e \qquad (5\text{-}3)$$

where F_i is the boundary-correction factor for an internal surface crack,

$$F_i = \frac{t}{R_i}\left(\frac{R_o^2}{R_o^2 - R_i^2}\right)\left[2G_0 - 2\left(\frac{a}{R_i}\right)G_1 + 3\left(\frac{a}{R_i}\right)^2 G_2 - 4\left(\frac{a}{R_i}\right)^3 G_3\right] \tag{5-4}$$

where R_i = inner radius
R_o = outer radius
t = shell thickness
p = pressure

For an external surface flaw,

$$F_e = \frac{t}{R_i}\left(\frac{R_i^2}{R_o^2 - R_i^2}\right)\left[2G_0 + 2\left(\frac{a}{R_o}\right)G_1 + 3\left(\frac{a}{R_o}\right)^2 G_2 + 4\left(\frac{a}{R_o}\right)^3 G_3\right] \tag{5-5}$$

There is an additional option to use the K_I equation provided in older version of Appendix G of the ASME Code directly.

5.5.2. Circumferential Semielliptical Surface Flaws

The K_I solution for a circumferential flaw shown in Fig. 5-2 is from Kumar et al [16].

$$K_I = \sigma\sqrt{\frac{\pi a}{Q}}F(a/l, a/t, R/t) \tag{5-6}$$

where: σ = applied stress

$$\sigma = p\left(\frac{R_i^2}{R_o^2 - R_i^2} + 1\right)$$

p = internal pressure

$$F = 1.026 + 0.27(a/t) + 0.40\,(a/t)^2 \tag{5-7}$$

There is an additional option to use the K_I equation provided in older version of Appendix G of the ASME Code directly.

5.5.3. Nozzle Corner Semielliptical Surface Flaw

The determination of stress intensity factor for a nozzle corner crack is based on the method contained in WRCB 175, which gives the following equation:

$$K_I = \sigma\sqrt{\pi a}F(a/r_n)$$

where $F = 2.5 - 6.108(a/r_n) + 12(a/r_n)^2 - 9.1664(a/r_n)^3$

σ = hoop stress

a = crack depth

r_n = apparent radius of nozzle, in which is given by the equation,

$r_n = r_i + 0.29r_c$

r_i = actual inner radius of nozzle, in.

r_c = nozzle radius, in.

The membrane stress is the hoop stress due to pressure and is determined using Lame's solution for thick wall cylinders subjected to internal pressure. The maximum hoop stress is developed at the inside surface of the wall and is given by:

$$\sigma = p\,\frac{R_o^2 + R_i^2}{R_o^2 - R_i^2}$$

The maximum hoop stress at the inside surface is conservatively assumed as a uniform membrane stress across the entire wall thickness.

5.6. Determination of Appendix G Limits

The governing equation for determining the pressure-temperature operating limit curves is equation 1 from Article G-2215;

$$(S.F.)\ K_{Im} + K_{It} \leq K_{Ia}$$

where S.F. - safety factor,

= 2 for normal and upset conditions

= 1.5 for ISLH condition

= 1.2 for preservice test condition

K_{Im} - stress intensity factor due to internal pressure

K_{It} - stress intensity factor due to thermal gradient

K_{Ia} - reference fracture toughness and

$$K_{Ia} = 26.78 + 12.445\exp[0.0145(T - RT_{NDT})],\ \text{ksi}\sqrt{\text{in.}}$$

From a thermal transient analysis, a temperature profile is calculated for a given time point during a heatup or cooldown transient. Once the crack tip temperature, T, and the material RT_{NDT} at that location are known, K_{Ia} is determined. RT_{NDT} is determined from RG 1.99 with fluence and chemical composition of the material. Also the temperature profile determines the thermal stresses at various points throughout the reactor vessel wall. If one defines the stress intensity factor due to a pressure of one psi, $\overline{K}_{Ip}$, then, the allowable pressure is determined using the following equation:

$$P_{allow} = \frac{K_{Ia} - K_{It}(T_f)}{(S.F.) \cdot \overline{K}_{Ip}}$$

where,

K_{Ia} = reference stress intensity factor
$\overline{K}_{Ip}$ = stress intensity factor due to 1 psi pressure
T_f = fluid temperature.

A plot showing the allowable pressure as a function of bulk coolant temperature is a pressure-temperature limit curve. The same procedure applies to the ISLH conditions except the safety factor.

Code Case N-640

This code case provides two significant changes to Appendix G; the first is the replacement of the K_{Ia} fracture toughness curve used in Appendix G with K_{Ic}. This change enabled an opening of the plant-operating window by at least 20% or more. The second change is that if this Code Case is used then the LTOP set point is limited to 100% of the Appendix G limit. This code case was issued in February 1999. It is anticipated that most of the P-T limits generated in 1999 and beyond will be based on this code case.

6. PROTECTION FROM LOW TEMPERATURE OVER-PRESSURIZATION (LTOP)

The methodology for protecting against LTOP events at B&W designed plants is described in this section. The following topics are presented and discussed:

(1) Definition of reactor vessel pressure limit
(2) Definition of enable temperature
(3) Definition of LTOP transients
(4) Consequences of LTOP events
(5) Definition of methods to protect the reactor vessel pressure limit

6.1. Definition of the LTOP Pressure Limit

According to Appendix G to Section XI [8], the LTOP allowable pressure for the reactor vessel is defined as 110% of the steady-state Appendix G pressure-temperature limits. The reference flaw size and the analytical methodology are identical to those used in the Appendix G calculations. Steady-state temperatures, rather than the transient temperatures resulting from technical specification heatup and cooldown ramps, are used to establish the LTOP setpoint. Operational experience indicates that the LTOP event is an isothermal case; therefore, the steady-state approach is appropriate.

The current applicable ASME Code allows the use of 110% of the ASME Section XI, Appendix G pressure-temperature limits, when determining LTOP limits. However, Code Case N-640 (issued in February of 1999) allows the use of the K_{Ic} curve in place of the K_{Ia} curve in pressure-temperature determinations. To use this Code Case, the LTOP allowable pressure limit is set at 100% of the Appendix G pressure-temperature limits.

6.2. Definition of Enable Temperature

The enable temperature is the temperature below which the LTOP system is required to be operable. The original NRC Branch Technical Position RSB 5-2 defines the enable temperature for the LTOP systems as RT_{NDT} + 90EF at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G pressure-temperature limit calculations. However, the ASME Code Appendix G (1995 edition) defines an enable temperature of RT_{NDT} + 50EF or 200EF, whichever is greater. The identical enable temperature definition applies to Code Case N-640.

6.3. Definition of LTOP Transients

LTOP events occur as the result of equipment malfunction or operator error that results in mass or energy addition to the reactor coolant system. In the B&W plant operating history, only once has the technical specification Appendix G limits been violated due to an LTOP event [17]. Because of restrictions that preclude water-solid operation of the pressurizer (i.e., a steam or nitrogen bubble is maintained with the reactor vessel head on), this plant design is less likely to exceed Appendix G limits.

In the B&W-designed NSSS, mass can be injected into the system through:
1) the four HPI nozzles (one or two of which also serve as normal makeup);
2) the core flood nozzles through which the core flood tank system, decay heat removal system, and low pressure injection system (LPI) can provide added inventory; and
3) the pressurizer spray nozzle via HPI, LPI, or the nitrogen addition system.

Energy can be added to the RCS via:
1) failure of the decay heat removal system;
2) actuation of the pressurizer heaters; and
3) reactor coolant pumps.

As a result, the following transients were postulated and evaluated for their potential to increase reactor vessel pressure:
1) Erroneous actuation of the High Pressure Injection (HPI) system
2) Erroneous opening of the core flood tank discharge valve
3) Erroneous addition of nitrogen to the pressurizer
4) Makeup control valve (makeup to the RCS) fails full open
5) All pressurizer heaters erroneously energized
6) Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS
7) Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator

6.4. Consequences of LTOP Events

Each of the postulated LTOP events was analyzed to determine the rate of RCS pressure increase and/or the total amount of pressure increase that the system would experience. A stand alone thermal hydraulic model of the pressurizer was used for these predictions. Capabilities to model RCS inventory increases (e.g., makeup, HPI), inventory decreases (e.g., letdown), RCS expansion, and pressurizer heaters were included. A range of initial pressures and pressurizer levels were applied so that the pressurization rates could be applied to different initial P-T operating conditions. A brief summary of each transient response is provided below.

1) Erroneous actuation of the High Pressure Injection (HPI) system - this event would be the most limiting LTOP transient. However, HPI actuation results in a very rapid pressurization of the RCS and precludes achieving the necessary 10 minutes for operator action. Thus, this event is prevented below the LTOP enable temperature through plant procedures.

2) Erroneous opening of the core flood tank discharge valve - this event is precluded by closing and locking out the breakers of the motor operated block valves before the RCS pressure decreases below the CFT pressure (600 psig). This will occur prior to cooling below the ART.

3) Erroneous addition of nitrogen to the pressurizer - this event can not overpressurize the RCS because of plant equipment that regulates the nitrogen pressure to 150 psig (i.e., pressure regulator and relief valves).

4) Makeup control valve (makeup to the RCS) fails full open - this event results in a pressurization rate of 20 to 30 psi/minute and is the most limiting of the remaining LTOP events.

5) All pressurizer heaters erroneously energized - this event is a slow transient (9 to 12 psi/minute) and is bounded by the *failed open makeup control valve event.*

6) Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS - this event is a slow transient (7 psi/minute) and is bounded by the *failed open makeup control valve event.*

7) Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator - this event results in a finite increase in pressure that is less than the margin between the Appendix G and LTOP limits. Because of the presence of a pressurizer bubble, this event is much less severe than at other PWRs.

In summary, the most limiting, credible event is the *failed open makeup control valve event.* Because of system design differences, the plant response is sensitive to the makeup pump head-capacity curve and system resistance. This requires each plant to evaluate a plant specific response.

6.5. Definition of Methods to Protect the Reactor Vessel Pressure Limit

In general, each plant is equipped with either; (1) a dual setpoint pilot operated relief valve that is set below the LTOP limit, or (2) an additional relief valve (e.g., decay heat removal system relief valve) that is also set below the LTOP limit. In the event of relief valve failure, plant operation is limited (i.e., combination of operating pressure and pressurizer level) such that, in the event of the most limiting LTOP event, the failed- open makeup control valve event, either: (1) ten minutes of operator action time are available between the time the pressure-temperature operating limits are exceeded and the LTOP limits are violated, thus, providing adequate time for the operator to terminate the event, or (2) the available makeup tank volume would be exhausted and thus terminate the event before the LTOP limit is violated.

Two means of setting operating limits were used for the failed open makeup control valve event. The first approach assumes that the plant is operating at the maximum allowable pressure (as defined by bounds of the Appendix G heatup and cooldown limits and the PORV setpoint) at the time at which the failed open makeup control valve event occurs. Then, using plant specific makeup flow vs. RC pressure curves, the maximum allowable initial pressurizer level that will cause the tenth minute pressure to equal the LTOP pressure is determined. Thus, if the pressurizer level is maintained below this value for temperatures less than the enable temperature and if the RC pressure is less than the Appendix G heatup/cooldown pressure, the LTOP limit will not be exceeded during ten minutes of the failed open makeup control valve event.

The second approach is similar except that the maximum allowable pressurizer level is set and the maximum allowable pressure vs. temperature curve is determined. If this curve results in higher pressures than the Appendix G heatup/cooldown curve, the LTOP limit is protected by the Appendix G curves (for this pressurizer level). If this curve results in lower pressures, this curve is implemented as the limiting operator curve.

In performing either approach, the integrated makeup flow is determined. This allows the makeup inventory that will be exhausted during ten minutes of operation to be determined which can then be used as a means of LTOP protection.

In addition, RCS vent size calculations that will prevent pressurization during the failed open makeup control valve event are calculated to provide backup LTOP protection. For example, one plant can prevent RCS pressurization if a $0.75 in^2$ vent is available. Thus, if the PORV is declared inoperable, a vent (e.g., steam generator hand-hole) can be opened to protect against LTOP events.

7. SUMMARY AND CONCLUSION

This report presents the B&W Owners Group Reactor Vessel Working Group approach for compliance with the fracture toughness and operational requirements of 10 CFR Part 50, Appendix G. 10 CFR 50, Appendix G references Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code for technical requirements. The ASME Appendix G has been in use for more than 26 years. During this period, there have been significant technical improvements made in fracture mechanics methods and NDE techniques. To accommodate new technical developments and implement the experience gained, this procedure has been regularly updated since its first issuance in 1976.

This Revision 4 contains three particularly significant changes from Revision 2:

1. Use of improved stress intensity factor solutions based on the Raju-Newman solution.
2. Use of weld metal flaw orientation that is in the weld seam direction (Code Case N-588)
3. Use of the K_{Ic} curve in place of the K_{Ia} curve (Code Case N-640)

It is concluded that the use of these changes noted above reduces excessive conservatism in application of 10 CFR 50, Appendix G, without decreasing overall safety margins.

8. REFERENCES

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2. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," BAW-2178PA, B&W Nuclear Technologies, Lynchburg, VA.

3. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads," BAW-2192PA, B&W Nuclear Technologies[1], Lynchburg, VA, April 1994.

4. ASME Boiler & Pressure Vessel Code, Section XI, Division I, Rules for Inservice Inspection of Nuclear Power Plant Components, 1992 edition including December 1993 addenda, American Society of Mechanical Engineers, New York, NY.

5. ASME Boiler and Pressure Vessel Code Case N-588, Section XI, Division 1, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," Approval date: December 12, 1997.

6. ASME Boiler and Pressure Vessel Code Case N-640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Approval date: February 26, 1999.

7. ASME Boiler and Pressure Vessel Code Case N-514, Section XI, Division 1, "Low Temperature Overpressure Protection," Approval date: February 15, 1998.

8. ASME Boiler & Pressure Vessel Code, Appendix G to Section XI, 1995 Edition

9. M. J. DeVan, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity B&W Owners Group – Reactor Vessel Working Group," BAW-2325, Revision 1, Framatome Technologies, Inc., Lynchburg, VA, January 1999.

10. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements for Light Water Reactor Pressure Vessels; Final Rule," October 1995.

11. Regulatory Guide 1.99, Re vision 2, 1988, *Radiation Embrittlement Damage to Reactor Vessel Materials*, U..S. Nuclear Regulatory Commission.

12. PVRC Ad Hoc Group on Toughness Requirements, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," *Welding Research Council Bulletin* No.175 (August 1972).

13. BAW-2208A, "Fracture Mechanics Analysis of Nozzle-to-Pipe Welds," Framatome Technologies, February 1995.

14. Not used.

15. I.S. Raju and J.C. Newman, Jr. "Stress Intensity Factor for Internal and External Surface Cracks in Cylindrical Vessels," J. of Pressure Vessel Technology, Nov. 1982.

16. V. Kumar and M. D. German, "Elastic-Plastic Fracture Analysis of Through-Wall and Surface Flaws in Cylinders," EPRI Report NP-5596, January 1988.

17. C. Santos Jr., "Development of a Low Temperature Overpressure Protection (LTOP) Database," Office of Nuclear Regulatory Research, US Nuclear Regulatory Commission, presented at the ASME Section XI, Working Group on Operating Plant Criteria, August 11, 1998.

9. CERTIFICATION

This report is an accurate description of the compliance methodology of Appendix G to 10CFR50.

Kenneth K. Yoon 11/15/99

K. K. Yoon Date

Materials and Structural Analysis

This report was reviewed and was found to be an accurate description of the work reported.

11/15/99

D. E. Killian Date

Materials and Structural Analysis

11/29/99

M. J. DeVan Date

Materials and Structural Analysis

Verification of independent reviews

K. E. Moore 11/29/99

K. E. Moore, Manager Date

Materials and Structural Analysis Unit

This report has been approved for release.

D L Howell 11/29/99

D. L. Howell, Program Manager Date

B&W Owners Group - RV Integrity Program

APPENDIX A. SAMPLE PRESSURE-TEMPERATURE LIMITS

Using the methods described in Section 5, the maximum allowable pressure at each selected temperature point can be calculated and the resulting locus of these pressure points becomes a pressure-temperature limit curve. Nuclear power plants under normal operation condition should be operated with reactor coolant system pressure not to exceed this limit curve. There can be multiple pressure-temperature limit curves for a component. This report considers three components.

Each component has a heatup and a cooldown pressure-temperature limit curve. From operational considerations, there are two different conditions: (1) normal and (2) inservice system leak and hydrostatic (ISLH) test. For each transient analysis, both the limit that includes full transient temperature effects and the limit based on the steady-state condition without the transient thermal effects are needed. This consideration is to cover heatup or cooldown rates falling between the steady-state (0 degree per hour rate) and the maximum posted heatup or cooldown transient rate.

For most B&W Owner Group Reactor Vessel Working Group plants, the cooldown transient has multiple steps, starting with the highest rate of cooldown and decreasing to a lower cooldown rate as the bulk coolant temperature decreases. Typical examples are shown in Figures A-1 and A-2.

A.1 Pressure-Temperature Limits for Closure Head

The closure head does not receive significant neutron radiation and therefore the limit curve related to the closure head does not change over the life of the plant. Closure head limits are developed for two categories of pressure, defined in Table 2-1, as less than and greater than 20% of the preservice system hydrostatic test pressure. Since the hydrostatic test pressure is set at 125% of the system design pressure (2500 psi), system pressure is limited to 625 psi when the bulk fluid temperature is less than 180 °F (RT_{NDT} + 120 °F, from Table 2-1).

For normal heatup at 100 °F/hr, the allowable pressure increases for temperatures above 180 °F based on detailed calculations for the postulated flaw, as shown in Figure A-3. The 50 °F/hr heatup curve, which is limited by the RT_{NDT} + 120 °F restriction of Table 2-1, permits higher pressures than the 100 °F/hr case due to larger differences between the crack tip and the bulk coolant transient temperatures. The cooldown curve, controlled by the RT_{NDT} + 120 °F restriction above 625 psi, is the least limiting of the three curves presented in Figure A-3.

A.2 Pressure-Temperature Limits for Outlet Nozzle

The reactor vessel nozzles receive little neutron radiation and thus their reference temperatures remain nearly constant over time. The pressure limit curves for the outlet nozzle under a heatup transient are typically bounded by the beltline limit curves, as shown in Figure A-4. Similar behavior is observed for in the cooldown transient (Figure A-5). Pressure-temperature limits for the outlet nozzles are normally less restrictive the those for the beltline region, and therefore do not affect the final controlling limit curves.

A.3 Pressure-Temperature Limits for Beltline Region

Reference temperatures for the materials in the beltline region increase time of operation due to neutron radiation. The limit curves for this region continuously shift to the right as the neutron fluence increases, narrowing the operating pressure-temperature window. Typical contributing pressure-temperature limits curves are shown in Figures A-6 and A-7 for heatup and cooldown conditions, respectively. These curves present comparative limits for five regions of the vessel; beltline axial weld, beltline base metal, beltline circumferential weld, outlet nozzle, and closure head. The final limit is the composite minimum of the these five curves.

The pressure-temperature curves of Figures A-6 and A-7 are based on the material reference temperatures listed in Table A-1. The curves were also constructed using the K_{Ic} measure of fracture toughness, and no credit was taken for negative stress intensity factors at the crack tips.

Table A-1 RT_{NDT} Values Used for Sample Limit Curves

RT_{NDT}	Weld Metal	Base Metal
at 1/4 t	248 F	64 F
at 3/4 t	190 F	50 F

A.3.1 Limit Curves for Reactor Vessels with Circumferential Welds

Reactor vessels that have only circumferential welds in the beltline region can benefit from new analysis procedures, introduced by Code Case N-588, that permit a circumferential semi-elliptical flaw to be postulated in the weld metal (a longitudinal semi-elliptical flaw must still be used for the plate material). Comparative results for the required axial- and circumferential-oriented flaws are presented in Figure A-8 for the material reference temperatures listed in Table A-1. It is noted that in this case the weld material is bounding for all temperatures except those near the lower end of the range.

A.4 Pressure-Temperature Limit Curves for ISLH Condition

The same procedure applies to the ISLH condition as for the normal operating condition except the safety factor is 1.5 instead of 2. In practice, plants can use the normal condition heatup and cooldown condition curves since they bound the ISLH condition limits. If a soaking period is provided prior to testing, only a steady-state curve is needed without consideration of the transient conditions. If this approach is adopted, the steady-state limit can be used to determine the core criticality limit temperature discussed below.

A.5 Core Criticality Limit

The minimum fluid temperature for required core criticality is determined from ISLH limit curve at a pressure of 2500 psi. Per 10CFR Part 50, Appendix G, fluid temperatures must be also at least 40 F higher than the values specified by the pressure-temperature limit curves for both normal heatup and normal cooldown conditions.

A.6 Technical Specification Limit Curves

The heatup and cooldown limits specified in plant technical specifications are based on the specified maximum heatup or cooldown ramp rates. Since it is difficult for plants to adhere to one specific rate, heatups and cooldowns are usually conducted by operating between conditions of steady-state and the specified ramped transients. If the specified maximum ramp rate is exceeded, a pressure-temperature limit violation occurs, since the thermal stresses used to arrive at the allowable pressure limits are based on the given maximum ramp rate. Therefore, in preparing the limiting pressure-temperature limit (P-T limit) curve, all applicable P-T curves are compared and the lower bounding pressure points are chosen to form the final composite curve. Figure A-9 is an example of a final P-T limit curve. In the formulating such curves, any portion that is characterized by a negative slope (decreasing pressure limit with increasing temperature) is modified by specifying that the minimum value of pressure be maintained until the temperature reaches a value where the pressure is also increasing.

A.6.1. Pressure Sensor Location Adjustment

The pressure in the downcomer region is the controlling pressure for most of the pressure-temperature limit curve, although there is no pressure sensor in this area. The downcomer pressure must therefore be inferred from sensors located in other regions of the reactor coolant system. The difference in pressure between the downcomer and sensor locations, which is a plant specific parameter, can be as high as 100 psi. Such an adjustment for pressure must be included in the final technical specification pressure-temperature limit curve.

A.6.2. Instrument Error Correction

In addition to the pressure sensor location adjustment, temperature-sensing instruments have an error band which is typically 10 to 12 degrees Fahrenheit. The final technical specification pressure-temperature limit curve may also include such a correction for instrument error.

Figure A-1
Typical Heatup Transient

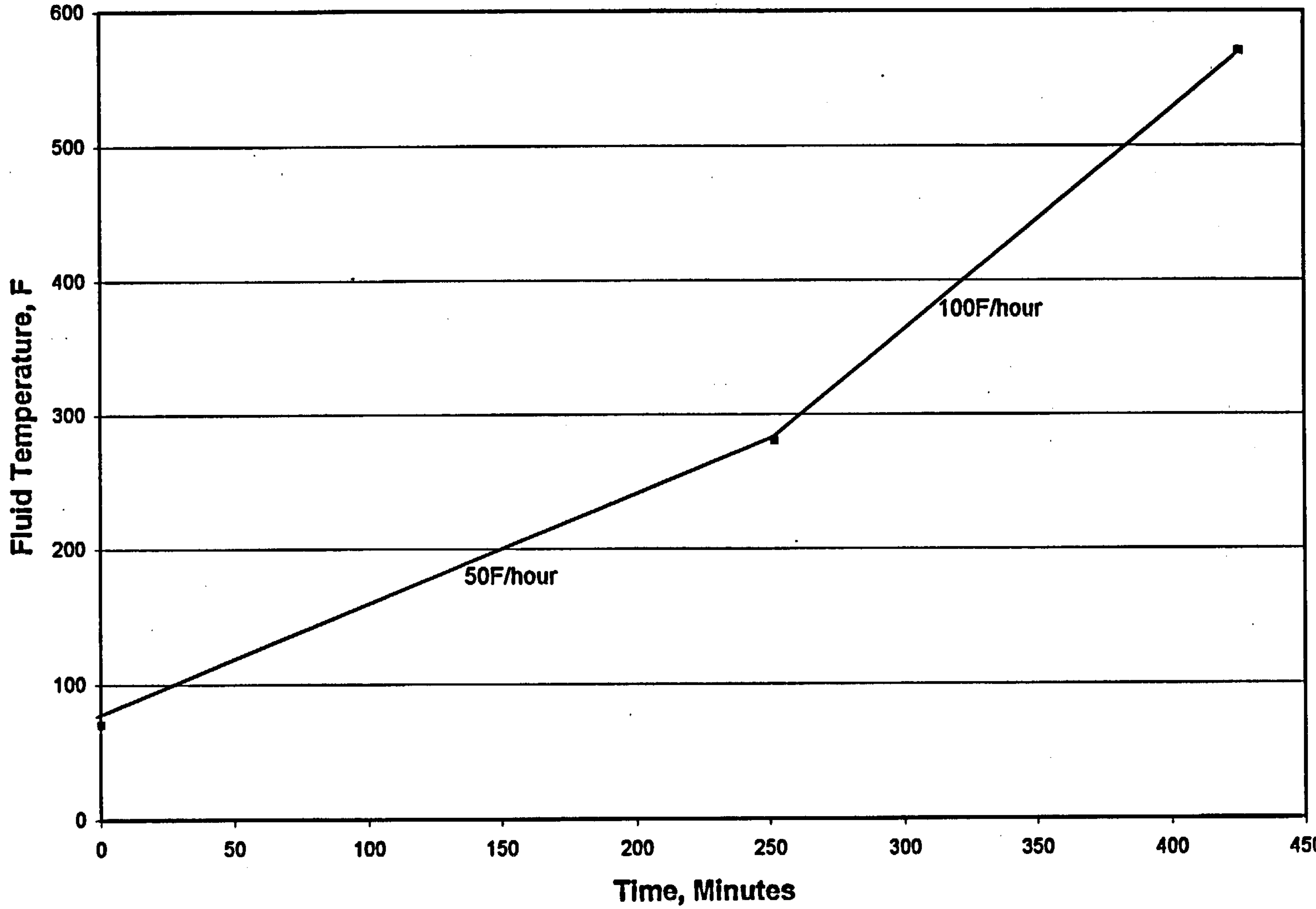

Figure A-2
Typical Cooldown Transient

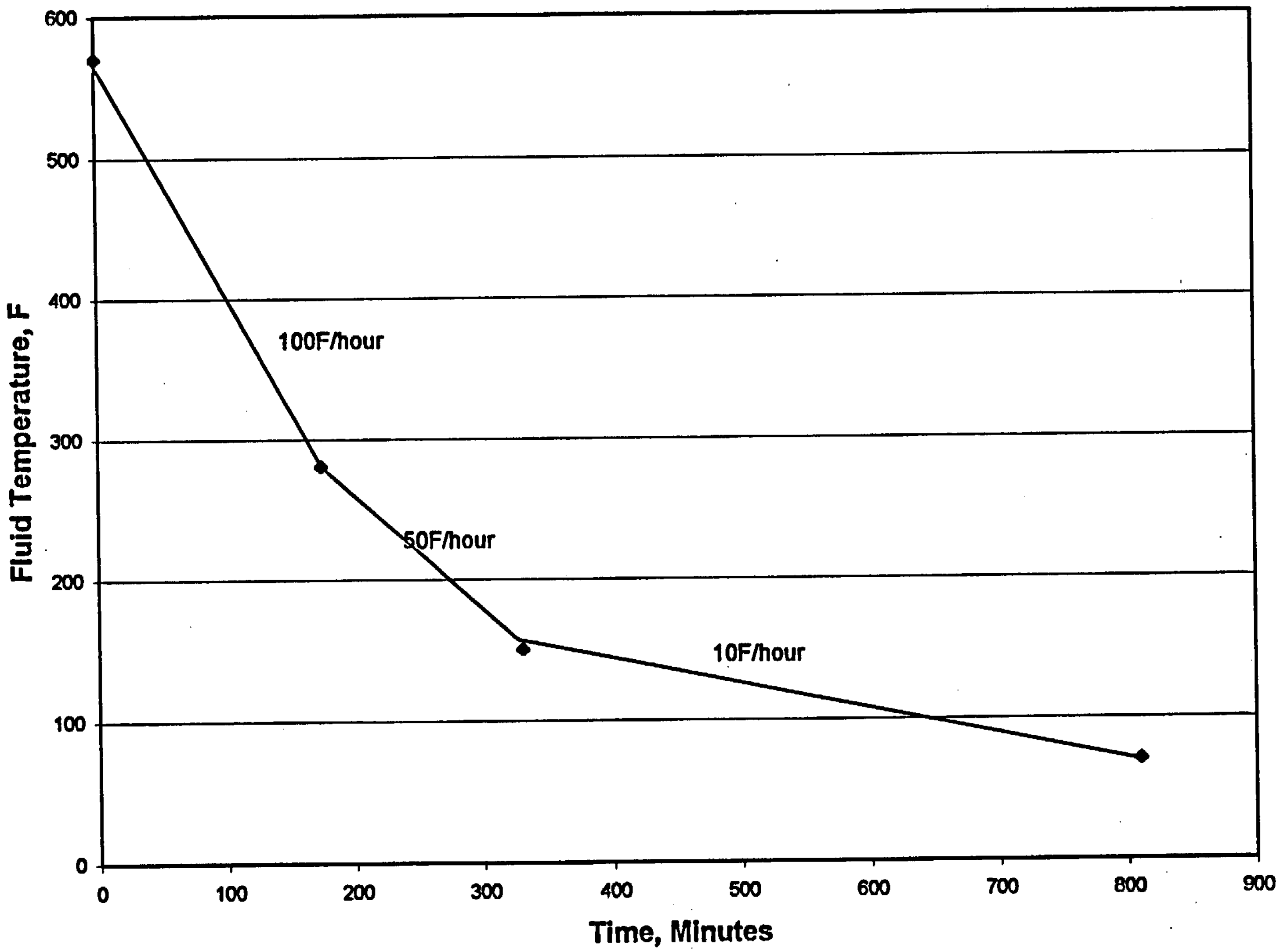

Figure A-3
Closure Head Limits for Normal Operation

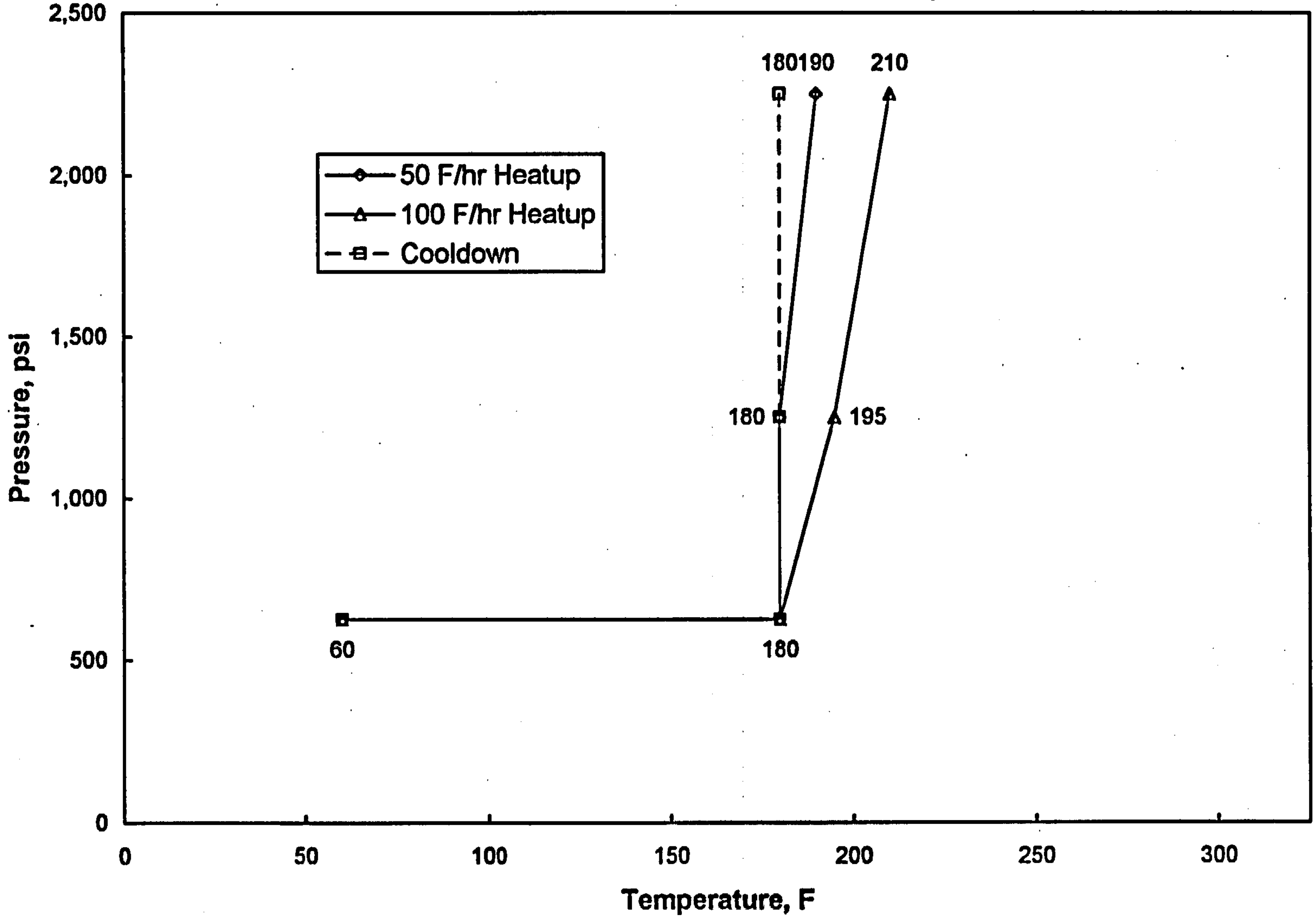

Figure A-4
Normal Operation Heatup Limits for Flaws in Beltline Axial Weld and Outlet Nozzle

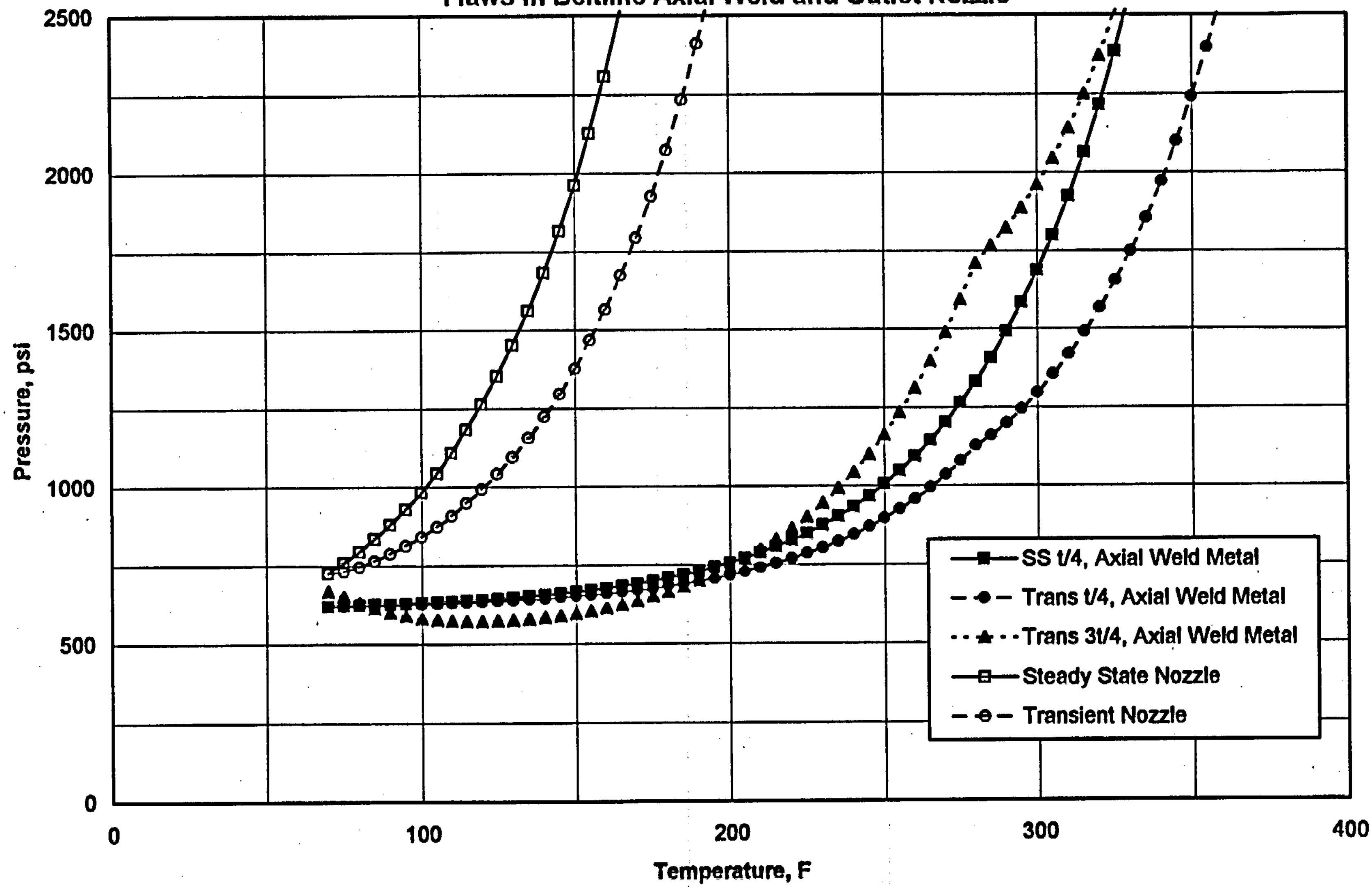

Figure A-5
Normal Operation Cooldown Limits for
Flaws in Beltline Axial Weld and Outlet Nozzle

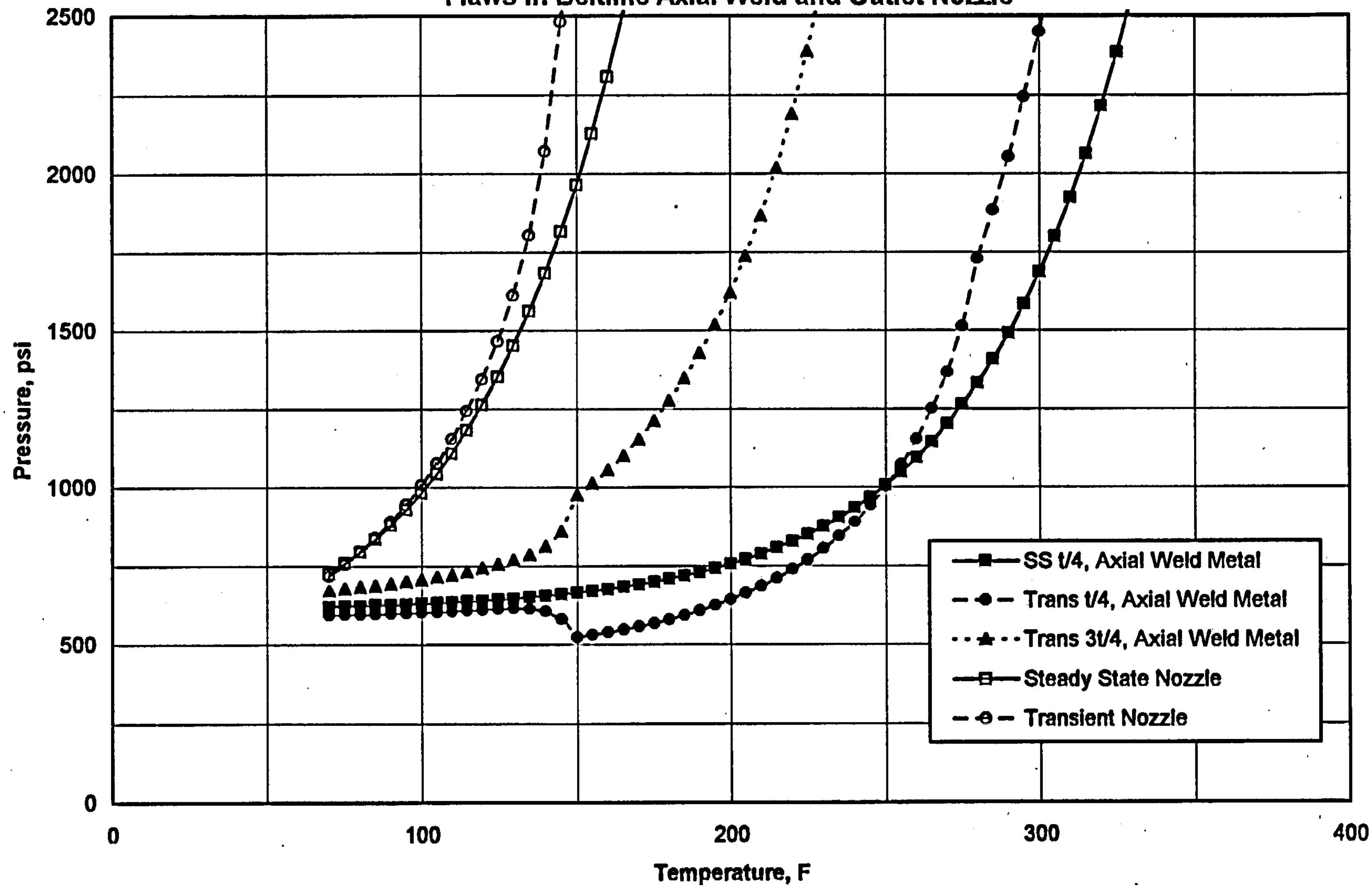

Figure A-6
Normal Operation Heatup Limits
(Axial Weld Controlling)

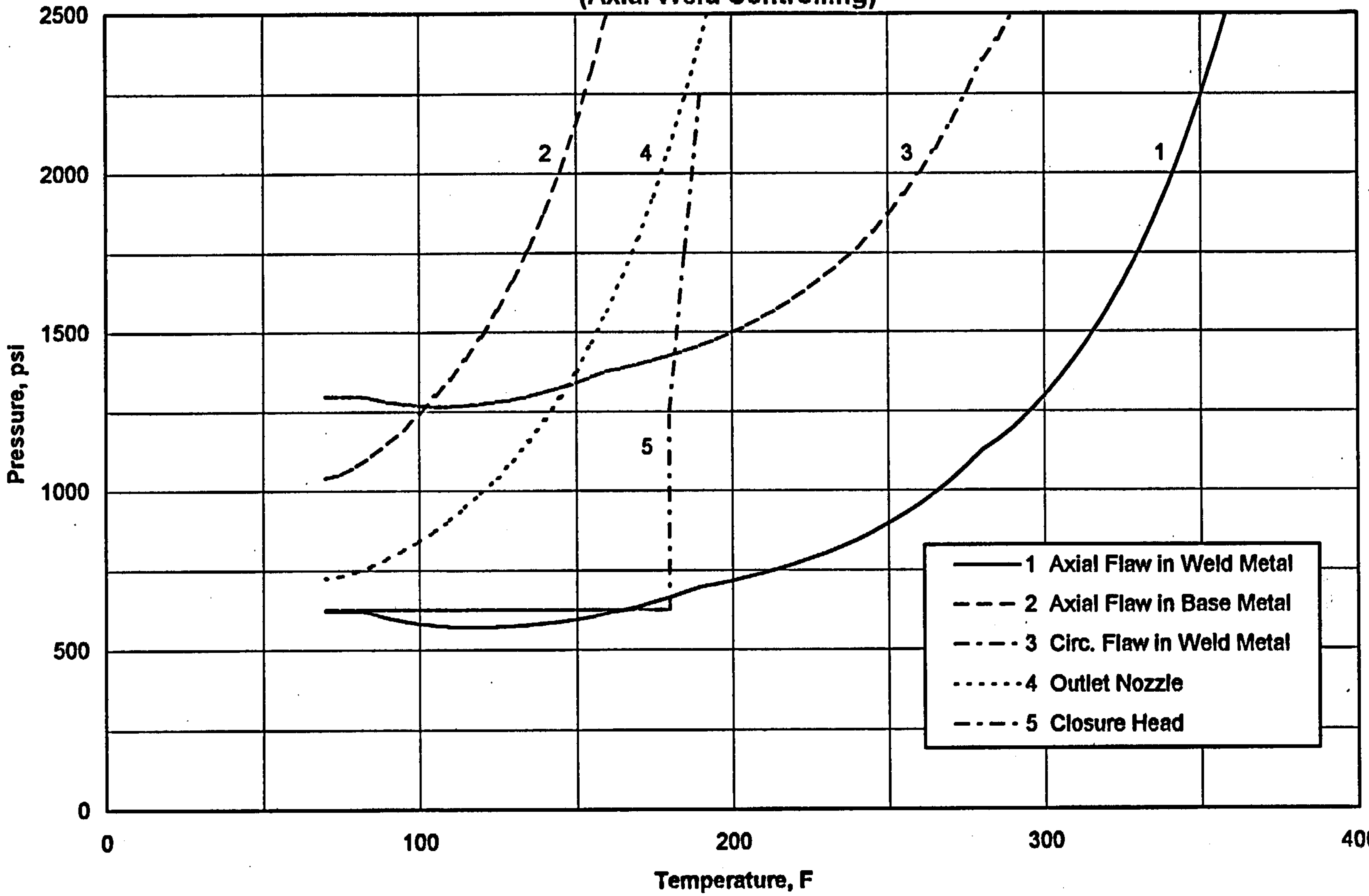

Figure A-7
Normal Operation Cooldown Limits
(Axial Weld Controlling)

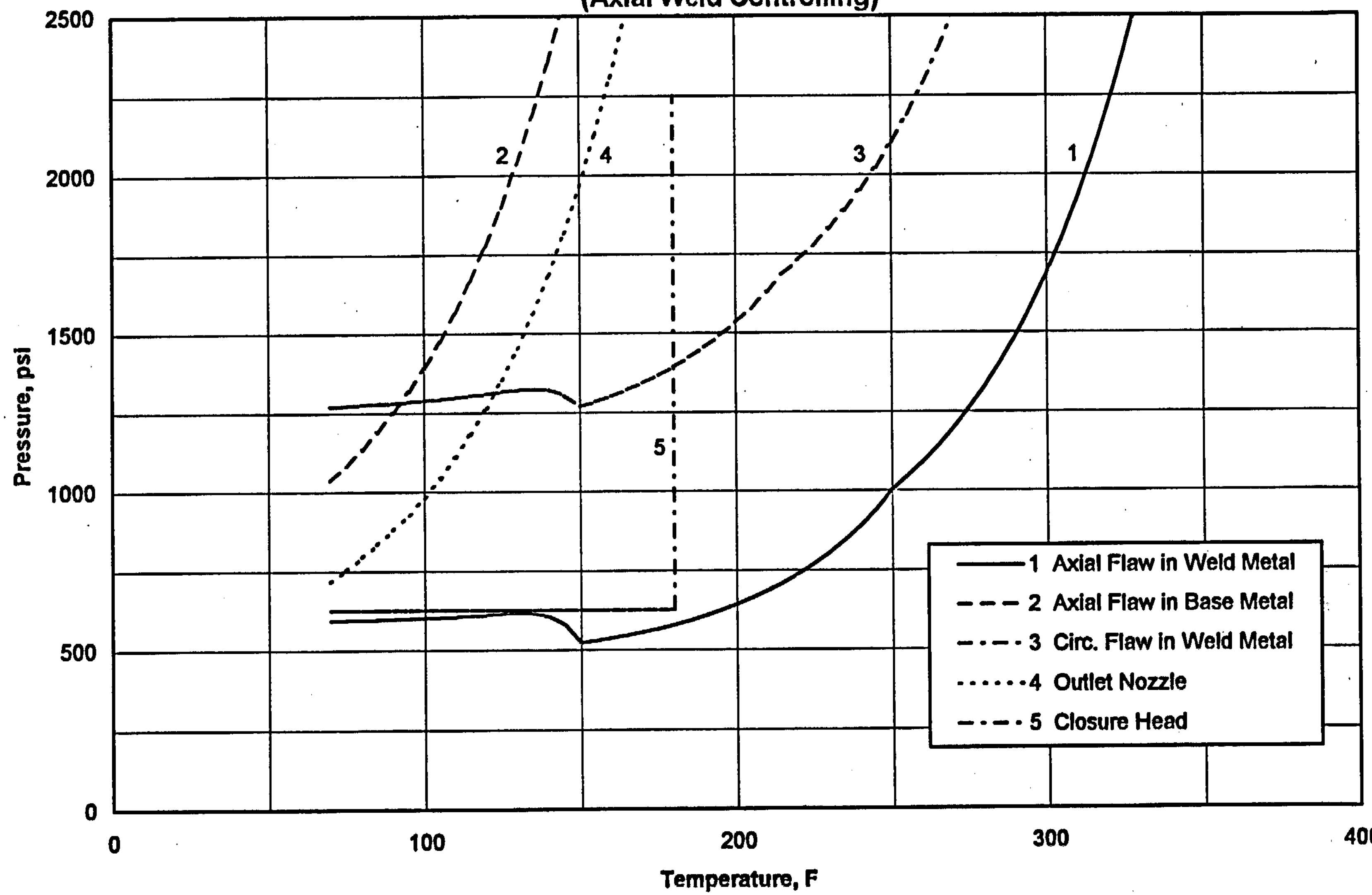

Figure A-8
Normal Operation Heatup Limits
for Vessel with Circumferential Welds Only

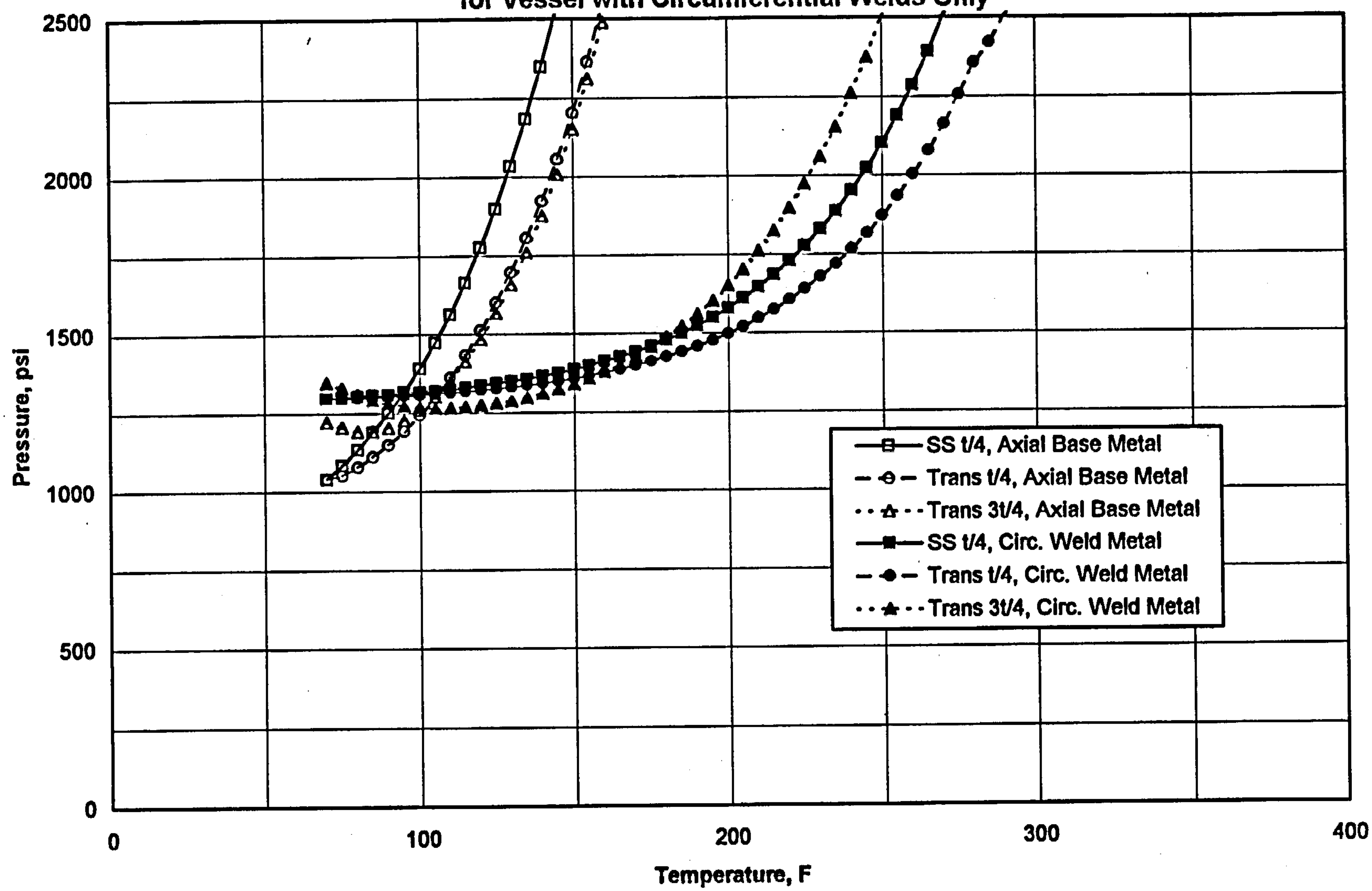

Figure A-9
Techical Specification Heatup Limits