TOLEDO
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Docket No. 50-346

License No. NPF-3

Serial No. 1510

May 4, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Supplemental Information Regarding the License Amendment Request to Revise the Reactor Coolant System Pressure-Temperature Operating Limits and Reactor Vessel Material Surveillance Program (TAC No. 66699)

Gentlemen:

In response to a request made during an April 6, 1988 telephone conversation between Mr. A. W. DeAgazio, the Nuclear Regulatory Commission (NRC)/Nuclear Reactor Regulation (NRR) Davis-Besse Project Manager, and Toledo Edison, additional information is being provided to assist in the review of the License Amendment Request which was submitted to the NRC on March 31, 1988 (Serial No. 1490). This License Amendment Request proposes revising the Reactor Coolant System Pressure-Temperature (P-T) Curves and other related changes necessary to allow operation to ten Effective Full Power Years (EFPY). Additionally, changes to the Reactor Vessel Material Surveillance Program Schedule were requested. Each NRC question, followed by Toledo Edison's response, is provided below:

Question: Provide the Reference Temperature (RT_{NDT}) at 10 EFPY and at End of Reactor Vessel Life (EOL).

Response: The 10 EFPY reactor vessel material properties used in the preparation of the P-T curves are included in Babcock and Wilcox (B&W) Topical Report BAW 2011, "Pressure-Temperature Limits for 10EFPY", November 1987, as referenced by the License Amendment Request submitted in Serial No. 1490. The controlling beltline weld (WF-182-1) RT_{NDT} values are summarized below. The predicted EOL Reference Temperatures from Topical Report BAW 1882, "Analysis of Capsule TE1-A", September 1985, are also listed:

A001
1/1

Docket No. 50-346
License No. NPF-3
Serial No. 1510

Region	10 EFPY Predicted RT_{NDT}(°F)	10 EFPY Assumed RT_{NDT}(°F)	EOL Predicted RT_{NDT}(°F)
Beltline 1/4T	183	147	222
Beltline 3/4T	139	107	182
Closure Head	60	60	60
Outlet Nozzle	60	60	60

<u>Question:</u> Provide the calculations for the Pressure-Temperature (P-T) Curves.

<u>Response:</u> Enclosure 1, B&W Topical Report BAW 10046A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational requirements of Appendix G to 10CFR50", describes the methodology used to calculate the P-T curves. Enclosure 2, B&W Topical Report BAW 2011, "Pressure-Temperature Limits for 10EFPY", describes the results of the calculation. Note that Appendix A of BAW 2011 has not been included since it only presented B&W's recommendation for the wording of Technical Specification 3/4.4.9 and its Bases. Toledo Edison made additional changes to this and related Technical Specifications and the B&W recommendations do not accurately reflect those changes submitted by Serial No. 1490. Therefore, Toledo Edison has elected to omit Appendix A of the enclosed BAW 2011. Each proposed change to the Technical Specifications has been described and justified in the License Amendment Request of Serial No. 1490.

Toledo Edison believes the above addresses the NRC concerns regarding this License Amendment Request. Should there be any additional questions, please contact R. W. Schrauder, Nuclear Licensing Manager at (419) 249-2366.

Very truly yours,

DRB/tlt

Attachments

cc: DB-1 NRC Resident Inspector
A. B. Davis, Region III Regional Administrator
A. W. DeAgazio, NRC/NRR Davis-Besse Project Manager
State of Ohio

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

APR 3 0 198

Babcock and Wilcox Company
ATTN: James H. Taylor
Manager, Licensing
Nuclear Power Generation
P.O. Box 10935
Lynchburg, Virginia 24506-0935

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT BAW-10046, REV. 2 B&W OWNERS GROUP MATERIALS COMMITTEE "METHODS OF COMPLIANCE WITH FRACTURE TOUGHNESS AND OPERATIONAL REQUIREMENTS OF 10 CFR 50, APPENDIX G"

We have completed our review of the subject topical report submitted by Babcock and Wilcox (B&W) by letter dated December 21, 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that B&W publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, B&W and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
for Technical Support
Division of PWR Licensing-B

Enclosure: As stated

8605080065 3pp

ENCLOSURE

SAFETY EVALUATION OF TOPICAL REPORT
BAW-10046, Rev. 2, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G"

SUMMARY OF REPORT

This Topical Report was submitted by letter of December 21, 1984 to Mr. Cecil O. Thomas, Chief, Standardization and Special Projects Branch. It is an update of BAW 10046A, Rev. 1 issued in July 1977, which was evaluated and accepted for referencing in licensing applications by letter of June 22, 1977. There are two principal changes in Revision 2: (1) additions and changes made to reflect the revised criteria in the amendments to Appendix G, 10 CFR 50, effective July 26, 1983, and (2) a new Chapter 6 that was added to describe the method of analysis and the material properties data to be used when a more advanced ductile fracture analysis is called for by evidence that the Charpy upper shelf energy of the reactor vessel beltline material has fallen below 50 ft lb. Specifically, Chapter 6 was intended to comply with the requirement for continued operation given in paragraph V.C of Appendix G, 10 CFR 50.

The initial review of BAW 10046, Rev. 2 prompted a number of requests for explanation of the methods described in the report. The explanations were given in BAW 1868, March 1985 entitled "BAW 10046A Rev. 2 Supplement." However, only BAW 10046, Rev. 2, which contains the requirements and criteria, is being accepted for referencing in licensing applications. Another round of questions and comments in December 1985 was answered at a meeting on January 8, 1986 followed by a submittal of proposed additions and corrections by letter of April 24, 1986 to H. Denton. In this evaluation it is assumed that these changes will be made in the final report.

Chapters 1 through 5 of Revision 2 of BAW 10046 cover the same material as did Revision 1. They give an introduction (Chapter 1), discuss operating modes such as bolt up, heat up, cool down, operation, and testing (Chapter 2), describe the measurement of the material properties pertinent to fracture toughness and to radiation damage of the reactor vessel beltline (Chapter 3), show how B&W applies ASME Code and NRC requirements in the generation of pressure temperature (P-T) limits (Chapter 4), and give examples of P-T limits for 5 and 32 EFPY (Chapter 5). Chapter 6, as described above, is new.

REGULATORY EVALUATION OF REPORT

We have reviewed the B&W method of calculation of pressure - temperature limits against the requirements of Appendix G, 10 CFR 50, Regulatory Guide 1.99, and the Standard Review Plan. Special attention was paid to the updates made in compliance with changes in Appendix G, 10 CFR 50 that became effective July 26, 1983. These concerned consideration of the closure flange regions as potential controlling locations for the P-T limits, the hydrotest temperature requirement when there is no fuel in the reactor, and the more specific requirements for consideration of reactor vessel integrity when the beltline material exhibits low upper shelf behavior. Assuming the final report is edited as indicated in the submittal received April 24, 1986, Chapters 1-5 are satisfactory with the following comment on paragraph 3.1.3. Chapter 6 is discussed separately .

3.1.3 Radiation Effects

The B&W report states that the methodology used to adjust the RT_{NDT} values as used in developing pressure-temperature limits will be "the currently accepted procedures." A similar statement is made concerning the decrease in Charpy upper shelf energy. As further stated in the report, the decision on these issues is simply passed to the utility, to be made when a submittal is made to the NRC. It should be noted that Revision 2 of Regulatory Guide 1.99, which addresses these topics, was issued for public comment in February 1986. When issued in final form, it will be the basis for review of subsequent submittals, but if a utility chooses to use Rev. 2 as the basis for a submittal before that date, it will be evaluated on the basis of Rev. 2. The Guide covers the use of plant surveillance data as well as calculative procedures to be used when there are no credible plant surveillance data.

As a cautionary note, alternatives to the use of either plant surveillance data or procedures that derive from analysis of a broad data base are difficult to justify; because significant scatter in these data gives rise to the possibility that a small subset of the data base will not give representative values of the mean and the uncertainty pertinent to the vessel in question.

Chapter 6 EPFM Analytical Procedures

This chapter describes a method of fracture analysis for low upper shelf (below 50 ft lb) material. Paragraph V.C of Appendix G requires such an analysis to demonstrate that the margin of safety is equivalent to that required by Appendix G of the ASME Code. Chapter 6 has been reviewed against the guidelines for resolution of this issue given in NUREG-0744*, supplemented and in some cases superseded by the ongoing effort of the Working Group on Flaw Evaluation (WGFE) of the Subcommittee on Nuclear Inservice Inspection of the ASME Boiler and Pressure Vessel Committee. The report of the WGFE, "Development of Criteria for Assessment of Reactor Vessels With Low Upper Shelf Fracture Toughness" is still in preparation. Actually, the technology involved in both the analytical effort and the materials testing is still developing, as the foregoing illustrates. Thus, this evaluation must be regarded as an interim one, subject to change as further developments occur.

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* R. Johnson, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue, Vols. 1 & 2, October 1982.

The method of analysis used in BAW 10046, Rev. 2, meets the recommendation of NUREG-0744 that the analysis should use a J-integral formulation and that the material toughness should be characterized by J-R curves. In addition, the B&W analysis meets the WGFE crack stability criteria:

$$\frac{d\,J_{applied}}{da} < \frac{d\,J_{material}}{da}$$

at $J_{applied} = J_{material}$

The B&W analysts have their own way of solving these simultaneous criteria, but it appears in the WGFE criteria document and it gives the same result as the other methods given there and is therefore acceptable. To resolve any question about the physical meaning of the mathematical procedures (at the request of NRC staff), the B&W authors included a figure (Fig. 6-5) that summarizes the results by giving the pressures to cause crack initiation and tearing instability and the predicted amounts of crack growth at which initiation and instability occur. The same figure also shows the pressure to cause plastic instability as a function of crack size.

The acceptance criteria for normal and upset conditions given in BAW 10046, Rev. 2, meet the criteria given in the WGFE draft report that margin should be based on load (not J) and that separate requirements should be given for crack stability and crack initiation. The instability criterion given in the WGFE draft report is as follows:

> "The postulated crack* shall be demonstrated to be stable under ductile crack growth with a factor of safety of two on pressure and a factor of one on thermal loading for all service level A and B conditions except for hydrostatic tests. For hydrostatic tests, the factor of safety shall be 1.5 on pressure and a factor of one on thermal loading."

This meets the requirement of Appendix G, 10 CFR 50, that margins against fracture should be equivalent to those required by Appendix G of the ASME Code. The latter requires for the determination of allowable pressure during Service Conditions for which Level A and Level B Service Limits are specified, use of the following formula:

$$2\,K_{IM} + K_{IT} < K_{IR}$$

The quantity K_{IM} (membrane) is proportional to pressure. This is the origin of the factor of 2 on pressure. Thus, it seems clear that the acceptance criteria in BAW 10046, Rev. 2 should refer to Level A and B Service Conditions, rather than to operating pressure. The effect of the B&W criterion is that crack stability under ductile tearing conditions must be shown for a pressure of 4500 psi (2 x operating pressure) whereas the WGFE criterion, which meets the requirements of Appendix G, 10 CFR 50, could require

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* An ASME Section III, Appendix G flaw (a semi-elliptical surface flaw with the depth equal to one-quarter of the wall thickness and length six times the depth).

stability at as much as 5500 psi pressure, if the Design Specification designated a Level B service loading that produced a pressure stress of 110 percent of the design stress intensity value, S_m.

The B&W criteria omit thermal stresses on the grounds that their effect is small and the methods used for calculation of J applied due to thermal stress are controversial. It is agreed that for the Service Level A and B conditions where ductile tearing stability is an issue the effects of thermal stress are small; hence, the omission of thermal stress is acceptable.

With regard to crack initiation, the criterion in BAW 10046, Rev. 2 is that the crack initiation pressure must exceed 3000 psi where the J value for crack initiation is defined as $J_{material}$ at 0.01 inch crack growth. This is conservative, compared to the WGFE criterion, which is based on 2750 psi and one millimeter (0.04 inch) crack growth.

In summary, Chapter 6 of the Topical Report gives a satisfactory method of analysis of the resistance to ductile tearing instability and plastic instability of the reactor vessel beltline with the exception that the acceptance criterion for instability should be for all Level A and B loadings not just operating pressure.

REGULATORY POSITION

Topical Report BAW 10046, Rev. 2 describes acceptable methods for the development of allowable pressure - temperature limits for normal operation and for test conditions to assure the prevention of non-ductile fracture. It may be referenced in future applications for setting these limits in Technical Specifications. It is understood that the report dated December 1984 will be edited per the submittal of corrections and additions to the text, submitted by letter to H. Denton, dated April 24, 1986.

Topical Report BAW 10046, Rev. 2 also describes acceptable methods for the analysis and materials properties data required to demonstrate resistance of the reactor vessel beltline to ductile tearing instability when the Charpy upper shelf energy of the beltline materials falls below 50 ft lb. It may be referenced in license submittals made in conformance to the requirements of Appendix G, 10 CFR 50, except the acceptance criteria should be for all Level A and B loadings. It should also be noted that the technology for treatment of ductile tearing instability is less mature than for example that for non-ductile fracture; hence, future revision of this requirement may be expected.

Babcock & Wilcox
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Nuclear Power Division
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April 24, 1986
JHT/86-075

Mr. Harold Denton
Executive Director of Operations
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: B&W Topical Report BAW-10046, Rev. 2, dated December 1984, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G.

References: Letters, J. H. Taylor to C. O. Thomas, dated December 21, 1984 and April 25, 1985.

Attachment: Additions and Clarifications to BAW-10046, Rev. 2.

Dear Sir:

In order to expedite the NRC's review and approval of the subject report, the attached supplemental information requested by the Staff's Dr. P. N. Randall is being forwarded to you. It is intended that this information will be included in the approved version of BAW-10046, Rev. 2.

Your immediate approval of the subject report is requested by Babcock & Wilcox, the B&W Owners Group, and the following Utilities:

Arkansas Power & Light Co.	GPU Nuclear Corporation
Duke Power Company	Sacramento Municipal Utility Dist.
Florida Power Corporation	Toledo Edison Company

If you should have any questions, please do not hesitate to call.

Very truly yours,

G J Hucka for

J. H. Taylor
Manager
Licensing Services

JHT/leh

Attachment

~~8604280158~~ 3pp

cc: P. N. Randall
W. Paulson
B. J. Elliot

<u>B&WOG Materials Committee</u>

D. F. Spond	-	AP&L
M. A. Haghi	-	DPCo
D. N. Miskiewicz	-	FPC
R. L. Miller	-	GPUN
S. W. Rutter	-	SMUD
D. R. Cox	-	SMUD
R. J. Gradomski	-	TED

The following additions and corrections are proposed for BAW-10046 Rev. 2

Para 4.2.1.1 (New Paragraph)

6. Appendix G of the ASME Code Section III recommends the temperature of the closure area be RT_{NDT} at bolt-up. The forgoing procedure yields similar results with the exception of low stressed closures. This procedure is considered consistent with the philosophy of the ASME Code and will be used for establishing temperature requirements.

Para. 4.2.1.2 (New paragraph)

3.C 10CFR50 Appendix G Paragraph IV.A.2 requires the highly stressed regions of the closure region to be at a temperature of at least RT_{NDT} + 120°F for pressures above 625 psig. The forgoing procedure results in a similar temperature requirement. The required temperature is lower than 120°F if slow heat-up rates are specified and higher than 120°F for the operating pressure condition and maximum heat-up rates. The forgoing procedure is considered to be consistent with the requirements of 10CFR50 Appendix G and is used in lieu of the stated requirement.

Para. 4.2.3.2 (Add at end of paragraph)

The requirement of 10CFR50 Appendix G specifying a temperature of RT_{NDT} + 90°F for highly stressed regions of the closure for pressures above 625 psig is essentially met by this procedure. As for the normal heat-up case higher or lower temperatures may be required depending on heat up rate.

Table 4-1

Revise materials property RT_{NDT} location for closure head to 1/4t.

Table 5-1

Revise line (K.) in chemistry column under S, 0.61 should be .016. Also for material B, revise initial RT_{NDT} to +30.

Table 5-2

Revise table for typographical errors first heading RT_{NDT} should be ΔRT_{NDT}. For MAT ID B 260 should be 250.

Page 6-1

Change "preventation" to "prevention" and change J_{IR} to J_I-R.

Page 6-5

Revise paragraph 6.2.5 to 6.3. In paragraph 6.3 second paragraph is revised "as well as the local" to "as well as the local plastic instability pressure calculated by the ratio". Table 6-12 should be Table 6-1.

Revise page 6-8 as follows: In Figure 6-3 H1-6.9 should be H1 = 6.9 and lower table should be labeled Table 6-1.

THE B&W OWNERS GROUP

Arkansas Power & Light Company — ANO-1
Duke Power Company — Oconee 1, 2, 3
Florida Power Corporation — Crystal River 3
GPU Nuclear Corporation — TMI-1
Sacramento Municipal Utility District — Rancho Seco

Toledo Edison Company — Davis Besse
Tennessee Valley Authority — Bellefonte 1, 2
Washington Public Power Supply System — WNP 1
Babcock & Wilcox Company

Working Together to Economically Provide Reliable and Safe Electrical Power

April 25, 1985

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(301) 951-3344

Mr. Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: B&W Report, BAW-10046, Rev. 2, dated December 1984, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G."

References: 1) Letter, N. P. Kadambi to E. C. Simpson, dated February 20, 1985.

2) Letter, J. H. Taylor to C. O. Thomas, dated December 21, 1984.

Attachments: 1) B&W Report, BAW-1868, "BAW-10046A, Rev. 2, Supplement," dated March 1985.

2) B&W Report, BAW-1814, "Analysis of HSST Intermediate Vessel V-8A Test by the Deformation Plasticity Failure Assessment Diagram Method," dated November 1983.

Dear Mr. Thomas:

The attached reports are being submitted on behalf of the B&W Owners Group at the request of your Dr. P. N. Randall to facilitate the review and approval of the subject report. BAW-10046, Rev. 2, was submitted by Reference 2 and it is understood that the review is still scheduled to be complete by June 1985 (Reference 1).

Attachment 1 contains additional information in support of BAW-10046, Rev. 2, including details on the deformation plasticity failure assessment diagram (DPFAD), calculation of the Ramberg-Osgood stress-strain relationship and the reactor vessel closure analysis.

Attachment 2 presents a DPFAD analysis of the NRC sponsored test of HSST vessel V-8A which serves to benchmark the analytical approach to experimental results.

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Should you have any questions or comments, please contact our Mr. C. J. Hudson (804-385-2550).

Very truly yours,

C. J. Hudson FOR

J. H. Taylor
Manager
Licensing Services

JHT/leh

Attachments

cc: B. J. Elliot
P. Kadambi
D. Moran
P. N. Randall
G. Vissing

B&WOG Materials Committee

D. F. Spond - AP&L
M. A. Haghi - DPCo
R. A. Webb - FPC
J. A. Janiszewski - GPUN
D. R. Cox - SMUD
S. W. Rutter - SMUD
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December 21, 1984
ESC-852

Mr. Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: B&W Topical Report BAW-10046, Rev. 2, dated December 1984, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, App. G.

Dear Mr. Moran:

Enclosed are ten (10) copies of the subject report which is being submitted on behalf of the B&W Owners Group and the following Utilities:

Arkansas Power & Light Company
Duke Power Company
Florida Power Corporation
GPU Nuclear Corporation
Sacramento Municipal Utility District
Toledo Edison Company

BAW-10046, Rev. 1 has previously been approved by the NRC. Revision 2 is being submitted as required by 10CFR50, Appendix G, Paragraph IV.A.1 which states that the Director, Office of Nuclear Reactor Regulation must approve the manner in which "equivalent margins of safety" are provided for reactor vessel beltline materials which do not maintain 50 ft-lbs of Charpy Upper Shelf Energy. That manner is described in BAW-10046, Revision 2.

The method of Elastic-Plastic fracture mechanics known as the Failure Assessment Diagram has been under development for some time. The submittal of BAW-10046, Revision 2 culminates years of work by the B&W Owners Group and the Babcock and Wilcox Company. Approval of the methodology contained in the Topical Report is a key milestone in the B&W Owners Group Reactor Vessel Materials Program which has been underway since 1976. In order to assure continued compliance with Federal Regulations, NRC approval of this document is requested by May 1, 1985.

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If you should have any questions please contact Mr. C. J. Hudson (804) 385-2550 or Mr. H. W. Behnke (804) 385-2417.

Very truly yours,

CJ Hudson For

J. H. Taylor, Manager
Licensing

JHT/mct

cc: W/Attachment

D. Moran
B. J. Elliot
W. S. Hazelton
P. N. Randall
P. Kadambi
R. Johnson

B&W Owners Group Materials Committee

D. F. Spond	- AP&L
P. Guill	- DPCo
R. A. Webb	- FPC
J. A. Janiszewski	- GPUN
S. W. Rutter	- SMUD
R. J. Gradomski	- TED
E. C. Simpson	- FPC

Docket No. 50-346
License No. NPF-3
Serial No. 1510
Enclosure 1

BAW-10046A, Rev 2

Topical Report
June 1986

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

by

H. W. Behnke
A. L. Lowe, Jr.
J. M. Bloom
W. A. Van der Sluys

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Topical Report BAW-10046A, Rev 2

June 1986

Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G

H. W. Behnke, A. L. Lowe, Jr., J. M. Bloom, W. A. Van der Sluys

Key Words: Ferritic Materials, Reactor Coolant Pressure Boundary, Reference Temperature, Charpy Upper Shelf Energy, Appendix G to 10 CFR 50, Appendix G to ASME Code, Fracture Prevention, Pressure Temperature Limitation, Technical Specifications, Ductile Tearing Instability, Elastic-Plastic Fracture Mechanics, Deformation Plasticity Failure Assessment Diagram

ABSTRACT

This report describes B&W's practices, methods, and criteria for compliance with the requirements of Appendix G to 10 CFR 50, "Fracture Toughness Requirements." The ferritic materials and the operational parameters of the reactor coolant system for nuclear power plants designed by B&W are described as are the methods for obtaining and estimating the reference temperature and the Charpy upper shelf energy. The acceptance criteria for unirradiated Charpy upper shelf energy is given. The adequacy of fracture toughness properties of bolting materials and type 403 materials are demonstrated. The methods employed to determine the reactor coolant system pressure-temperature limit curves are given for each of the loading conditions required by Appendix G to 10 CFR 50. The pressure-temperature limit curves imposed by several regions of the reactor vessel are illustrated as is the development of the composite limit curves. Furthermore this report describes the methods used to preclude ductile tearing instability. This analysis applies to irradiated vessels with low upper shelf energies. The Technical Specifications pressure-temperature limit curves and the Preservice System Hydrostatic Test limit curve of a typical 177 FA plant are also described.

CONTENTS

List of Tables

List of Figures

Figures (Cont'd)

1. INTRODUCTION

1.1. Background

On July 17, 1973, a new appendix to 10 CFR 50, entitled "Appendix G - Fracture Toughness Requirements" was published in the Federal Register. 10 CFR 50, Appendix G has been revised in subsequent years. This report reflects the revised criteria including effective issue July 26, 1983. This 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. Although the requirements of Appendix G became effective August 13, 1973, they are applicable to all boiling water and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

At the time 10 CFR 50, Appendix G, became effective, immediate compliance with some of its provisions was not possible for plants whose pressure boundary components were ordered in accordance with an edition or addenda of Section III of the ASME Boiler and Pressure Vessel Code (hereafter ASME Code) published before the Summer 1972 Addenda. For these plants, neither the fracture toughness data required by Appendix G nor the material for performing toughness tests is available. Also, the stress calculations required to quantitatively define the allowable pressure at any given temperature were not readily available. Appropriate, conservative methods of compliance for these plants have been developed and are described in this report.

1.2. Scope and Organization

This report presents B&W's practices, methods, and criteria for compliance with the requirements of 10 CFR 50, Appendix G. It is applicable to all current B&W nuclear steam systems (NSSS). The definitions and terminology of 10 CFR 50, Appendix G, and the ASME Code are used whenever appropriate.

The report is divided into seven parts and is summarized in Part 7. Part 2 describes the reactor coolant pressure boundary (RCPB) and includes a list of the components and ferritic materials used in their construction. Part 2 also describes the operational modes of the RCPB related to nonductile failure for each of the loading conditions for which pressure-temperature limit curves are required.

Part 3 presents the fracture toughness properties of the ferritic materials of the RCPB. These materials are grouped as follows:

1. Ferritic materials other than (a) bolting and (b) type 403 stainless steels
2. Bolting materials
3. Type 403 stainless steel

For the first group, Part 3 describes methods for (1) determining the unirradiated reference temperature (RT_{NDT}) for the ferritic materials and the unirradiated Charpy upper shelf energy (C_vUSE) level of the beltline region materials. The justification for use and acceptance criteria for unirradiated beltline region materials with C_vUSE lower than 75 ft-lbs are presented.

For the second group, bolting materials, Part 3 presents justification for allowing the lowest service temperature, and the minimum preload temperature to be 40F. The impact properties of these materials are also presented.

For the third group of materials, Part 3 includes a demonstration of adequate fracture toughness properties.

Part 4 presents the basis for a step-by-step description of the calculational procedure to determine the pressure-temperature limitations of the reactor coolant system; this is done to ensure adequate fracture toughness under the loading conditions of interest.

Part 5 gives an example of beginning- and end-of-life pressure-temperature limit curves that were developed using the material properties in Part 3 and the calculational procedure of Part 4. Similar curves were developed for each plant and conservatively adjusted for use in the Technical Specifications issued by the Nuclear Regulatory Commission (NRC) as a part of the plant operating license. Typical limit curves, as they appear in the Technical Specifications, and the limit curve for the preservice system hydrostatic test are shown in Part 5.

Part 6 presents the supplemental analysis performed in the event a reactor vessel beltline is predicted to be below 50 ft-lbs upper shelf. This analysis is an elastic plastic fracture mechanics assessment confirming that the vessel has sufficient toughness to preclude ductile tearing instability.

2. REACTOR COOLANT PRESSURE BOUNDARY

2.1. Components

The RCPB is defined by NRC Regulation 10 CFR 50.2, (v) as follows:

> "'Reactor coolant pressure boundary' means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessel, piping, pumps, and valves, which are:
>
> (1) Part of the reactor coolant system, or
>
> (2) Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
>
> (iii) The reactor coolant system safety and relief valves.
>
> For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping."

The reactor coolant system (RC system) for B&W nuclear power plants is 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 core flooding injection piping and valves; the vent, drain, and other piping and valves used for maintaining the RC system; and the incore instrumentation on 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, which reads as follows:

Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary defined in 50.2(v) need not meet these requirements, provided:

(a) In the event of postulate failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only, or

(b) the component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

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 in some instances the core flood tanks, which are carbon steel in some B&W 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.

2.2. Ferritic Materials and RCPB Operational Parameters

The ferritic materials used in construction of the RCPB for B&W nuclear power plants are listed for each component in Table 2-1. The pressure boundary of the RC system is fabricated primarily from ferritic materials, while that of the auxiliary systems is fabricated primarily from austenitic material.

Consequently, the RC system components are the only ones that require special protection against nonductile failure and that must comply with the fracture toughness requirements of 10 CFR 50, Appendix G. This protection against nonductile failure is ensured by imposing pressure-temperature limitations on 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. Normal operations including bolt preloading, heatup and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic tests.
4. Reactor core operation.

To impart a better understanding of the required protection against nonductile failure, typical operational parameters of the RC system are described in the following paragraphs for each of the loading conditions.

2.3. Normal Operation

2.3.1. Bolt Preload

During bolt preload, the reactor vessel closure studs are tensioned to the specified load. Bolt preloading is not allowed until the reactor coolant temperature and the volumetric average temperature of the closure head region (including the studs) is 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 net positive suction head (NPSH) required for RC pump operation. The heatup transient begins when the RC pumps are started.

2.3.3. Heatup

During heatup the RC system is brought from a cold shutdown condition to a hot shutdown condition. The heat sources used to increase the temperature of the system are the RC pumps and any residual (decay) heat from the core. Normally, when the pumps are started, the temperature of the water in the pressurizer is about 400F; this corresponds to the pressure in the RC system of about 300 psig. The coolant temperature is at or 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 130F for subsequent refueling.

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

2.3.3. Cooldown

RC system cooldown brings the system from a hot to a cold shutdown condition. The cooldown is normally accomplished in two phases. The first phase reduces the fluid temperature from approximately 550F to below the design temperature of the decay heat removal system (approximately 300F). This temperature reduction is accomplished using the steam generators but bypassing the turbine and dumping the steam directly to the condenser. Once below its design temperature (and pressure), the decay heat removal system (DHRS) is activated in the second phase to further reduce the reactor coolant temperature to that desired.

Before cooldown, the RC system temperature is maintained constant by balancing the heat removal rate from the steam dump with the heat contributed by the RC pumps and core decay heat. The system pressure is maintained by the pressurizer. The cooldown is normally initiated by stopping one RC pump in each loop. The two remaining pumps provide coolant circulation through both steam generators, and the turbine steam bypass flow controls the cooldown rate. The primary pressure during cooldown is controlled with the pressurizer heaters and spray. After cooling down below the DHRS design temperature and pressure, the cooling mode is changed from the steam generators to the DHRS. Before the switch, the RC system pressure is below 625 psig (20% of the preoperational system hydrostatic test pressure) and below the DHRS pressure but above the pressure required for the RC pumps to operate.

To minimize the thermal shock on the RCPB, the two RC pumps remain in operation as the water flow of the DHRS is initiated. The DHRS flow rapidly mixes with the reactor coolant, but during this period, the indicated RC temperature may fluctuate until mixing is complete. After the switch is completed, the RC pumps are stopped. During this phase, the cooldown rate is controlled by the temperature and flow of the DHRS.

2.4. 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. This 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 NPSH required for the RC pumps 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 the examination for leakage follows in accordance with the ASME Code.

2.5. Inservice System Leakage and Hydrostatic Tests

When inservice system leakage tests are required, the system is brought from a cold to a hot shutdown condition. 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. For the inservice hydrostatic test, ASME Section XI gives a table (Table IWB-5222-1) of the minimum test pressure versus the test temperature at which the system must be tested. The test temperature for both the inservice leakage and hydrostatic tests is determined by the requirements for fracture toughness.

2.6. Reactor Core Operation

The reactor core is not allowed to become critical until the RC system fluid temperature is above 525F except for brief periods of low-power physics testing. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

Table 2-1. Ferritic Materials Used in Reactor Coolant Pressure Boundary

Component	Material
Reactor Vessel	
Plates	SA 533, Grade B, Class 1
Forgings	SA 508, Class 2; SA 182, Grade F6
Bolting	SA 540, Grade B-23 or -24
Welds	SFA 5.5, SFA 5.17
Bars	A276 Type 403 (Code Case 1337 or N-4)
Steam Generator	
Plates	SA 533, Grade B, Class 1; SA 516, Grade 70
Forgings	SA 508, Class 1
Bolting	SA 540, Grade B-23 or -24
Welds	SFA 5.5, SFA 5.17
Pressurizer	
Plates	SA 533, Grade B, Class 1
Forgings	SA 508, Class 2
Bolting	SA 540, Grade B-23; SA 320, Grade L43
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Piping	
Plates	SA 516, Grade 70
Forgings	SA 105, Grade 2
Seamless Pipe & Tubing	SA 106, Grade C
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Pump	
Forgings	SA 508, Class 2; SA 350, Grade LF2
Bolting	SA 540, Grades B-21, -23, -24
Valves	
Forgings	SA 105 Grade 2

3. MATERIAL PROPERTIES

3.1. Impact Properties of Ferritic Materials

To determine the pressure-temperature operating limitations for the RCPB the reference nil-ductility temperature (RT_{NDT}) of the ferritic materials must be established. The RT_{NDT} is needed to calculate the critical stress intensity factor (K_{IR}). In ASME Appendix G, K_{IR} is related to temperature, T, and to RT_{NDT} by the following equation:

$$K_{IR} = 26.77 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)] \text{ksi}\sqrt{\text{in}}.$$

This relationship is applicable only to ferritic materials that have a specified minimum yield strength of 50,000 psi or less at room temperature.

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

3.1.1. Determination of RT_{NDT}

3.1.1.1. ASME Code Method

The RT_{NDT}s of the ferritic materials, which were specified and tested in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or later Editions and Addenda, are determined as required by that Code. When sufficient material is available, the RT_{NDT}s of the beltline region materials (which were specified and tested in accordance with an Edition or Addenda of ASME Section III earlier than the Summer 1972 Addenda) 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).

3.1.1.2. Estimating Method

The RCPBs of several plants were designed and constructed in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda. Except for the beltline region materials for which sufficient test material is available, the RT_{NDT}s of the ferritic materials must be estimated. This is necessary because obtaining the test data required for the exact determination of RT_{NDT} was not required by the applicable ASME Code. Generally, drop weight tests were not performed, and the Charpy V-notch tests were limited to "fixed" energy level requirements for specimens oriented in the longitudinal (principal working) direction at a temperature of 40F or lower.

To obtain an RT_{NDT} estimate that is appropriately conservative, B&W has collected and evaluated the data from tests conducted on pressure-retaining ferritic materials to which the new fracture toughness requirements were applied.

3.1.1.3. Estimated RT_{NDT}

In the preceding section pertinent impact data for each type of ferritic material are discussed as a basis for estimating conservative RT_{NDT}s. Estimated RT_{NDT}s are needed for all materials that were specified to meet the requirements of an Edition or Addenda of ASME Section III earlier than the Summer 1972 Addenda. This section summarizes the data and the estimated RT_{NDT} of the ferritic materials used in construction of the RCPB.

The data are summarized in Table 3-1. For each type of material, the table lists the number of cases considered; the highest measured RT_{NDT}; the average of the measured RT_{NDT}s; the estimated RT_{NDT}; and the difference between the average measured and the estimated temperatures.

3.1.2. Determination of Charpy V-Notch Level

3.1.2.1. Specified Method

Appendix G to 10 CFR 50 requires complete characterization of the unirradiated impact properties of all the beltline region materials of the reactor vessel. This includes determination of RT_{NDT} and Charpy (C_v) test curves for the directions normal to and parallel to the principal working direction (other than the thickness direction). Appendix G also requires a minimum Charpy upper shelf energy (C_vUSE) of 75 ft-lb for all beltline region materials unless it is demonstrated that lower values of upper shelf fracture energy provide an adequate margin against irradiation induced degradation.

To comply with Appendix G, the beltline region materials (not including HAZ) of reactor vessels for later plants meet the following test requirements:

> In addition to the Charpy V-notch impact tests needed to determine RT_{NDT}, 15 Charpy V-notch impact tests shall be conducted in each required direction (for base metals the required directions are normal and parallel to the principal direction in which the material was worked, other than the thickness direction). The tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_v test curves (including upper shelf levels) in terms of both fracture energy and lateral expansion. Three specimens shall be tested at each test temperature for the determination of RT_{NDT}. The Charpy upper shelf energy shall be determined as follows:
>
> (1) Two sets of three Charpy specimens each shall be tested at two temperatures at which the percent of shear fracture is approximately 95%. The Charpy upper shelf energy shall be the higher average energy value of the two sets of Charpy specimens.
>
> (2) If either of the two average upper shelf energy values of step (1) is below 75 ft-lb, another set of three Charpy specimens shall be tested at a temperature at least 50F higher than the highest temperature of step (1). The Charpy upper shelf energy shall be the highest average value of the three sets of Charpy specimens.
>
> The location and orientation of the impact test specimens shall comply with the requirements of paragraph NB-2322 of Section III of the ASME Code.

The requirements for the minimum C_vUSE are described in section 3.1.3.3. The requirements above are also met for the HAZ of the beltline region base metal(s) that are selected to be monitored by the reactor vessel surveillance program. The requirements are not specified for the HAZ of the other beltline region materials because the ASME Code (Paragraph NB-4335 of the Winter 1974 Addenda) deleted the requirements for toughness testing of HAZs in the weld procedure qualification tests. B&W has elected to follow the new ASME requirements.

For the beltline region materials of reactor vessels that were specified in accordance with the requirements of an Edition or Addenda of ASME Section III issued before the Summer 1972 Addenda, the complete C_v test curves, including C_vUSE, is determined when the material is included in the reactor vessel material surveillance program. For the beltline region materials that are not included in the surveillance program, and when sufficient material is available, the C_v test curve and USE are determined only in the direction normal to the principal working direction. No minimum C_vUSE is required, other than the 50 ft-lbs/35 mils of lateral expansion for the beltline region materials of these reactor vessels, one of the conditions required to establish RT_{NDT}. When the unirradiated C_vUSE of these materials is below 75 ft-lb, the currently accepted procedure is applied to predict the end-of-service C_vUSE.

3.1.2.2. Estimating Method

The C_v USE must be estimated for reactor vessel beltline region materials that were specified in accordance with the requirements of an Edition or Addenda of ASME Section III issued before the Summer 1972 Addenda and for which insufficient material is available for testing. All available data from tests conducted on reactor vessel beltline region materials were collected and evaluated in order to obtain an appropriately conservative estimate. Not all the data were obtained in accordance with the methods specified in section 3.1.2.1 since in some cases the absorbed energy was obtained only at one temperature.

3.1.2.3. Estimated C_vUSE

The data used for estimating conservative C_vUSE is discussed in the preceding section. The estimated C_vUSE is needed for all of those beltline region materials for which test material is not available, i.e., for which the actual C_vUSE data and the estimated energy for each type of beltline region material are summarized in Table 3-2. For each type of material, the table lists the number of tested heats, the lowest measured, average measured, and estimated C_vUSEs and the average difference between the estimated and measured C_vUSE.

3.1.3. Radiation Effects

3.1.3.1. Adjustment of RT_{NDT}

Adjustment of the RT_{NDT} to accommodate the radiation-induced changes in fracture toughness of beltline region materials is an important factor in developing pressure-temperature limits. Correlations have been developed for predicting the radiation-induced RT_{NDT} to be used in adjusting the initial RT_{NDT} for pressure-temperature analyses. These correlations are not perfected and, therefore, subject to continuous updating as additional data and information is developed.

The methodology used to adjust the RT_{NDT} values as used in developing pressure-temperature limits will be in accordance with the currently accepted licensing procedures. The method used will be referenced in all pressure-temperature analyses and will be reported in the Owners licensing documents.

3.1.3.2. Decrease in C_vUSE

Neutron irradiation of the beltline region materials cause a decrease in C_vUSE. Correlations have been developed for predicting this decrease in Charpy USE. These correlations are not perfect and, therefore, are subject to updating as additional data and information is obtained.

The methodology used to predict the decrease in C_vUSE (used in the evaluation of beltline region materials) will be in accordance with the currently accepted licensing procedures. The method used will be referenced in all analyses and will be reported in the Owners licensing documents.

3.1.3.3. Acceptance Criterion for Unirradiated C_vUSE

Appendix G to 10 CFR 50 requires that the C_vUSE of the unirradiated beltline region materials be equal to or greater than 75 ft-lb except if it is demonstrated by appropriate data and analyses that lower values still provide adequate margin for degradation resulting from neutron irradiation.

This section demonstrates that for some beltline region materials, a C_VUSE lower than 75 ft-lb still provides an adequate margin for degradation from irradiation. This section also presents an acceptance criterion for C_VUSE lower than 75 ft-lb which is applied to later plants.

The beltline region of the reactor vessel includes all the ferritic material in the reactor vessel that (1) directly surrounds the effective height of the active core and (2) adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in selecting the limiting material with regard to radiation damage. The beltline region material above and below the effective height of the fuel element assemblies are irradiated to a neutron fluence received by the material directly surrounding the fuel element assemblies. Since not all beltline region material is subjected to the same neutron fluence, it is not necessary for all of this material to have a C_VUSE greater than 75 ft-lb. Also, the radiation-induced drop in C_VUSE depends not only on the neutron fluence but on the material's chemical composition. The required C_VUSE of unirradiated beltline region materials is defined in terms of the material's chemical composition and the predicted end-of-service neutron fluence to which the material will be subjected.

Complete Charpy V-notch impact curves are required for all of the unirradiated beltline region materials used in later reactor vessels. The test requirements are in accordance with Appendix G to 10 CFR 50 and are described in section 3.1.2.1. C_VUSE requirements are as follows:

1. The C_VUSE of the beltline region materials directly surrounding the effective height of the fuel assemblies shall be equal to or greater than 75 ft-lb.

2. The C_VUSE of the beltline region materials above and below the effective height of the fuel assemblies shall be equal to or greater than the sum of the following energies:

   a. The energy calculated using the material's chemical composition, end-of-service neutron fluence at the 1/4T vessel wall location, and an accepted prediction technique which will provide an end of service life C_VUSE no less than 50 ft-lbs.

   b. The energy equivalent to 5% of the energy calculated in step a.

The minimum C_vUSE above provides adequate margin for degradation from irradiation. All the beltline region materials of later reactor vessels have been specified to have a low copper content ($\leq$0.10%), and the predicted drop in C_vUSE is very small for the neutron fluence of interest.

3.2. Impact Properties of Bolting Materials

3.2.1. Code Requirements

Appendix G to 10 CFR 50 requires that materials for bolting and other fasteners meet the ASME Code. In the early editions of the ASME Code, up to and including the Winter 1971 Edition, it was required that the bolting materials exhibit a "fixed" minimum average energy at a temperature of 10F. One specimen in a set of three was allowed to be less than the fixed ft-lb value, but not less than the fixed value minus 5 ft-lb. In the Summer 1972 Addenda to the 1971 Edition, the fracture toughness requirements for bolting materials were changed to be consistent with the requirements of Appendix G except that no requirements were made in terms of absorbed energy (ft-lb). The requirements were changed again by the Summer 1973 Addenda to the 1971 Edition. In this revision and subsequent editions of ASME Section III, 45 ft-lb absorbed energy was required only for bolting materials having a nominal diameter greater than 4 inches.

All bolting materials ordered after the effective data of Appendix G to 10 CFR 50 (August 16, 1973) meet the requirements of Appendix G. Bolting materials ordered before this date must meet the requirements of the applicable ASME Code.

3.2.2. Estimating Method

To establish the minimum preload temperature and the lowest service temperature of a pressure-retaining component, it is necessary to know the lowest temperature at which the bolting materials have adequate fracture toughness. This lowest temperature is either the temperature at which the bolting materials exhibit a 25-mil lateral expansion and 45 ft-lb absorbed energy or the temperature at which the bolting materials are at the C_vUSE. For bolting materials of pressure-retaining components ordered before August 16, 1973, it is necessary to estimate the lowest temperature at which these Charpy impact properties are met. The preload temperature and the lowest service temperature are defined by the applicable equipment specification for components ordered after August 16, 1973.

Impact data from 13 heats of SA 540 Class 3 bolting were evaluated in order to estimate the lowest temperature at which bolting materials have adequate fracture toughness. The principal criteria defining the fracture toughness requirements for the bolting materials used in the reactor coolant pressure boundary are described in WRC Bulletin 175.[3] The fracture mechanics analysis performed and described in WRC Bulletin 175 shows that for the reference flaw size of 0.3 inch (nominal diameter over 3 inches), the required fracture toughness (K_{IC}) is about 125 ksi$\sqrt{in.}$ for bolting materials with a specified minimum yield strength of 130 ksi. To protect against nonductile failure, fracture toughness values exceeding 125 ksi$\sqrt{in.}$ would be needed at the lower service temperature at which maximum Code-allowed stresses occur. In WRC Bulletin 175 K_{IC} versus C_v energy correlations were used to estimate the C_v energy that would correspond to 125 ksi$\sqrt{in.}$ The K_{IC} versus C_v correlations were those of Barson and Rolfe.[4] Their empirical correlations are between slow-bend K_{IC} tests and the results of standard Charpy V-notch impact tests for the transition-temperature and upper shelf regions. The transition-temperature K_{IC}-CVN correlation is

$$\frac{(K_{IC})^2}{E} = 2(CVN)^{3/2} \qquad (1)$$

and the upper shelf K_{IC}-CVN correlation is

$$\left(\frac{K_{IC}}{\sigma_y}\right)^2 = \frac{5}{\sigma_y}\left(CVN - \frac{\sigma_y}{20}\right) \qquad (2)$$

The relationship in equation 1 suggests that at the transition-temperature region of the Charpy curve, 41 ft-lb corresponds to 125 ksi$\sqrt{in.}$ For the upper shelf region of the Charpy curve, the relationship of equation 2 relates 28 and 30 ft-lb to 125 ksi$\sqrt{in.}$ for bolting materials having yield strengths of 160 and 130 ksi, respectively.

Even though two of the bolting material heats evaluated do not meet the requirements of Appendix G, the materials have adequate fracture toughness to provide a conservative margin of safety against nonductile failure. At +40F, the bolting materials evaluated are at the upper shelf region of their C_v test curves. For the bolting materials under consideration, C_vUSEs of 28 ft-lb would have sufficient fracture energy to prevent failure because the upper shelf K_{IC}-CVN correlation shows that 28 ft-lb corresponds to 125 ksi$\sqrt{in}$. The lowest C_vUSE of the data collected, 42 ft-lb, corresponds to a fracture toughness value of 165 ksi$\sqrt{in}$. To ensure adequate margin of safety, the lowest service temperature and the minimum preload temperature are defined to be higher than 40F.

3.3. Impact Properties of Type 403 Modified Steel

3.3.1. Code Requirements

Appendix G to 10 CFR 50 requests that the adequacy of the fracture toughness properties of ferritic materials such as Type 403 modified stainless steel be demonstrated to the Commission on a case-by-case basis. The Type 403 modified steel is used as a RCPB material in the motor tube of the control rod drive mechanism. This section demonstrates that, for this application, the material has adequate fracture toughness for protection against non-ductile failure.

The nominal wall thickness of the motor tube section of interest is more than 1/2 inch and less than 5/8 inch. In the early editions of ASME Section III up to the Winter 1971 Addenda to the 1971 Edition, materials with a nominal section thickness of 1/2 inch or less did not require impact testing. Starting with the Summer 1972 Addenda, the nominal section thickness increased to 5/8 inch or less. Thus, in the early editions of ASME Section III, the Type 403 modified steel required impact testing, but in the new editions it does not. However, since this material was selected for use, B&W has ordered it to meet the impact toughness requirements of ASME Section III, as if its nominal wall thickness exceeded 5/8 inch. For materials order to ASME Section III, Summer 1972 and later Addenda, the imposed acceptance standard for nominal wall thicknesses from 5/8 to 3/4 inch, inclusive, is presented in Paragraph NB-2332. The material has also been specified to meet the requirements of SA 182 Grade F6 (forgings) or ASTM A276 (bars) as modified by ASME Code Case 1337.

When ordered according to the early revisions of Code Case 1337 (including Revision 6) and to the early editions of ASME Section III, the Type 403 modified forgings or bars were required to be impact-tested at 20F. The minimum average energy of a set of three Charpy V-notch specimens was 35 ft-lb, with one specimen allowed to be less than 35 but not less than 30 ft-lb. For both forgings and bars, the Charpy specimens were oriented in the axial (longitudinal) direction.

In the Summer 1972 Addenda to the 1971 Edition of ASME Section III, the fracture toughness requirements of all pressure boundary ferritic materials changed; however, no acceptance criterion was given for the martensitic high-alloy chromium steels, such as Type 403 modified steel. A year later, the Summer 1973 Addenda re-established the acceptance criteria for the type 4XX steels. Beginning with this addenda, the fracture toughness requirements and acceptance criteria for the type 4XX steels are described in Paragraph NB-2332 of ASME Section III. This paragraph requires that three Charpy V-notch specimens be tested at temperatures lower than or equal to the lowest service temperature. The lateral expansion of each specimen must be equal to or greater than 20 mils. The test temperature has been specified as equal to or less than 40F. The orientations of the specimens are transverse (normal to principal working direction) for the forgings and axial for the steel bars. The fracture toughness requirements of Code Case Summer 1337, starting with Revision 7, are the same as those of ASME Section III, Summer 1973 Addenda to the 1971 Edition.

3.3.2. Demonstration of Adequate Toughness

It is B&W's position that the fracture toughness requirements of the new editions of ASME Section III provide adequate protection against nonductile failure. The proof of adequate toughness is based on demonstrating that the Type 403 modified steels used in the construction of components designed to an Edition or Addenda of ASME Section III prior to the Summer 1973 Addenda meet or exceed the toughness requirements of that Addenda.

Data from 15 lots of SA 186 F6 forgings and 15 lots of ASTM A276 bars were evaluated. Based on these data, the lowest service temperature of the control rod drive mechanism can be as low at 40F; however, for additional protection against non-ductile failure, B&W has defined the component's lowest service temperature at 100F. This specified lowest service temperature is

60F above the temperature at which the fracture toughness requirements are specified and met. The additional 60F provides margins of safety beyond that required by the ASME Code and by Appendix G to 10 CFR 50.

3.4. Supplemental Fracture Toughness Properties

In the event the beltline material reaches a radiation level which causes the predicted Charpy upper shelf energy value to decrease below 50 ft-lb at 1/4T, supplemental fracture toughness data will be obtained to assess reactor pressure vessel integrity. The data are used to demonstrate equivalent margins of safety as established in Appendix G of ASME Code.

3.4.1. Terminology Related to Ductile Fracture Analysis

The terminology used in the development of material properties for analysis of the reactor vessel resistance to ductile fracture will be in accordance with the following standards.

3.4.1.1 Mechanical Properties -- ASTM Specification E6, Standard Definitions of Terms Relating to Methods of Mechanical Testing

3.4.1.2 Fracture Toughness Properties -- ASTM Specification E616, Standard Terminology Relating to Fracture Testing

3.4.2. Fracture Toughness Properties of Ductile Materials

When the beltline region materials of the reactor pressure vessel reach an irradiation level which causes the Charpy upper shelf energy value of the material to decrease to a value below 50 ft-lb, supplemental fracture toughness data to assess re r vessel integrity are required by 10 CFR 50. These data are used to provide input to the elastic-plastic fracture mechanics analysis as described in section 6.

The data base for this fracture mechanics analysis is being developed in the integrated reactor vessel material surveillance program described in BAW-1543 and the interpretation of the materials data obtained from this surveillance program will be presented in RVSP reports. The data that are most important to the analysis are those which define the initiation of ductile tearing and the resistance of the material to ductile tearing as a function of crack growth. The interpretation of the data is presented in the load- displacement curves obtained from the individual tests and the resulting J-R curves derived from the data. Supporting data is obtained from the stress-strain curves of the tension tests. These data are analyzed to obtain the true stress-true strain curves to provide the work hardening coefficients.

The actual material properties used in the establishment of the reactor pressure vessel operating limitations and supporting references will be reported in the appropriate licensing document.

3.4.3. Relationship Between Fracture Toughness Properties and the Fracture Mechanics Analysis

The technical approach used in Appendix G of 10 CFR 50 is to establish the reactor vessel operating limitations with adequate margins of safety using a fracture mechanics analysis assuming that the vessel material may behave in a non-ductile manner. In the temperature region characterized by the Charpy lower shelf and transition region a LEFM analysis is required using the procedure described in Appendix G of 10 CFR 50. In the Charpy upper shelf temperature region no additional fracture mechanics analysis is required as long as it is demonstrated that the Charpy upper shelf energy is greater than 50 ft-lbs. If the Charpy energy is predicted to drop below the 50 ft-lb level, it is required to provide supplemental fracture toughness information and an analysis to demonstrate an equivalent margin of safety as required by Appendix G of 10 CFR 50. This necessitates the use of elastic-plastic fracture mechanics analysis methods.

As part of the required supplemental analysis, a criteria must be established such that a smooth transition will occur in the vessel operating limitations between the required LEFM analysis and the supplemental EPFM analysis. A conservative approach to establishing this transition is to perform both LEFM and EPFM analyses and establish the vessel operating limits as the lower bound of the two results.

The temperature at which the transition is made from the LEFM analysis to the EPFM analysis is therefore defined as the temperature at which the allowable pressure versus temperature curves calculated by the two procedures intersect. Since the allowable pressure versus temperature curve obtained from the EPFM analysis is based on a structural instability analysis which is a function of both the structure's geometry and the material properties, the temperature at which this transition is made in general is not a function of material properties alone (see section 6).

For the specific case where the J-R curve is obtained from small RVSP fracture toughness specimens, the temperature can be determined from the material properties data alone. Because of the limited crack extension and limitations on the maximum J values allowed by ASTM, the J value at calculated instability will always be the maximum J value measured on the surveillance specimen. This limitation imposed by the specimen size provides additional conservatism in the EPFM analysis since the applied J value to cause instability of the structure will always be greater than the maximum J value obtained from the surveillance specimens.

The temperature at which the transition is made from LEFM to EPFM can be obtained by the procedure shown schematically in Figure 3-1. The K_{JR} in this figure is obtained using the procedures for converting from J to K values found in ASTM E813.

This procedure for determining the temperature for the transition from LEFM to EPFM will always be conservative because it is based on the K_{IR} curve. Since the K_{IR} curve is based on dynamic fracture tests (both dynamic loading and crack arrest), it is impossible for cleavage fracture to occur at temperatures greater than those obtained using this procedure and the K_{IR} curve.

Table 3-1. Summary of RT_{NDT} Data and Estimated Temperatures

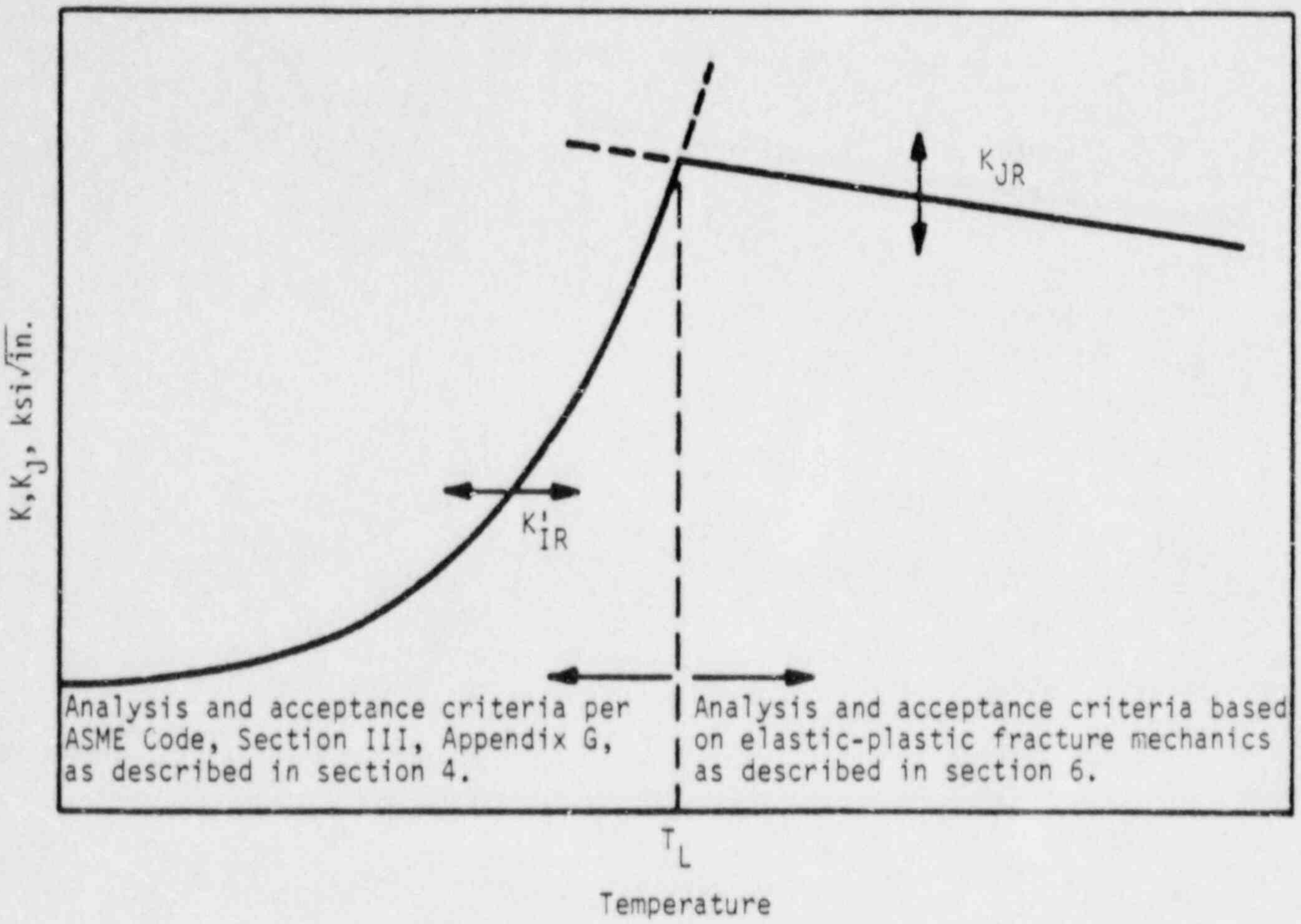
Material/type	No. of cases	RT_{NDT}, F High meas	Avg. meas	Est	Diff between ave. measured and estimated RT_{NDT},F
SA 508, Class 2 low-alloy forgings	24	60	4	60 or T_{NDT} [a]	56
SA 533 B low-alloy plates	13	40	0	40	40
SA 516 C carbon steel plates	20	10	-11	10	21
Submerged-arc Linde 80 weld	10	20	0	20	20
Submerged-arc Linde 0091 weld	10	-50	-66	20	86
Manual metal arc weld	9	-10	-67	20	87
SA 508 Class 2 HAZ	6	30	-25	30	55
SA 533 B HAZ	11	10	-23	10	33
SA 516 C HAZ	7	-20	-26	-20	6
SA 106 C piping	11	50	5	50	45

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

Table 3-2. Summary of C_vUSE Data and Estimated Upper Shelf Energies

Material/type	No. of cases	C_vUSE, ft-lb Low meas	Avg. meas	Est	Diff between ave. measured and estimated C_v USE, ft-lb
SA 508, Class 2 low-alloy forgings	5	91	124	75	49
SA 533B low-alloy plates	8	85	91	75	16
Submerged-arc weld	20	66	81	66	15

Figure 3-1. Relationship Between Fracture Toughness Properties and the Fracture Mechanics Evaluation Methods



Legend

K'_{IR} -- K_{IR} relationship as defined in ASME Code, Section III, Appendix G, for a specific material and adjusted for initial properties and effects of neutron irradiation.

K_{JR} -- Materials elastic-plastic fracture toughness relationship as developed from appropriate data base.

T_L -- Temperature at which the linear-elastic fracture mechanics and elastic-plastic fracture mechanics analytical methods interface.

4. LEFM ANALYTICAL PROCEDURES

4.1. Basis

The calculational procedures used to determine the pressure-temperature limitations on the reactor coolant (RC) system are based on ASME Appendix G, as incorporated in the Winter 1973 Addenda, and on WRC Bulletin 175.[3] To determine the minimum bolt preload temperature, the calculational procedure is partially based on Appendix A to ASME Section XI since it uses the static critical stress intensity factor K_{IC} rather than the reference critical stress intensity K_{IR} of ASME Appendix G.

Procedures for quantitatively obtaining the maximum allowable pressure at a given temperature for Class 1 ferritic pressure-retaining components are given in ASME Appendix G and are described in more detail in WRC Bulletin 175. The methods of calculating applied stress intensity factor are simplified, and the postulated flaw is defined by a reference flaw of specified size and shape. The procedures are not applicable to pressure boundary regions near geometric discontinuities, such as nozzles, and in such cases the technology of Bulletin 175 is applied directly.

The components of the RC system in a typical B&W power plant have been analyzed to determine the minimum required reactor coolant temperature for pressures of 626, 2250, and 3125 psig. The 626 psig pressure was selected because it is 1 psig above the pressure corresponding to 20% of the preoperational system hydrostatic test pressure. This is the maximum allowable pressure (625 psig) for a component when the reactor coolant temperature (or the volumetric average metal temperature) is below the lowest service temperature of the component. The components for which a lowest service temperature must be defined include the RC loop piping and the control rod drive mechanism (the CRDM is an appurtenance to the reactor vessel). The lowest service temperature of these components is 150F (based on

RT_{NDT} + 100F) for the piping and 100F (as derived in section 3.3) for the CRDM. The 2250 psig pressure was selected because it is approximately the normal operating pressure; 3125 psig was selected because it is the preservice system hydrostatic test pressure.

The reactor vessel closure head region, the reactor vessel outlet nozzles, and the beltline region are the only portions of the RC system with a relatively high minimum required temperature at 626 and 2250 psig. The reactor vessel outlet nozzle and the closure head region show the highest minimum required temperature at 3125 psig. These three regions are the only ones that, at different stages of the vessel's design life, regulate the pressure-temperature limitations of the RC system for normal operation and inservice pressure tests. The outlet nozzles and the closure head region regulate the minimum allowable preservice hydrostatic test temperature. Each region has the following characteristics:

The beltline region directly surrounds the effective height of the fuel assemblies and is exposed to continual neutron flux throughout the service life of the reactor vessel. The neutron fluence (flux x time) will change the mechanical properties of the beltline region materials. This continual change necessitates periodic adjustments to the pressure-temperature operating limitation throughout the service life of the reactor vessel. This region is remote from geometric discontinuities, and the applied stresses are proportional to the internal pressure and to the heatup or cooldown rates.

The closure head region of the reactor vessel is subject to significant stresses due to mechanical loads resulting from bolt preload. In this region, the applied stresses are not proportional to the internal pressure. This region is subjected to high stresses at relatively low temperatures. The highest stress levels occur at the head-to-head flange juncture of the closure head region.

The outlet nozzle of the reactor vessel is the largest nozzle in the RC system. The inside corner of the nozzle is subjected to high local stresses produced by pressure. The local stresses can be two to three times the membrane stress of the shell. As the radius of the nozzle increases, the magnitude of the stress intensity factor increases for a constant assumed flaw.

For loading conditions other than the preservice system hydrostatic test (PSHT), the nozzles and most other regions near geometric discontinuities are analyzed using the same safety margins as those required by ASME Appendix G for shells and heads remote from discontinuities. For the analysis of the head-to-head flange juncture of the closure head region, the safety factors are the same; however, the size of the postulated flaw is smaller than the referenced flaw. The assumed flaw on the head-to-head flange juncture is a sharp surface flaw with a depth of 1/6 t and a length of t (where t is the section thickness). The thickness of the juncture varies from 6.5 to 8 inches depending on the size of the reactor vessel. This juncture is inspected prior to service and at several intervals throughout the service life of the power plant. The inspection techniques can detect very small surface defects (defects with areas greater than 1 in.^{2} are considered detectable). For the wall thickness of 6.5 inches, the area of the postulated flaw (semi-elliptical) is 8.5 in.^{2}. The area of the postulated flaw is 8.5 times larger than the minimum detectable defect area.

For the PSHT all geometries are analyzed using a margin of safety of 1.0 on the stress intensity factor and postulated flaws that are smaller than the reference flaw of ASME Appendix G. Smaller postulated flaws are justifiable since this test is performed before initial operation. The postulated flaws employed to determine the pressure-temperature limit curve for the PSHT are described in section 4.2.2.2. Additionally a pressure exceeding 2/3 of the test pressure is not allowed until the component temperature exceeds RT_{NDT} + 60.

The reference flaw of ASME Appendix G is a sharp surface flaw perpendicular to the direction of maximum stress, having a depth of 1/4 t and length of 1-1/2 t (for section thicknesses of 4 to 12 inches). ASME Section III also requires that the test coupons be at least 1/4 t from any surface unless the material is a very thick forging and the test location is very near the surface (0.75 inch from a heat-treated surface). Since for most geometries the depth of the postulated flaw and the test location is 1/4 t (from either surface), the analytical calculations used on all geometries depend on the metal temperature and impact properties (including effects of irradiation) at 1/4 t and 3/4 t. The impact properties of thick and complex forgings at 1/4 and 3/4 t are assumed to be equal to the properties determined near the surface. The metal temperature and impact properties for the head-to-head flange juncture are taken at 5/6 t. For the analysis of the head-to-head flange juncture, the impact properties at 5/6 t are assumed to be equal to those determined by the ASME Code.

At the beginning of service life, the closure head region and the outlet nozzles control the pressure-temperature limitations of the loading conditions of interest. After several years of neutron irradition exposure, the RT_{NDT} of the beltline region materials will be high enough for the beltline region to regulate parts of the pressure-temperature limit curves. The maximum allowable pressure as a function of fluid temperature for the service period of the limit curves is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzle, and beltline region. The maximum allowable pressure is the lower of the three calculated pressures. For additional years' operation, the adjusted RT_{NDT} of the beltline region materials will continue to increase; therefore, periodic adjustments on the pressurization limit curves are required throughout the service life of the RC system. Since every surveillance capsule withdrawal will produce pertinent irradiated beltline region material impact data, adjustment of the pressurization limit curves may be required after each capsule withdrawal. The initial and subsequent adjusted pressure-temperature limits include the predicted radiation-induced RT_{NDT} (determined as described in section 3.1.3.1) for the period until the next capsule withdrawal.

After each capsule withdrawal, the RT_{NDT}s of the beltline region materials are predicted by adding the unirradiated values to the predicted radiation-induced ΔRT_{NDT}s and then confirmed by the material surveillance program test results. Both the predicted ΔRT_{NDT} and the data obtained from the surveillance program are used to define the adjusted RT_{NDT} that will be used to recalculate the pressurization limit curves.

4.2. Description

The methods used to obtain the pressure-temperature limitations for each of the loading conditions of interest are described in this section. Table 4-1 summarizes the analytical assumptions.

4.2.1. Normal Operation

4.2.1.1. Bolt Preloading

To define the minimum preload temperature, it is necessary to analyze the bolt preloading conditions. The minimum preload temperature can be the lowest temperature at which the bolting materials meet the toughness requirements of the ASME Code or the calculated minimum temperature required for protection against nonductile failure of the closure head region, whichever temperature is higher. Section 3.2 of this report shows that at 40F the bolting materials meet the requirements of ASME Code; now it is necessary to calculate the minimum allowable temperature of the closure head region to determine whether it is higher than 40F.

During bolt preloading, the maximum tensile stresses occur at the outside surface (5/6 t) of the head-to-head flange juncture of the closure head region. The stresses are primarily bolt preload bending stresses. The pressure stresses are very small since the maximum allowable pressure for this loading condition is relatively low (~450 psig). The minimum temperature required for protection against nonductile failure is first calculated at 0 psig and then at 626 psig. Both pressures are analyzed because higher temperatures may be required at 0 than at 626 psig. For both cases, the thermal stresses are nil since the coolant temperature is essentially at steady state throughout this loading condition. The method used to calculate the minimum preload temperature is as follows:

1. The membrane and bending stresses at the 5/6 t vessel wall location that result from bolt preload and internal pressure are calculated by the stress analysis of the head-to-head flange juncture at both 0 and 625 psig.

2. Using the membrane and bending stresses calculated in step 1, the stress intensity factor for both cases is calculated by the following equation:

$$K_I = 1.1\ \sigma_m M_K \frac{\sqrt{\pi a}}{\sqrt{Q}} + \sigma_b M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}$$

where the assumed flaw of a = 1/6 t;

$$K_I = 0.82\ \sigma_m \frac{\sqrt{t}}{Q} + 0.64\ \sigma_b \frac{\sqrt{t}}{Q}$$

where K_I = stress intensity factor based on reference flaw at 5/6 t vessel wall location,

M_K, M_B = correction factors for membrane and bending load conditions, respectively (values from WRC Bulletin 175, Figures A3-1 and A3-2); for a 1:6 crack depth: thickness ratio the values are 1.03 and 0.88, respectively;

Q = flaw shape factor modified for plastic zone size (reference 3 gives basic expression)

a = assumed crack depth,

t = section thickness,

σ_m = calculated membrane stress,

σ_b = calculated bending stress.

3. The relative temperature T-RT_{NDT} at which the critical static stress intensity factor K_{IC} equals the highest calculated stress intensity factor K_I (from step 2) is calculated using Figure 4-1, which is based on Figure A-4200-1 from ASME XI, Appendix A.

4. Using the relative temperature calculated in step 3 and the highest RT_{NDT} of the closure head region materials, we can calculate the minimum temperature required for protection against nonductile failure.

5. The minimum preload temperature is the one calculated in step 4 or 40F, whichever is higher.

6. Appendix G of the ASME Code Section III recommends the temperature of the closure area be RT_{NDT} at bolt-up. The forgoing procedure yields similar results with the exception of low stressed closures. This procedure is considered consistent with the philosophy of the ASME Code and will be used for establishing temperature requirements.

4.2.1.2. Heatup

The heatup transient starts at the minimum preload temperature. For temperatures above minimum preload, the heatup pressure-temperature limit curve is calculated by a point-by-point comparison of the limits imposed by the closure head region, the outlet nozzles, and the beltline region. The heatup limit curve is the composite or lower bound curve of the limits imposed by the three controlling regions.

The limits imposed by the closure head region are established by assuming a 1/6 t x t surface flaw located at the outside surface of the head-to-head flange juncture. During heatup all the stresses, including the bolt preload and the thermal stresses, are in tension at the outside surface of the closure head region. The 5/6 t location corresponds to the depth of the assumed flaw on the outside surface of the head-to-head flange juncture. The minimum required fluid temperatures are calculated at several coolant pressures above 625 psig. This is done by first calculating the fluid temperature as a function of metal temperatures for each heatup rate of interest and then calculating the minimum required metal temperature at each pressure. For fluid temperatures between the minimum preload temperature and the minimum required fluid temperature at 626 psig, the maximum allowable pressure is 625 psig.

The limits imposed by the outlet nozzles are calculated by assuming a flaw at the inside corner of the nozzle. The depth of the assumed flaw is 3 inches, which is the depth of the reference flaw of ASME Appendix G for a section thicker than 12 inches. During heatup, the inside corners of nozzles are subjected to high local stresses produced by pressure; however, the thermal stresses are in compression. The limit curve is calculated by determining the metal temperature at the inside corner 1/4 t of the outlet nozzle as a function of fluid temperature. The critical stress intensity factor is indexed to the fluid temperature using the highest RT_{NDT} of the two outlet nozzles. The maximum allowable pressure is then calculated as a function of fluid temperature. The thermal stress intensity factors for these calculations are assumed to be zero. This assumption is conservative since during heatup, the contributing thermal stress intensity factor at the inside corner of the nozzle is negative.

The pressure-temperature limits imposed by the beltline region are calculated using the postulated reference flaw of 1/4 t depth. The reference flaw is assumed to be located at both the inside and outside surfaces of the beltline region. During heatup, the thermal stresses are in compression at 1/4 t of the section thickness of the beltline region and are in tension at 3/4 t. The 1/4 t location corresponds to the depth of the reference flaw on the inside surface of the reactor vessel wall. The 3/4 t location corresponds to the depth of the reference flaw on the outside surface of the reactor vessel wall. The metal temperatures at the 1/4 and 3/4 t lag the fluid temperature during the normal heatup conditions. Since the neutron fluence attenuates through the thickness of the beltline region material, the RT_{NDT} at the 1/4 t location will be higher than that at the 3/4 t location. Because of these variables, two sets of calculations must be performed to obtain the pressure-temperature limitations imposed by the beltline region.

First, the pressurization limit for the steady-state condition is calculated as a function of fluid temperature. For this calculation the metal and fluid temperatures are the same and the impact properties used are those of the 1/4 t location. There are no thermal stresses in this case, and the only contributing stress intensity factors are those produced by pressure.

Second, the curve of pressure versus fluid temperature limit is calculated for each heatup ramp of interest assuming that the reference flaw is located at the outside surface of the beltline region wall. For this calculation, it is necessary to determine the metal temperature at 3/4 t as a function of fluid temperature and the stress intensity factor produced by the thermal stresses. The thermal stress intensity factor is added to the pressure stress intensity factor. The impact properties used in this calculation are those of the 3/4 t vessel wall location.

The methods employed to obtain the limits imposed by the closure head region, outlet nozzle, and beltline region and the pressure-temperature limit curve of the RCPB for normal heatup are described below.

Closure Head Region Heatup Limits

The heatup limits imposed by the closure head region are calculated as follows:

1. For each of the heatup ramps of interest, the metal temperature at 5/6 t of the head-to-head flange juncture is calculated as a function of fluid temperature.

2. The minimum allowable fluid temperatures of the closure head region for coolant pressures of 626, 1250, and 2250 psig are calculated as follows:

   a. The membrane and bending stresses at the 5/6 t location resulting from bolt preload, thermal gradient, and internal pressure are calculated by a detailed stress analysis of the head-to-head flange juncture.

   b. Using the membrane and bending stresses calculated in step a, the stress intensity factor is calculated by the following equation:

$$K_I = 2\left[1.1(\sigma_{mb} + \sigma_{mP})M_K \frac{\sqrt{\pi a}}{Q} + \sigma_{bb}M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}\right] + \sigma_{bT}M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}$$

   where the assumed flaw of a = 1/6 t;

$$K_I = 1.64(\sigma_{mb} + \sigma_{mP}) \frac{\sqrt{t}}{\sqrt{Q}} + 1.28\,\sigma_{bb} \frac{\sqrt{t}}{\sqrt{Q}} + 0.64\,\sigma_{bT} \frac{\sqrt{t}}{\sqrt{Q}}$$

   where σ_{mb}, σ_{mP} = calculated membrane stresses due to bolt preload and pressure,

   σ_{bb}, σ_{bT} = calculated bending stresses due to bolt preload and thermal gradient. (See section 4.2.1.1 for definition of other factors.)

   c. For each pressure, the minimum relative temperature is that at which the calculated stress intensity factor (K_I) equals the reference stress intensity factor (K_{IR}) of Figure G-2110.1 of ASME Appedix G.

   d. The minimum required metal temperatures are calculated using the minimum relative temperatures (calculated in step c) and the highest RT_{NDT} of the closure head region materials.

e. The minimum allowable fluid temperature for the three coolant pressures are calculated using the minimum required metal temperatures calculated in step d and the fluid-metal temperature relationship of step 1.

3. The pressure-temperature limits imposed by the closure head region during normal heatup are defined as follows:

   a. For fluid temperatures between the minimum preload temperature calculated in section 4.2.1.1 and the minimum allowable fluid temperature calculated in step 2 for 626 psig, the maximum allowable coolant pressure is 625 psig.

   b. For pressures of 1250 and 2250 psig, the minimum allowable fluid temperatures are those calculated in step 2. For coolant pressures between 626, 1250, and 2250 psig, the minimum fluid temperatures are defined by linear interpolation.

   c. 10CFR50 Appendix G Paragraph IV.A.2 requires the highly stressed regions of the closure region to be at a temperature of at least RT_{NDT} + 120°F for pressures above 625 psig. The forgoing procedure results in a similar temperature requirement. The required temperature is lower than 120°F if slow heat-up rates are specified and higher than 120°F for the operating pressure condition and maximum heat-up rates. The forgoing procedure is considered to be consistent with the requirements of 10CFR50 Appendix G and is used in lieu of the stated requirement.

Outlet Nozzle Heatup Limits

The heatup limits imposed by the outlet nozzles are calculated as follows:

1. For each of the heatup ramps of interest, the metal temperature at a depth of 3 inches (at the inside corner) location of the outlet nozzle is calculated as a function of fluid temperature. The thermal analysis calculations are performed using a one-dimensional transient distribution program.

2. The K_{IR} curve of ASME Appendix G is indexed to the highest RT_{NDT} of the two outlet nozzles. Using the fluid-metal temperature relationship calculated in step 1, the critical stress intensity factor is calculated as a function of fluid temperature, $K_{IR(T_f)3"}$.

3. The pressure-temperature limit curve imposed by the outlet nozzles during heatup is calculated using the following equation:

$$P(T_f) = \frac{K_{IR}(T_f)_{3"}}{2F(a/r_n)\,\dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}\,\sqrt{\pi a}}$$

where $K_{IR}(T_f)_{3"}$ = critical stress intensity factor as function of fluid temperature calculated in step 2,

$F(a/r_n)$ = obtained from WRC Bulletin 175, Figure A5-1,

r_n = apparent nozzle radius,

$P(T_f)$ = coolant pressure as function of fluid temperature,

r_o, r_i = outside and inside radius of reactor vessel nozzle belt,

a = flaw depth, assumed to be 3 inches.

Beltline Region Heatup Limits

The limits imposed by the beltline region are calculated as follows:

1. For each heatup ramp of interest, the metal temperatures at the 3/4 t vessel (beltline region) wall location are calculated as a function of fluid temperature (T_f). The thermal analysis calculations are performed by a one-dimensional transient distribution program.

2. Also, as part of the thermal analysis of step 1 in the preceding paragraph, the temperature distribution through the vessel (beltline region) wall is calculated as a function of fluid temperature for each heatup ramp.

3. The K_{IR} curve of ASME Appendix G (Figure G-2110.1) is indexed to the highest postulated RT_{NDT} of the 1/4 t wall location and to the highest postulated RT_{NDT} of 3/4 t. For each heatup ramp of interest the critical stress intensity factor for the 3/4 t vessel wall location is calculated as a function of fluid temperature using the data of step 1 and the RT_{NDT} at 3/4 t. Also, for the steady-state condition, K_{IR} is plotted as a function of fluid temperature using the RT_{NDT} at 1/4 t.

4. The K_I produced by the thermal gradient across the vessel wall is calculated as follows:

a. Utilizing the temperature distribution obtained in step 2 the equivalent linear bending stress is calculated due to the radial gradient. This is done by either integrating the thermal distribution or stress distribution across the wall.

b. The $K_{IT} = M_b \times Sth$ where M_b equals 2/3 Mm as defined in ASME Appendix G and Sth is the equivalent linear thermal bending stress.

5. The pressurization limit for a steady-state condition is calculated as a function of fluid temperature by the following equation:

$$P(T_f)_{ss} = \frac{K_{IR}(T_f)1/4\ t}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)1/4\ t$ = critical stress intensity factor for steadystate condition as a function of fluid temperature, based on RT_{NDT} at 1/4 t, calculated in step 4;

M_m = obtained from ASME Appendix G, Figure G-2214.1; a stress ratio $\geq$ actual is used (Checks are made to confirm that the proper M_m value is used.);

r_i, r_o = inside and outside radii of reactor vessel beltline region,

$P(T_f)_{ss}$ = allowed steady-state pressure as a function of fluid temperature.

6. The pressure versus fluid temperature data for each heatup ramp of interest are calculated as follows:

$$P(T_f) = \frac{K_{IR}(T_f)3/4\ t - K_{IT}(T_f)}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)3/4\ t$ = critical stress intensity factor based on 3/4 t RT_{NDT}, a function of fluid temperature calculated in step 4,

$K_{IT}(T_f)$ = K_I produced by thermal gradient across the vessel wall as a function of fluid temperature (calculated in step 5),

$P(T_f)$ = allowable pressure as a function of fluid temperature,

and the other factors are as defined above.

7. The pressure-temperature limits imposed by the beltline region during normal heatup are obtained by a point-by-point comparison of the data obtained in steps 5 and 6. The maximum allowable pressure is taken to be the lower of the two values.

Reactor Coolant Pressure Boundary Heatup Limits

The pressure-temperature limits during normal heatup of the RCPB are obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure at any given fluid temperature is taken to be the lower of the three calculated pressures.

4.2.1.3. Cooldown

The method used to obtain the cooldown pressurization limit curve for the RCPB is very similar to that used for the heatup curve. From the normal operating temperature to the minimum bolt preload temperature, the cooldown pressure-temperature limit curve is calculated through a point-by-point comparison of the limits imposed by the closure head region, the outlet nozzles, and the beltline region. The cooldown limit curve is the lower bound curve of the limits imposed by the three controlling regions.

The cooldown limits of the closure head region are established, as for heatup, by assuming a 1/6 t x t surface flaw located at the outside surface of the head-to-head flange juncture. Although the inside surface is subjected to positive thermal stresses during cooldown, the total stress is higher at the outside than at the inside surface. This is due to the high bolt preload bending stresses on the outside surface. The cooldown and heatup limits of the closure head region are calculated very similarly. The only differences are that (1) the fluid and metal temperatures are assumed to be equal (steady-state), and (2) the thermal stresses at the outside surfaces are assumed to be zero. The steady-state assumption is conservative since the metal temperature, especially at 5/6 t, is higher than the fluid temperature during cooldown. The assumption that the thermal stresses are zero is also conservative since the thermal stresses at the outside wall of the closure head region are negative during cooldown.

The cooldown limits imposed by the outlet nozzles are calculated, as for heatup, assuming a 3-inch-deep flaw at the inside corner of the nozzle. During cooldown, the inside corners of the nozzles are subjected to high local stresses produced by the pressure and temperature gradient. To calculate the limit curve, the metal temperature 3 inches from the inside corner locations is calculated as a function of fluid temperature. When calculating the maximum allowable pressure, the contributing thermal stress intensity factor is assumed to be equal to that calculated for the nozzle belt vessel wall. This assumption is conservative because the thermal stress intensity factor for a nozzle corner flaw is also lower than that for a surface flaw on the nozzle vessel wall owing to the lower postulated crack penetration (crack depth over section thickness) on the nozzle corner.

The method used to calculate the cooldown pressure limit curve imposed by the beltline region is also similar to that used for the heatup limit curve; however some differences exist. During cooldown, the thermal stresses are in tension at 1/4 t and in compression at 3/4 t. Because the thermal stresses are in tension at 1/4 t, and the RT_{NDT} at 1/4 t will be higher than that at 3/4 t after exposure to neutron irradiation, only the metal temperature and the impact properties of the 1/4 t location are used to obtain the cooldown limit curve. However, three calculational steps are required to obtain the cooldown limit curve of the beltline region:

1. The pressure limit curve for a steady-state condition is calculated as a function of fluid temperature. The assumed steady-state condition makes the fluid and metal temperatures equal. The impact properties are those of the 1/4 t location. The contributing thermal stress intensity factor is zero. This step is required because the metal temperature may not be higher than that of the fluid during an upset cooldown condition as it is during normal cooldown.

2. The pressure limit curve is calculated for each cooldown ramp of interest assuming that the reference flaw is located at the inside surface of the beltline region wall. For this calculation, the metal temperature at 1/4 t is determined as a function of fluid temperature, and the thermal stress intensity factor is added to the stress intensity factor produced by pressure. The impact properties at 1/4 t are used in this calculation.

3. A point-by-point comparison of the data obtained in the first two steps will obtain the lowest pressure at any temperature of the two data sets. The calculated lowest pressure becomes the maximum pressure at any temperature for the reactor vessel beltline region.

The methods used to obtain the limits imposed by the closure head region, outlet nozzle, and beltline region and the pressure-temperature limit curve for the RCPB for normal cooldown are described below.

Closure Head Region Cooldown Limits

The cooldown limits imposed by the closure head region are calculated as follows:

1. For each cooldown ramp of interest, the metal temperature at 5/6 t of the head-to-head flange juncture is assumed to be equal to the fluid temperature.

2. The minimum allowable fluid temperatures of the closure head region for pressures of 626, 1250, and 2250 psig are calculated as follows:

   a. The membrane and bending stresses at 5/6 t resulting from bolt preload and internal pressure are calculated by a detailed stress analysis of the head-to-head flange juncture.

   b. Using the membrane and bending stresses calculated in step a, the stress intensity factor is calculated using the following equation:

$$K_I = 2\left[1.1(\sigma_{mb} + \sigma_{mp})M_K\frac{\sqrt{\pi a}}{\sqrt{Q}} + \left(\sigma_{bb}M_B\frac{\sqrt{\pi a}}{\sqrt{Q}}\right)\right]$$

   where the assumed flaw of a = 1/6 t;

$$K_I = 1.64(\sigma_{mb} + \sigma_{mp})\frac{\sqrt{t}}{\sqrt{Q}} + 1.28\ \sigma_{bb}\frac{\sqrt{t}}{\sqrt{Q}}$$

   where K_I, M_K, M_B, Q, a, and t are defined in section 4.2.1.1 and other factors in "Closure Head Region Heatup Limits," step 2b.

c. For each pressure the minimum relative temperature is that at which the calculated stress intensity factor K_I equals the reference stress intensity factor (K_{IR} of ASME Appendix G, Figure G-2110.1).

d. The minimum required fluid temperatures are calculated using the minimum relative temperatures (calculated in step c) and the highest RT_{NDT} of the closure head region materials.

3. The pressure-temperature limits imposed by the closure head region during normal cooldown are defined as follows:

   a. For fluid temperatures between the minimum preload temperature calculated in section 4.2.1.1 and the minimum allowable fluid temperature calculated in step 2 for 626 psig, the maximum allowable pressure is 625 psig.

   b. The minimum allowable fluid temperatures for pressures of 1250 and 2250 are those calculated in step 2. For coolant pressures between 625, 1250, and 2250 psig, the minimum fluid temperatures are defined by linear interpolation.

Outlet Nozzle Cooldown Limits

The cooldown limits imposed by the outlet nozzles are calculated as follows:

1. For each cooldown ramp of interest, the metal temperature at the 3-inch depth (from the inside corner) location of the outlet nozzle is calculated as a function of fluid temperature. The thermal analysis is performed using a one-dimensional transient distribution program.
2. As part of the thermal analysis in step 1, the temperature difference through the nozzle belt vessel wall is calculated as a function of fluid temperature.
3. The K_{IR} curve of ASME Appendix G is indexed to the highest RT_{NDT} of the two outlet nozzles. Using the data calculated in step 1, the critical stress intensity factor at the inside corner of the nozzle is calculated as a function of fluid temperature, $K_{IR}(T_f)_{3"}$.

4. The K_I produced by the thermal gradient across the outlet nozzle corner is calculated by same method as step 4 in heatup procedure.

5. The pressure-temperature limit curve imposed by the outlet nozzles during cooldown is calculated using the following equation:

$$P(T_f) = \frac{K_{IR}(T_f)_{3"} - K_{IT}(T_f)}{2F(a/r_n)\,\dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}\,\sqrt{\pi a}}$$

where $K_{IR}(T_f)_{3"}$ = critical stress intensity factor as a function of fluid temperature calculated in step 3,

$K_{IT}(T_f)$ = thermal stress intensity factor as a function of fluid temperature calculated in step 4,

and all other factors are defined in "Outlet Nozzle Heatup Limits," step 3.

Beltline Region Cooldown Limits

The limits imposed by the beltline region during cooldown are calculated as follows:

1. For each cooldown ramp of interest, the temperature at 1/4 t (beltline region) is calculated as a function of fluid temperature (T_f).

2. As part of the thermal analysis of step 1, the temperature difference through the vessel (beltline region) wall is also calculated as a function of fluid temperature for each cooldown ramp.

3. The most limiting adjusted RT_{NDT} at 1/4 t is also used in the cooldown analysis.

4. The K_{IR} curve of ASME Appendix G is indexed to the adjusted RT_{NDT} of step 3. For each cooldown ramp of interest, K_{IR} is plotted as a function of fluid temperature using the data from step 1. For the steady-state condition, K_{IR} is also plotted as a function of fluid temperature using the same adjusted RT_{NDT}.

5. The K_I produced by the thermal gradient across the vessel wall during cooldown is calculated as describ... in step 4 of the heatup procedure. However, the ΔT values are those calculated in step 2 for each cooldown ramp.

6. The pressurization limit for steady-state condition is calculated as a function of fluid temperature, as described in "Beltline Region Heatup Limits," step 5.

7. The pressure-versus-fluid temperature data for each cooldown ramp are calculated as follows:

$$P(T_f) = \frac{K_{IR}(T_f)1/4\ t - K_{IT}(T_f)}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)1/4\ t$ = K_I based on RT_{NDT}, also a function of fluid temperature (see step 4),

$K_{IT}(T_f)$ = thermal stress intensity factor as a function of fluid temperature (see step 5).

and the other factors are as defined in "Beltline Region Heatup Limits," steps 6 and 7.

8. The pressure-temperature limits imposed by the beltline region during normal cooldown are obtained through a point-by-point comparison of the data obtained in steps 6 and 7; the maximum allowable pressure is taken to be the lower of the two values.

Reactor Coolant Pressure Boundary Cooldown Limits

The pressure-temperature limits during normal cooldown of the RCPB are obtained through a point-by-point comparison of the limits imposed by the closure head region ("Closure Head Region Cooldown Limits," step 3), the outlet nozzles ("Outlet Nozzle Cooldown Limits," step 5), and the beltline region ("Beltline Region Cooldown Limits," step 8). The maximum allowable presure at any given fluid temperature is taken to be the lowest of the three calculated pressures.

4.2.2. Preservice System Hydrostatic Test (PSHT)

4.2.2.1. Bolt Preloading

The minimum preload temperature for the PSHT is calculated by following the basic methods employed for normal operation (section 4.2.1.1). For the PSHT the minimum preload temperature is calculated using a postulated surface flaw 1/8 t deep and 3/4 t long (1/8 t x 3/4 t) located in the outside surface of the head-to-head flange juncture. This assumed flaw is smaller than that assumed during normal operation (1/6 t x t). The smaller flaw is conservative since the PSHT is performed after the nondestructive testing required by ASME Section III, and the system has not been subjected to cyclic loading.

For the smaller postulated flaw, the equation used to calculate the stress intensity factor (step 2) takes the following form:

$$K_I = 0.70\,\sigma_m \frac{\sqrt{t}}{\sqrt{Q}} + 0.57\,\sigma_b \frac{\sqrt{t}}{\sqrt{Q}}$$

where K_I is the stress intensity factor based on a 1/8 t x 3/4 t flaw, and all other factors are as defined in section 4.2.1.1.

The values of σ_m and σ_b are calculated as described for normal operation (step 1) for the higher specified preload. All other steps of the procedure for calculating minimum preload temperature for normal operation are followed when calculating the minimum preload temperature for PSHT.

4.2.2.2. Heatup and Cooldown

As described in section 2.4, the PSHT pressure is normally reached when the metal temperature of the controlling pressure boundary is at steady state, and it is higher than the calculated minimum test temperature. At temperatures lower than this minimum, the maximum allowable pressure is only 625 psig. However, for some plants, it may be necessary to gradually increase the maximum allowable pressure as the metal temperature increases, just as for normal heatup and cooldown. For these plants the thermally induced stresses are considered when calculating the pressure-temperature limit curve. The methods for calculating the PSHT limit curve are similar to those for normal operation except for the following deviations:

1. The analysis is only performed for the two regions of the RCS that potentially control the PSHT pressure-temperature limits: the closure head region and the outlet nozzle. The beltline region does not control these limits since the materials have not been affected by irradiation.

2. When calculating the limits imposed by the closure head region, the postulated flaw is a 1/8 t x 3/4 t semi-elliptical surface flaw in the outside surface. The applied factor of safety in the stress intensity factor is 1.0, and the minimum allowable temperature is also calculated for 3125 psig. The postulated flaw is the same as that assumed when calculating the minimum preload temperature for the PSHT (section 4.2.2.1).

3. When calculating the limits imposed by the outlet nozzles, the postulated flaw is a surface flaw 1.0 inch deep located at the inside corner, and the factor of safety applied on the stress intensity factor due to pressure is 1.0. The justification for the smaller postulated flaw (1.0 rather than 3.0 inches deep) is again the nondestructive examination prior to PSHT and the impossibility of fatigue crack growth.

4. The pressure-temperature limits are calculated for both heatup and cooldown; however, for simplicity, the most limiting curve is used to define these limits from initiation to completion of the PSHT. The PSHT limit curve for the RCPB is the composite or lower bound of the limits imposed by the two controlling regions during both heatup and cooldown.

4.2.3. Inservice System Leak and Hydrostatic Tests (ISLHT)

4.2.3.1. Bolt Preloading

The minimum preload temperature for the ISLHT is the same as that for normal operation since the same load is specified.

4.2.3.2. Heatup and Cooldown

Since the ISLHT can be performed throughout the service life of the power plant, the effects of irradiation are considered when establishing the pressure-temperature limit curve for each test. As for normal heatup and cooldown, the closure head region, the outlet nozzles, and the beltline region are the only regions of the reactor vessel that control the pressurization limits of the RC system during ISLHT. The normal means of heating or cooling the system, before or after reaching the desired pressure for each test, are those used during normal heatup and cooldown. Consequently, the methods used to obtain the pressure limit curves of these loading conditions are similar to those used for normal heatup and cooldown. As for the PSHT, the ISLHT pressure-temperature limits are calculated for both heatup and cooldown; however, for simplicity, the most limiting curve is used to define the pressure-temperature limits from initiation to completion of the ISLHT. Another deviation from the methods employed for normal heatup and cooldown is the magnitude of the applied factor of safety. The factor of safety applied to calculate the stress intensity factor and the allowable pressure in the preceding procedures is 1.5 rather than 2.0. The ISLHT pressure-temperature limit curve is the composite or lower bound curve of the limits calculated for heatup and cooldown. The requirement of 10CFR50 Appendix G specifying a temperature of RT_{NDT} + 90°F for highly stressed regions of the closure for pressures above 625 psig is essentially met by this procedure. As for the normal heat-up case higher or lower temperatures may be requied depending on heat up rate.

4.2.4. Reactor Core Operation

Except for low-power physics tests, the pressure-temperature limits for reactor core operation are as follows:

1. The fluid temperature must be equal to or higher than the minimum required for the ISLHT as calculated by the method described in section 4.2.3.

2. In addition, the fluid temperature must be at least 40F higher than the minimum pressure-temperature limit curve for both normal heatup and cooldown as calculated by the methods described in section 4.2.1.

3. The fluid temperature must be at least 525F.

These pressure-temperature limits for reactor core operation are in accordance with Appendix G to 10 CFR 50.

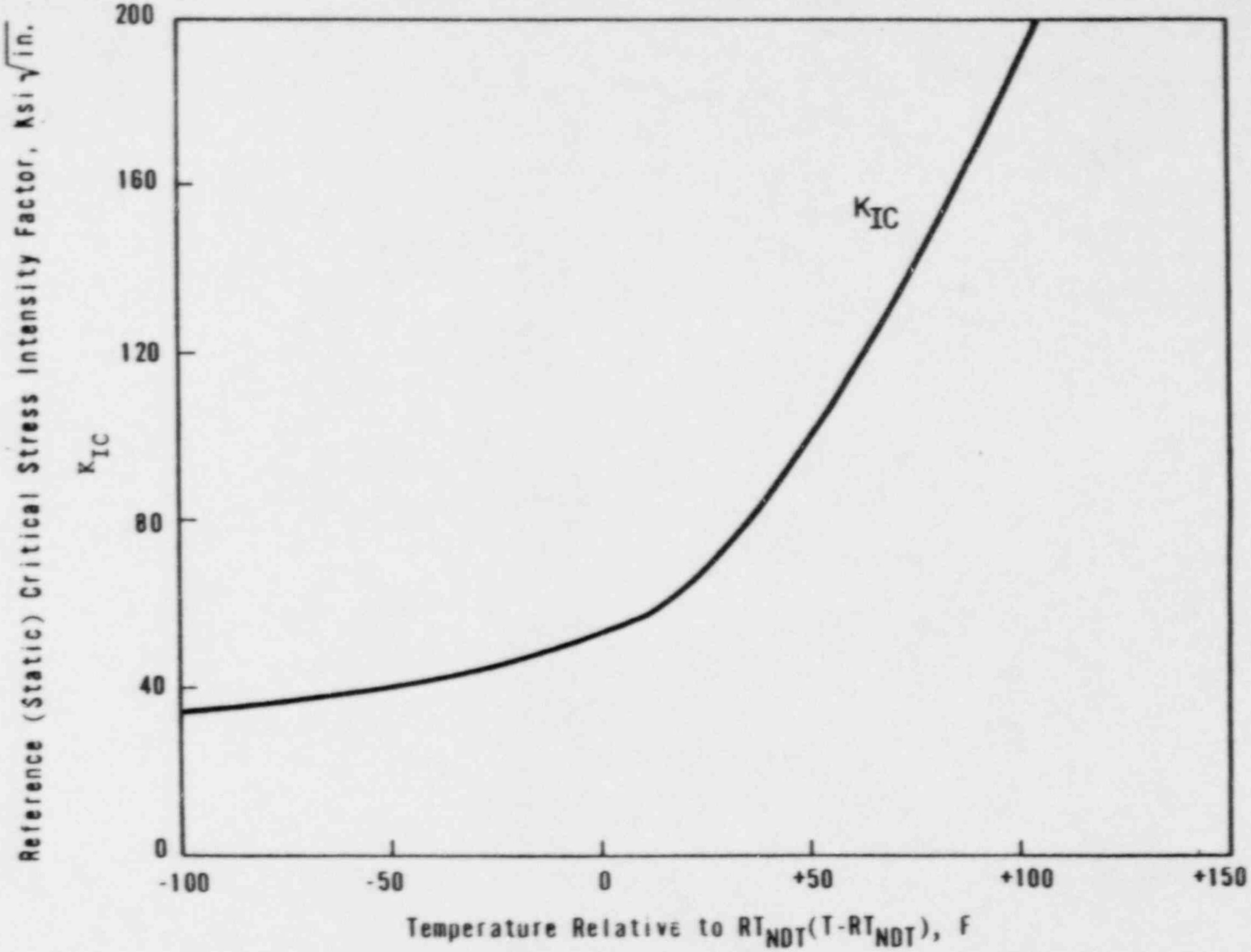
Table 4-1. Outline of Methods

Loading condition	Region analyzed	Flaw: Loc'n	Flaw: Depth	Appl safety factor[(a)]: K'_I	Appl safety factor[(a)]: K''_I	Appl safety factor[(a)]: K_{IT}	Temperature relationship	Mat'l property: RT_{NDT}[(b)]	Mat'l property: K_I
Normal bolt preload	Closure head	OD	1/6 t	1	1	--	Steady-state	1/4 t	K_{IC}
Normal heatup	Closure head	OD	1/6 t	2	1	1	$T_f(T_m)$	1/4 t	K_{IR}
	Outlet nozzle	ID	3 in.	2	--	--	$T_f(T_m)$	3/4 in.	K_{IR}
	Beltline	ID	1/4 t	2	--	--	Steady-state	1/4 t	K_{IR}
		OD	1/4 t	2	--	1	$T_f(T_m)$	3/4 t	K_{IR}
Normal cooldown	Closure head	OD	1/6 t	2	1	--	Steady-state	1/4 t	K_{IR}
	Outlet nozzle	ID	3 in.	2	--	1	$T_f(T_m)$	3/4 in.	K_{IR}
	Beltline	ID	1/4 t	2	--	--	Steady-state	1/4 t	K_{IR}
		ID	1/4 t	2	--	1	$T_f(T_m)$	1/4 t	K_{IR}
Preservice SH test bolt preload, heatup and cooldown	Same as normal bolt preload, heatup and cooldown; however the depths of the postulated flaws are 1/8 t and 1.0 inch for the closure head region and outlet nozzle, respectively, the applied safety factor is always 1.0, and the beltline region is not considered. The limit curve is the composite of the limits imposed by the two controlling regions during both heatup and cooldown.								
Inservice SLH test bolt preload, heatup and cooldown	Same as normal bolt preload, heatup and cooldown, however, the applied safety factor is 1.5 rather than 2.0. The limit curve is the composite of the limits imposed by the three controlling regions during both heatup and cooldown.								

(a)K_I = stress intensity factor resulting from primary stresses, K" = stress intensity factor resulting from secondary stresses.

(b)Location of the RT_{NDT} used in the calculation.

Figure 4-1. Reference (Static) Critical Stress Intensity Factor Vs Temperature Relative to RT_{NDT} ($T-RT_{NDT}$)



5. TYPICAL PRESSURE-TEMPERATURE LIMITS

5.1. Composite Limit Curves

The methods described in sections 3 and 4 have been applied to a typical 177 FA type plant to illustrate the development of the composite limit curves. The methods were applied for each of the loading conditions of interest. The analysis for normal heatup and cooldown was performed for the service periods ending at 5 and 32 effective full-power years (EFPY). The analysis for the inservice leak and hydrostatic tests (ISLHT) was performed for the service period ending at 5 EFPY. For consistency, the analysis was performed using a 100F/hour temperature ramp. For some transients the assumed 100F/hour ramp is not practical. The actual pressure-temperature limit curves for B&W plants may be different from those presented in this section because of the different maximum allowable temperature ramp rates and variations in ΔRT_{NDT}. The figures included here are for illustration only.

The analysis used the unirradiated impact properties, residual elements, predicted neutron fluence, and predicted radiation-induced ΔRT_{NDT} for the beltline region materials typical of an 177 FA-type plant. The unirradiated RT_{NDT}s of the closure head region materials and outlet nozzles are also those of a typical plant. The unirradiated impact properties and residual elements of the beltline region materials are listed in Table 5-1. The predicted neutron fluence values at the 1/4 t and 3/4 t beltline region locations for 5 and 32 EFPY and the corresponding ΔRT_{NDT}s and adjusted RT_{NDT} are listed in Table 5-2 for each of the beltline region materials.

Figures 5-1 and 5-2 illustrate the development of the composite pressure-temperature limit curves for a 100F/h normal heatup. The figures are applicable for the service periods ending at 5 and 32 EFPY, respectively. In addition to the composite limit curve, both figures show the limit curves imposed by the outlet nozzles, closure head region, and beltline region based on steady state and by the beltline region based on a finite heatup rate. As shown in Figure 5-1, the composite limit curve for 5 EFPY is the lower bound curve of the limits imposed by the outlet nozzles, closure head region, and beltline region based on a finite heatup rate. The composite limit curve for 32 EFPY is controlled by the limits set by the beltline region based on steady-state and finite heatup and the closure head region. At 5 EFPY, the limits set by the closure head region largely control the composite limit curve, and at 32 EFPY the same region only controls a small portion. This is because the limit curves set by the closure head region and outlet nozzles do not change throughout the service life of the power plant. Also, note that the limits set by the beltline region based on steady state do not control the composite limit curve for 5 EFPY, but they largely control the composite limit curve for 32 EFPY. This is due to the large difference in RT_{NDT} between 1/4 t and 3/4 t. Both figures illustrate the crossover of the limit curves imposed by the several regions and the need for composite limit curves.

Figures 5-3 and 5-4 are very similar to 5-1 and 5-2; however, they are for normal cooldown. Note that the beltline region steady-state limit curves for normal cooldown control the composite limit curves for 5 and 32 EFPY. At the high fluid temperatures ($T_f > 205F$) the normal cooldown composite limit curve for 5 EFPY (Figure 5-3) is less restrictive than the curve for normal heatup (Figure 5-1). This is primarily due to the large difference between the fluid temperature and the closure head region wall metal temperature at 5/6 t that occurs during heatup. However, at the lower fluid temperature ($T_f < 124F$), Figure 5-3 is more restrictive than Figure 5-1 because of the contributing thermal stresses at the inside corner of the outlet nozzles. The presence of the thermal stresses reduces the maximum allowable pressure. Again, Figures 5-3 and 5-4 illustrate the need for the composite limit curves.

Figures 5-5 and 5-6 present the limits imposed by the several regions of the reactor vessel and the composite limit curves for the PSHT and ISLHT. The allowable pressure-temperature combinations of these figures differ because of the different sizes of the postulated flaws, applied margins of safety, and assumed RT_{NDT}s. For both tests the limit imposed by the closure head region during heatup control the composite limit curves. For the

ISLHT the limits imposed by the outlet nozzles during cooldown also control the composite limit curve at the low temperatures. However, for the ISLHT, the beltline region would eventually control at higher EFPY. At 5 EFPY the temperature difference between the RT_{NDT}s of the beltline and the closure head region materials is not large enough to compensate for the higher stress intensities that the closure head region is subjected to (at the same internal pressure).

Figure 5-7 shows the development of the minimum pressure-temperature limit curve for reactor core operation up to 5 EFPY based on Appendix G to 10 CFR 50. The references used here are the limit curve for normal heatup and the minimum permissible temperature for the ISLHT pressure. The data used for Figure 5-7 are the composite limit curve of Figure 5-1 and the minimum permissible temperature for 2500 psi obtained from Figure 5-6. The criticality limits imposed by the Technical Specifications are based on other considerations since these limits are not controlling.

5.2. Technical Specification Limit Curves

The Technical Specificiations for each plant give allowable pressure and temperature combinations and require that the RC system be maintained within these limits during normal heatup and cooldown, criticality, and inservice leak and hydrostatic tests. The objective of these pressure-temperature limits is to prevent stresses from exceeding the ASME Code maximum allowable design stresses and the stresses allowed by ASME Appendix G for protection against nonductile failure. Since the stresses allowed by ASME Appendix G are generally more restrictive than the Code maximum allowable design stresses, the Technical Specifications pressure-temperature limits are the nonductile fracture prevention limits presented in section 5.1. However, there is one exception:

During cooldown, the stresses in the steam generator tubing may exceed the ASME Code maximum allowable stresses if cooldown rates are high, and the allowable pressure-temperature combination during cooldown is calculated according to ASME Appendix G. When high cooldown rates are desired, the pressure-temperature limit curve is modified by reducing the allowable pressure, which reduces stresses in the steam generator tubing.

Figures 5-8 through 5-10 are pressure-temperature limit that illustrate the limit curves that appear in the Technical Specifications of a typical 177 FA type plant. Figure 5-8 represents the normal heatup limits applicable for the first 5 EFPY. Figure 5-9 represents the normal cooldown limitations, and Figure 5-10 represents the ISLHT limits. These figures were obtained from Figures 5-1, 5-3, and 5-6, respectively. Figure 5-7 was also used to develop Figure 5-8. Figures 5-1, 5-3, and 5-8 were adjusted as follows:

1. The maximum allowable pressure had been reduced by the pressure differential between the point of system pressure measurement and the limiting region of the reactor vessel for all operating pump combinations. The applied pressure differential is 100 psig when either the beltline region or the outlet nozzles control the pressure-temperature curves and 75 psig when the closure head region controls them. These pressure differentials have been conservatively calculated.

2. Figures 5-3 and 5-6 were adjusted to include the pressure-temperature limits imposed by the steam generator tubing.

For some 177 FA plants and other B&W plants, the Technical Specification limit curves will be different from those presented in Figures 5-8 through 5-10. The differences are caused by the lower maximum allowable ramp rates and the material's RT_{NDT}, wall thickness, neutron fluence, etc.

5.3. Preservice System Hydrostatic Test Limit Curve

The Technical Specifications do not include the RC system pressure-temperature limits for the PSHT since this test is conducted before the plant operating license is issued. The limits for the PSHT are imposed by the test procedure.

Figure 5-11 is the PSHT limit curves as developed by adjusting the composite limit curve of Figure 5-5. The adjustments are the same as those used to develop Figures 5-8 through 5-10. Figure 5-11 is labeled as the PSHT limit curve for the typical 177-FA type plant. The actual curves may differ since this curve was calculated using 100°/hour heatup and cooldown ramp rates and during the PSHT, the ramp rates are much lower than 100F/hour.

Table 5-1. Unirradiated Impact Properties and Residual Element Content of Beltline Region Materials in a Typical 177 FA Plant

Material identification type, location	Core MP to weld CL, in.	Transverse C_v USE, ft-lb	RT_{NDT}, F	Chemistry, % Cu	P	S
A. SA 508 Class 2, nozzle belt	--	183	+10	0.054	0.008	0.006
B. SA 533 B upper shell	--	88	+30	0.20	0.008	0.016
C. SA 533 B upper shell	--	90	+20	0.20	0.008	0.016
D. SA 533 B lower shell	--	119	-20	0.12	0.013	0.015
E. SA 533 B lower shell	--	99	+40	0.12	0.013	0.015
F. Weld, upper long.	--	(66)[a]	(+20)	0.20	0.009	0.009
G. Weld, upper circ	123	(66)	(+20)	0.19	0.021	0.016
H. Weld, mid circ (100%)	-63	(66)	(+20)	0.27	0.014	0.011
I. Weld, lower long. (100%)	--	(66)	(+20)	0.22	0.015	0.013
J. Weld, lower circ (100%)	-249	(66)	(+20)	0.20	0.015	0.021
K. Weld, outlet nozzle	244.8	(66)	(+20)	0.19	0.021	0.016

[a] Numbers in parentheses indicate predicted values.

Table 5-2. Typical Material Data for Preparing Beltline Region Pressure-Temperature Limit Curves

Mat ID	Initial RT_{NDT}, F	End of 5th EFPY: Fluence, $E \geq 1$ MeV n/cm^2, 1/4 t	End of 5th EFPY: Fluence, $E \geq 1$ MeV n/cm^2, 3/4 t	End of 5th EFPY: ΔRT_{NDT}, F, 1/4 t	End of 5th EFPY: ΔRT_{NDT}, F, 3/4 t	End of 5th EFPY: RT_{NDT}, F, 1/4 t	End of 5th EFPY: RT_{NDT}, F, 3/4 t	End of 32nd EFPY: Fluence, $E \geq 1$ MeV n/cm^2, 1/4 t	End of 32nd EFPY: Fluence, $E \geq 1$ MeV n/cm^2, 3/4 t	End of 32nd EFPY: ΔRT_{NDT}, F, 1/4 t	End of 32nd EFPY: ΔRT_{NDT}, F, 3/4 t	End of 32nd EFPY: RT_{NDT}, F, 1/4 t	End of 32nd EFPY: RT_{NDT}, F, 3/4 t
A	+10	2.63E18	5.9E17	30	16	40	26	1.68E19	3.8E18	105	35	115	45
B	+30	2.63E18	5.9E17	75	38	105	68	1.68E19	3.8E18	220	90	250	120
C	+20	2.63E18	5.9E17	76	38	95	58	1.68E19	3.8E18	220	90	240	110
D	-20	2.63E18	5.9E17	45	21	25	1	1.58E19	3.8E18	140	59	130	39
E	+40	2.63E18	5.9E17	45	21	85	61	1.68E19	3.8E18	150	69	180	99
F	+20	2.25E18	5.1E17	70	34	90	54	1.44E19	3.25E18	212	85	232	110
G	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	110
H	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	110
I	+20	2.0E18	4.53E17	64	32	84	52	1.28E19	2.9E18	200	80	220	100
J	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--
K	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--

Figure 5-1. Normal Heatup Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

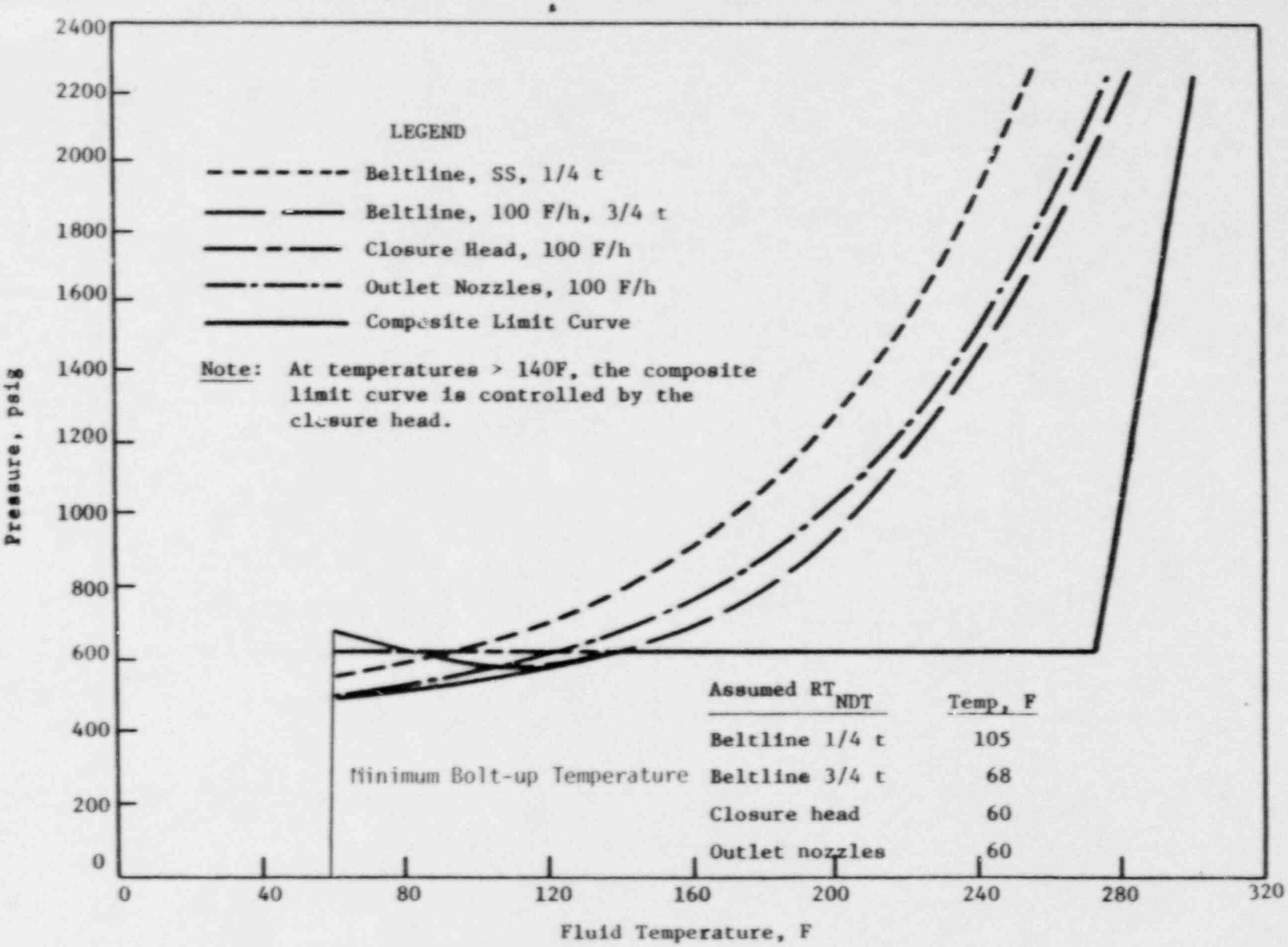


Figure 5-2. Normal Heatup Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 32 EFPY

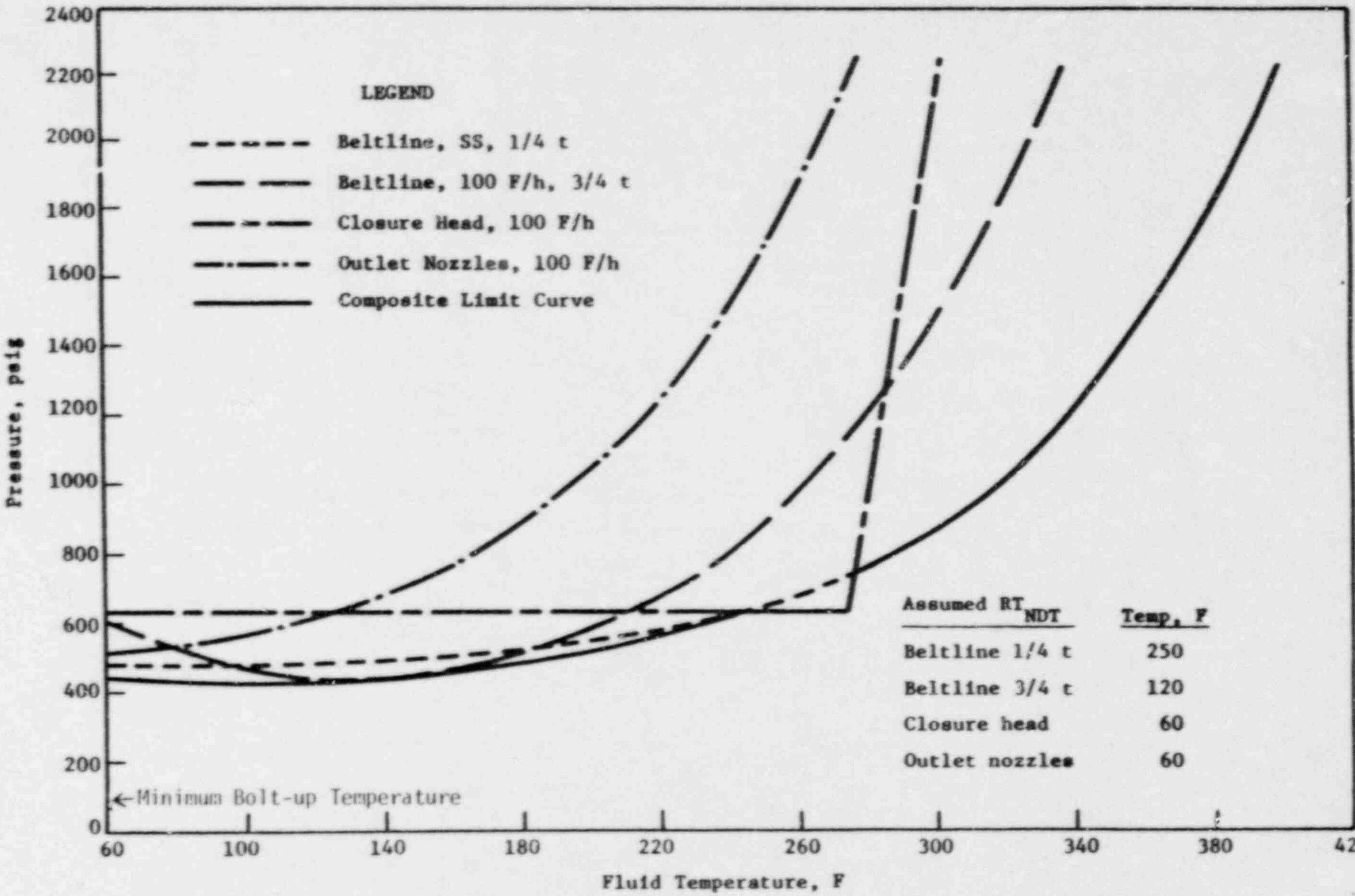


Figure 5-3. Normal Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

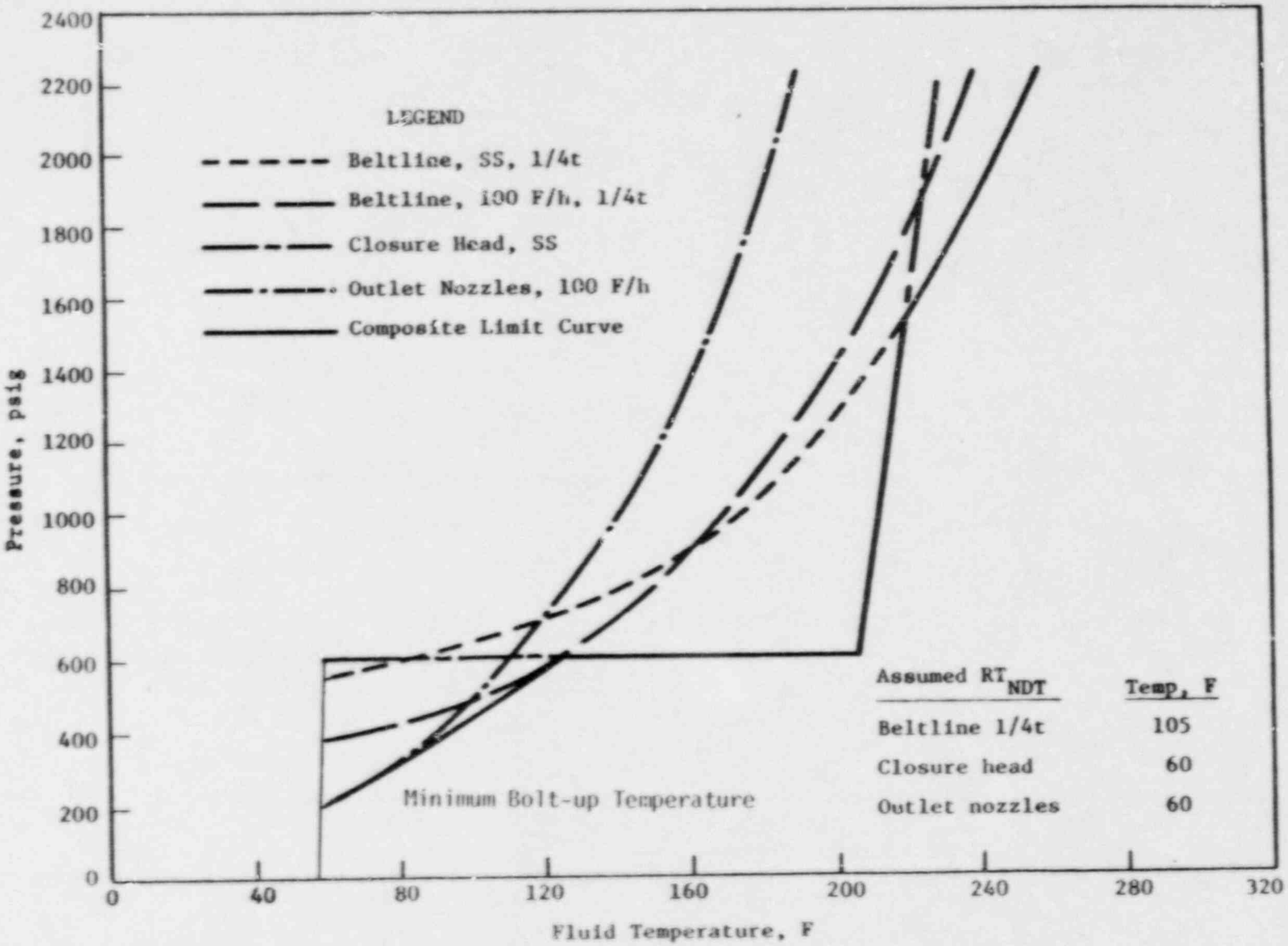


Figure 5-4. Normal Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 32 EFPY

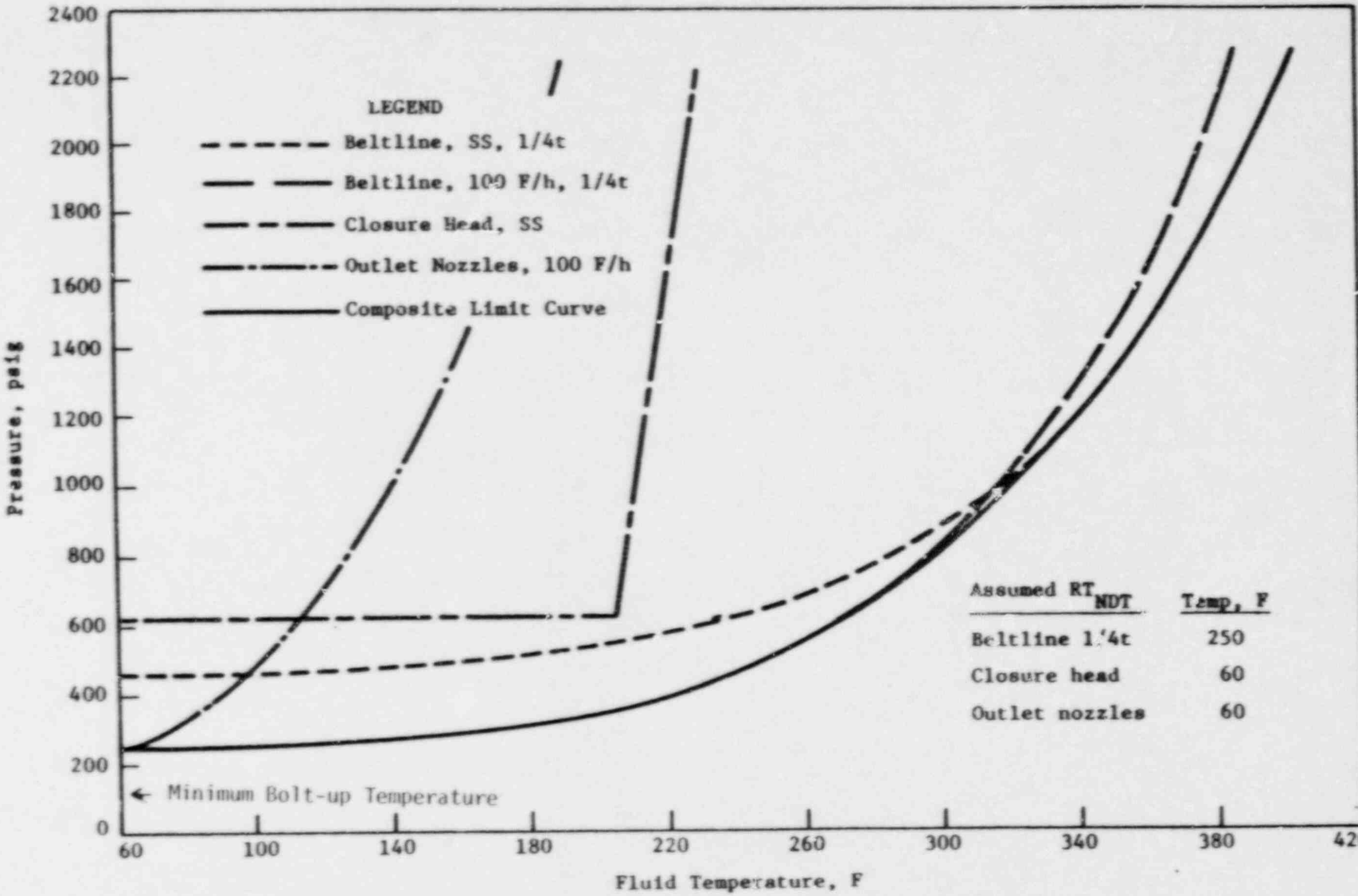


Figure 5-5. PSHT Heatup and Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve

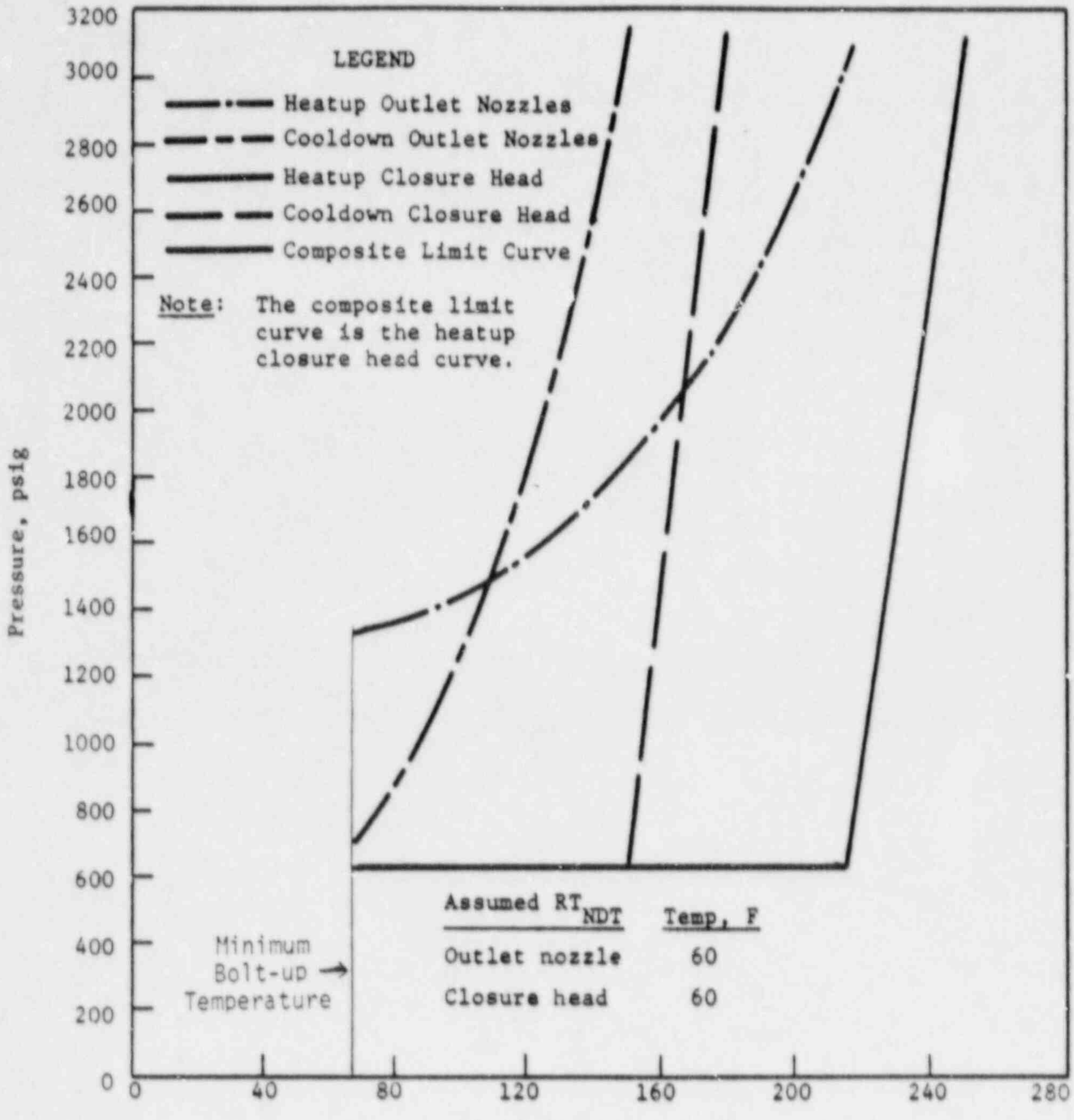


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Figure 5-6. ISLT Heatup and Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

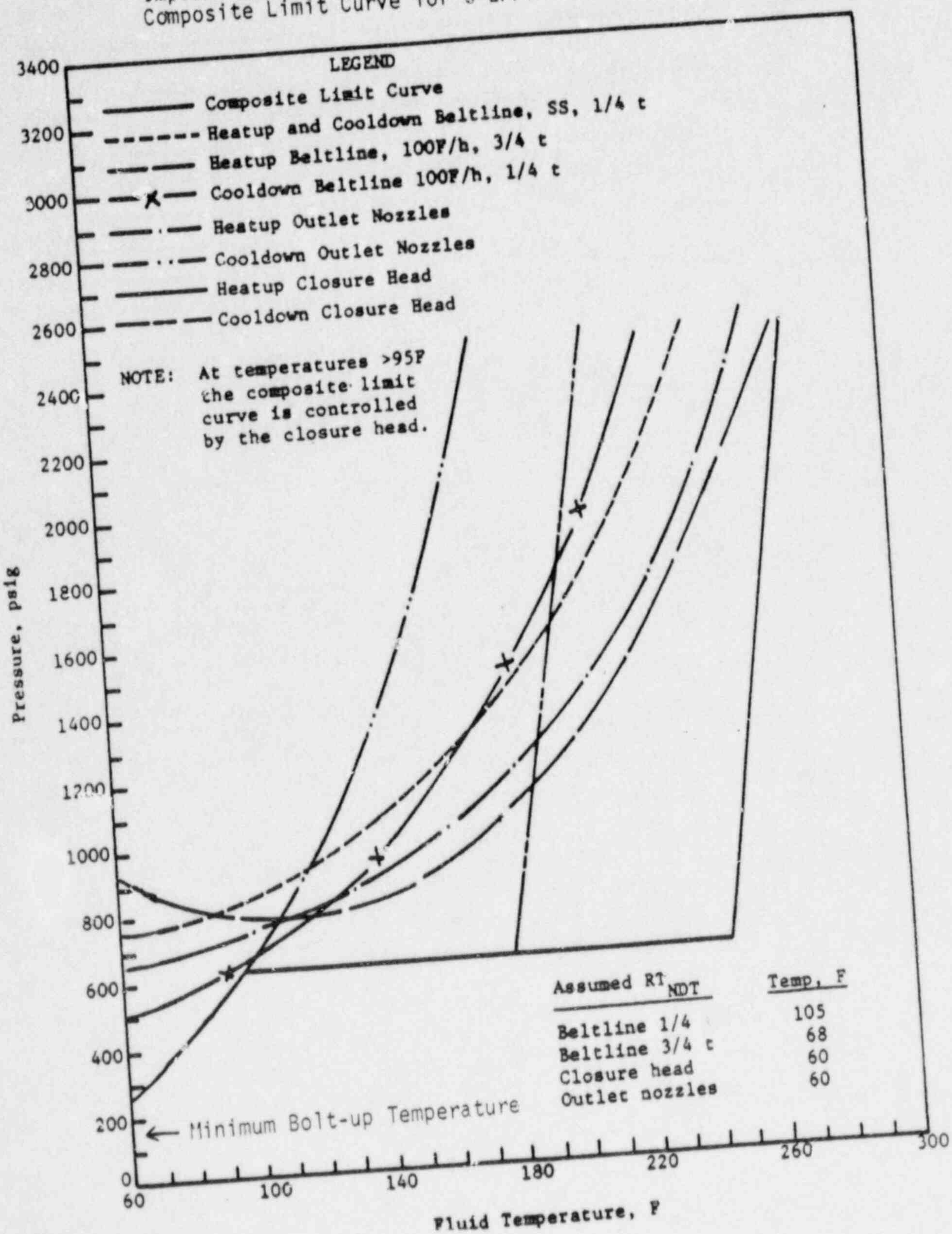

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Figure 5-7. Determination of Reactor Core Operation Pressure-Temperature Curve for 5 EFPY per Appendix G to 10 CFR 50

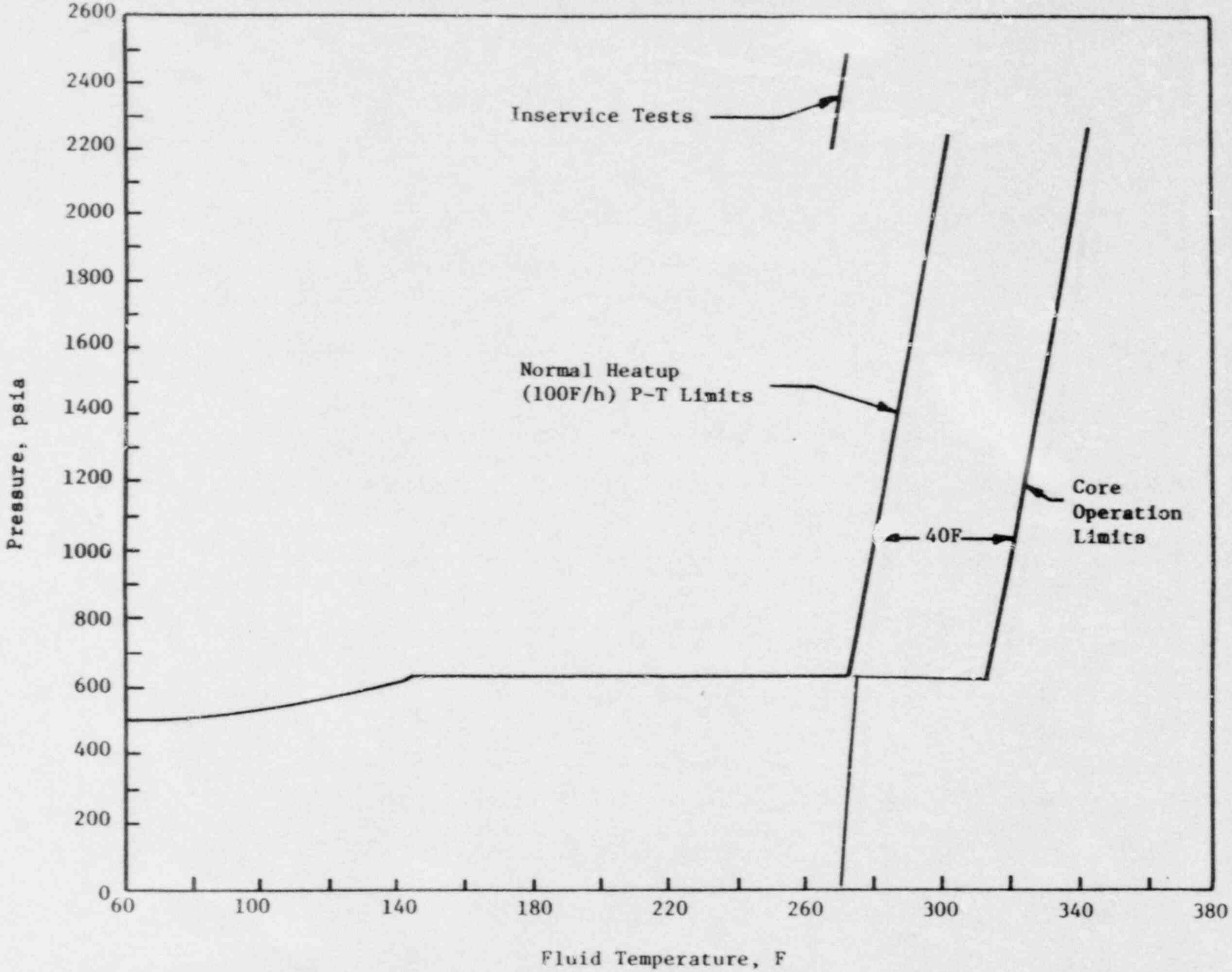


Figure 5-8. Normal Operation Heatup Pressure-Temperature Limit Curves for Typical Plant Technical Specifications, Applicable up to 5 EFPY

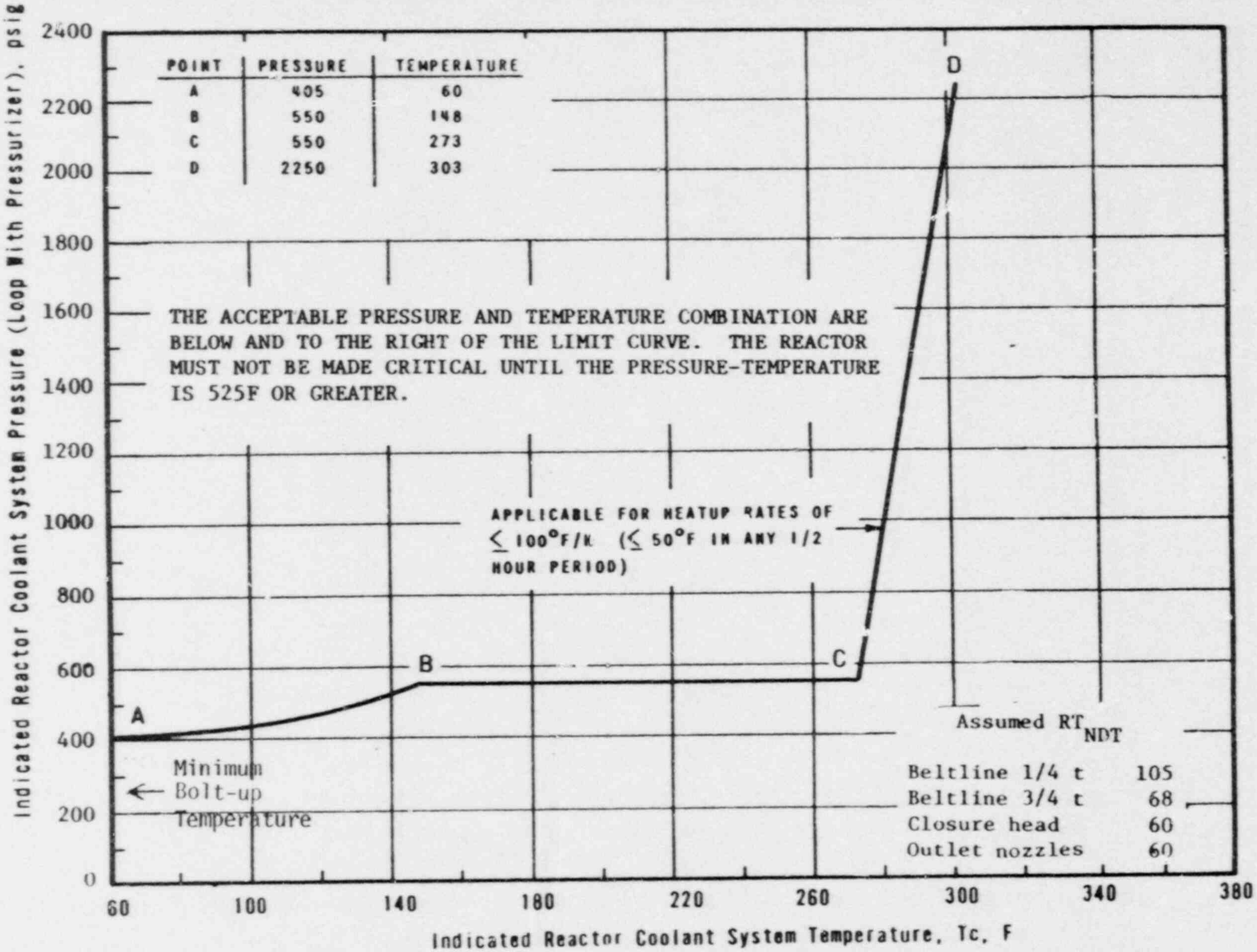


Figure 5-9. Normal Operation Cooldown Pressure-Temperature Limit Curve for Typical Plant Technical Specification, Applicable up to 5 EFPY

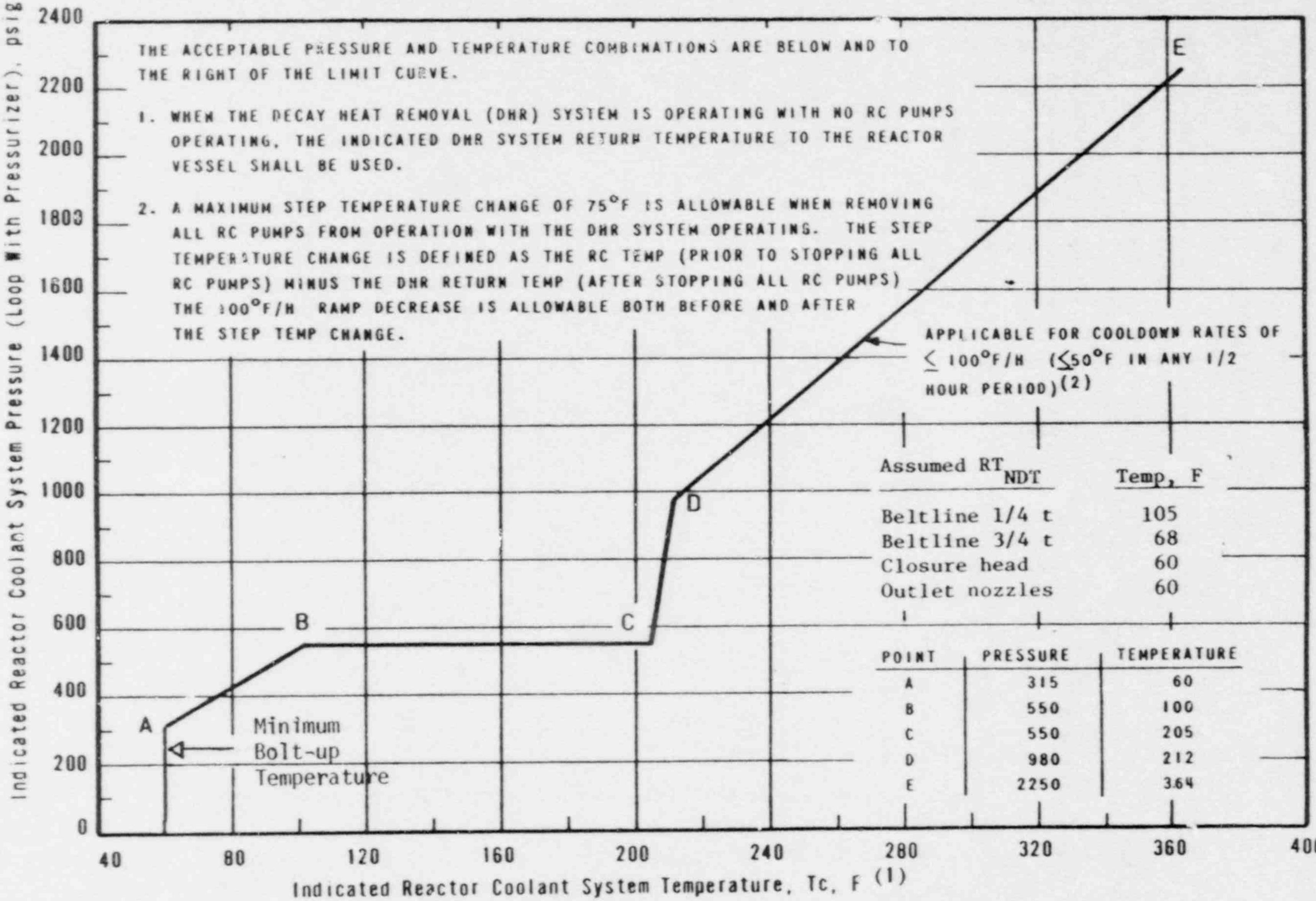


Figure 5-10. Inservice Leak and Hydrostatic Test Heatup and Cooldown Pressure-Temperature Limit Curve for Typical Plant Technical Specifications, Applicable up to 5 EFPY

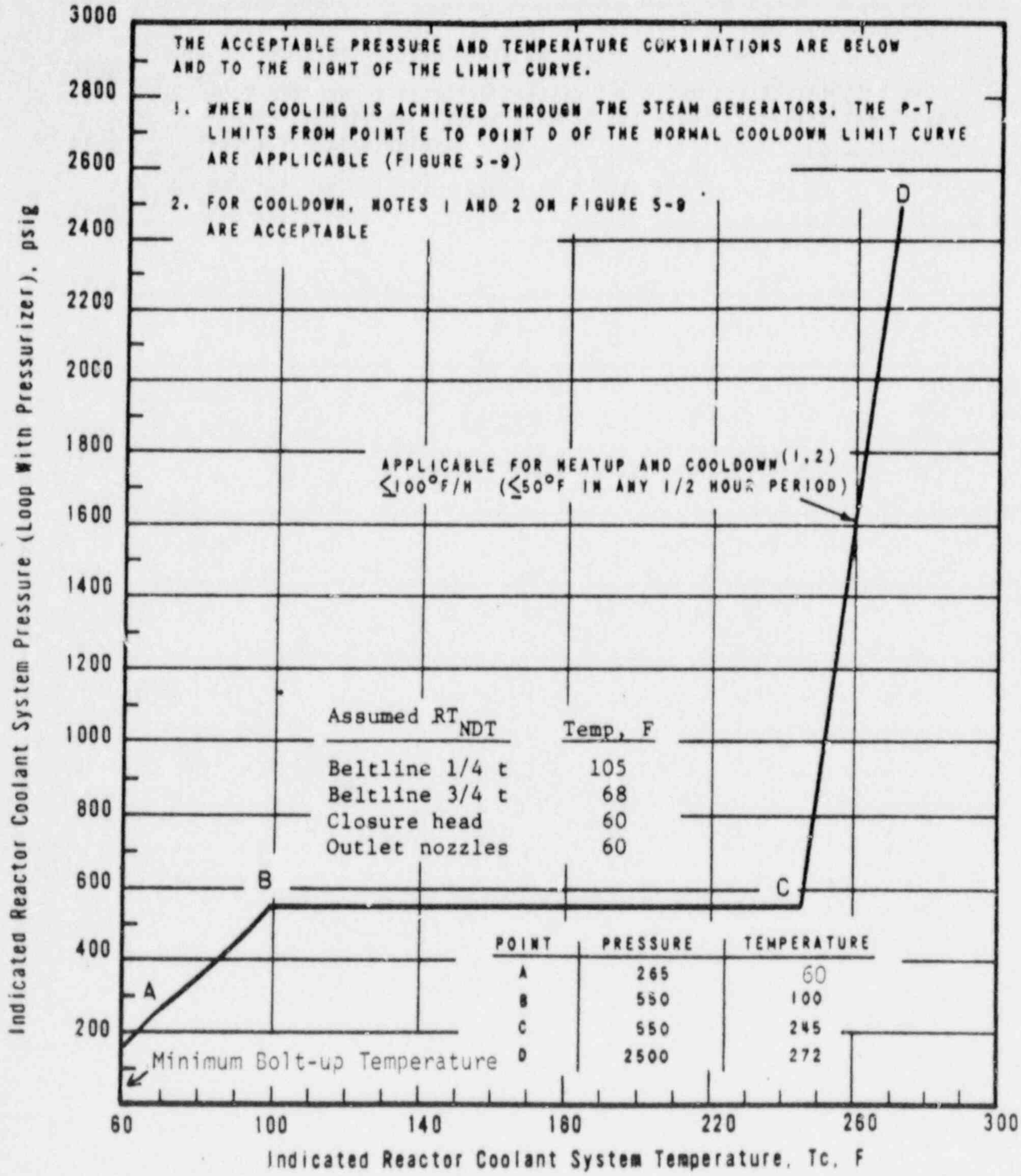
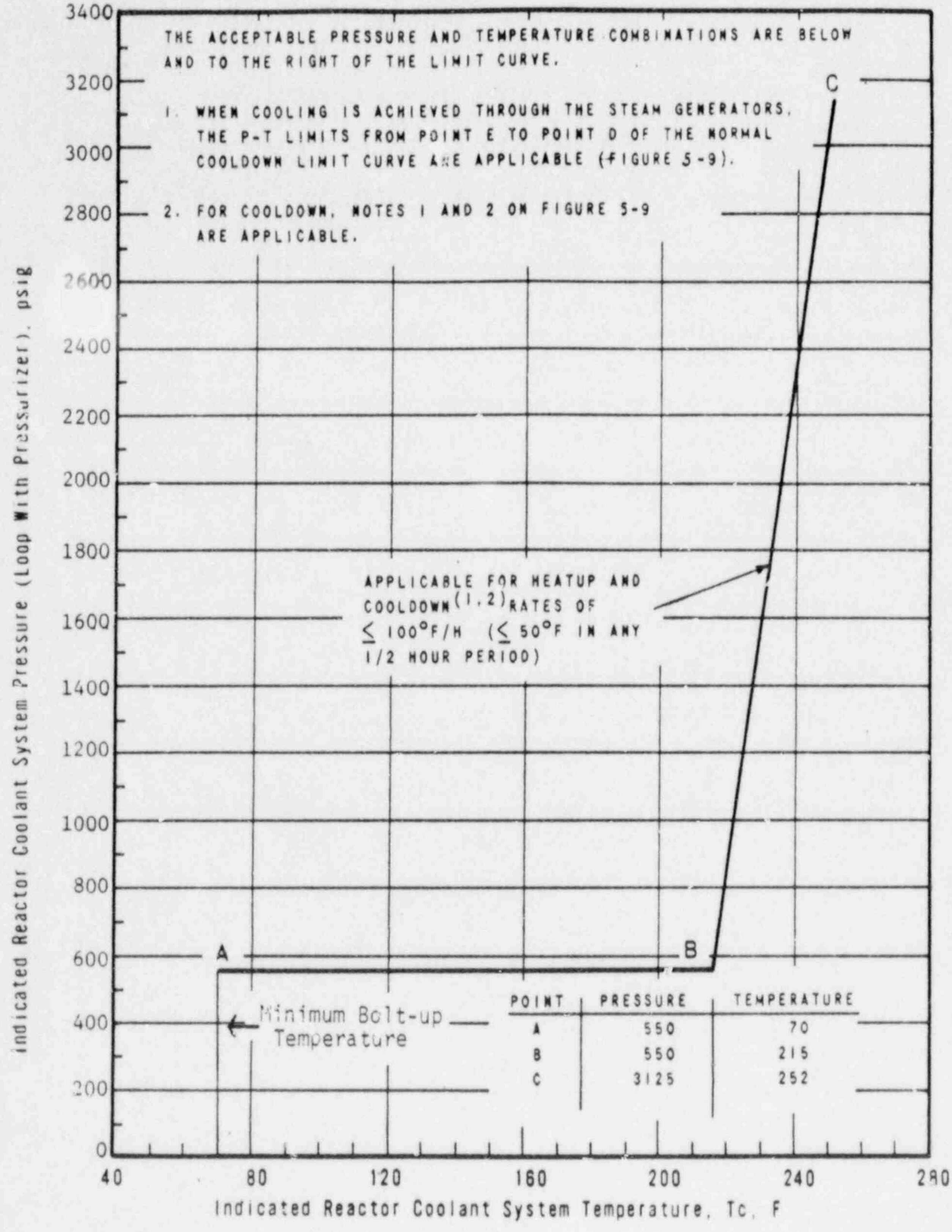


Figure 5-11. PSHT Pressure-Temperature Limit Curve for Typical Plant



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6. EPFM ANALYTICAL PROCEDURES

6.1. Basis

The analytical procedures given in Section 4 are applicable for the areas of the pressure boundary which comply with the material restrictions of ASME Appendix G. If the material does not comply with the restrictions then supplemental analysis is required to assure the reactor coolant pressure boundary integrity. The only anticipated divergence from the materials restrictions is the failure of irradiated materials, in particular weld metal, to exhibit a Charpy upper shelf exceeding 50 ft-lbs.

If a material exhibits less than 50 ft-lbs absorbed energy but greater than 30 ft-lbs the adjusted shift in RT_{NDT} is determined in accordance with 10 CFR 50 Appendix G. The analysis of section 4 is carried out in the same manner previously discussed. 10 CFR 50 Appendix G further specifies that Charpy upper shelves below 50 ft-lbs are permitted if the component is verified to still have a margin of safety equivalent to that specified in ASME Appendix G. The only area of the reactor coolant boundary which is predicted to potentially fall below 50 ft-lbs is the reactor vessel beltline. This evaluation will be restricted to that area but similar evaluation could be performed on other areas.

Appendix G of the ASME Code is a design guideline for the prevention of non-ductile failure. The general philosophy is to index the fracture toughness to temperature and require that the component be operated at a sufficiently high temperature to preclude non-ductile failure. ASME Appendix G is not adequate to control operating conditions in the higher temperature regime. In the high temperature regime ductile tearing is the controlling mechanism for possible loss of vessel integrity. Evaluation for ductile tearing can be accomplished utilizing the J-integral and the J_I-R curve for the material.

6.2. Elastic-Plastic Fracture Mechanics (EPFM) Analytical Model

An elastic-plastic fracture mechanics (EPFM) procedure based on deformation plasticity J-integral solutions in the format of a failure assessment diagram will be used to set the pressure-temperature limits for upper shelf material behavior.

The procedure for setting these pressure-temperature limits consists of four steps:

1. J-integral formulation.
2. Failure assessment diagram curve generation.
3. Assessment point evaluation.
4. Instability pressure prediction.

For reactor vessel materials which can be modeled by deformation plasticity and whose stress-strain behavior can be represented by a power law strain-hardening equation, the J-integral response ($J_{applied}$) can be evaluated for the reference flaw using the expression

$$J = J^e(a_{eff}, P) + J^P(a, P, n) \qquad (1)$$

where J^e is the elastic contribution based on Irwin's effective crack depth, a_{eff}, and J_p is the deformation plasticity contribution derived in reference 6. P is the applied pressure and n is the strain-hardening exponent. A convenient way to use this equation is to construct a deformation plasticity failure assessment diagram (DPFAD). The details of this procedure are found in references 10 and 11. The process is summarized here only for the beltline flaw evaluation.

6.2.2. DPFAD Curve Generation

The DPFAD curve expression is obtained by normalizing the sum of the elastic and plastic response by the "elastic" J-integral of the flawed reactor vessel in terms of "a", where

$$J^e(a) = \frac{(1 - \nu^2) K_I^2(a)}{E} \qquad (2)$$

and K_I is the linear-elastic fracture mechanics (LEFM) stress intensity factor. E and ν are Young's modulus and Poisson's ratio, respectively. The normalized J-response is then defined by

$$K_r = \sqrt{\frac{J^e}{J}} = f(S_r) \tag{3}$$

where $S_r = P/P_L(a)$.

P is the applied pressure and P_L is the reference plastic collapse pressure or limit pressure, a function of "a" and the material yield strength, σ_0.

Equation 3 defines a DPFAD curve which is a function of the flaw geometry, structural configuration, and stress-strain behavior of the material of interest. This curve, in terms of K_r, S_r is independent of the magnitude of the applied loading.

For the beltline area of the reactor vessel assuming a semi-elliptical axial flaw on the inside of the vessel.[9]

$$K_I = \frac{PR_i}{t} \sqrt{\frac{\pi a}{Q}} \quad F(a/\ell, a/t) \tag{4}$$

where
P = applied pressure,
R_i = inside radius,
t = thickness,
$F = .97[M_1 + M_2(a/t)^2 + M_3\,(a/t)^4]fc$,
$M_1 = 1.13 - 0.18\,{}^a/\ell$,
$M_2 = -.54 + .445/\,(.1 + a/\ell)$,
$M_3 = .5 - 1/(.65 + 2a/\ell) + 14\,(1-2a/\ell)^{24}$,
$fc = 1.152 - .05\,\sqrt{a/t}$.

$Q = 1 + 4.593(a/\ell)^{1.65}$
ℓ = length of flaw
a = flaw depth

then

$$J^e(a) = \frac{P^2R_i^2}{t^2} \; \frac{\pi a}{Q} \, \frac{F^2\,(1-\nu^2)}{E} \tag{5}$$

The effective crack correction is given by

$$a_{eff} = a + \frac{1}{6\pi}\,\frac{(n-1)}{(n+1)}\,\frac{K_I^2}{\sigma_0^2}\,\phi; \; \phi = \frac{1}{1 + S_r^2}$$

where
n = strain hardening exponent, (Ramberg-Osgood)
σ_0 = engineering yield stress,
$S_r = P/P_L$,

$$P_L = \frac{2}{\sqrt{3}}\sigma_o \frac{(t - a^*)}{(R_i + a^*)}$$

The limit pressure expression P_L is based on a continuous axial flaw. A correction is applied in the form of a* to account for the partial length flaw.

$$a^* = \frac{a(1 - s)}{1 - (a/t)s} \tag{6}$$

where:

$$s = (1 + \ell^2/2t^2)^{-1/2}$$

The plastic portion of J is given by the following expression

$$J^P = \frac{\alpha\sigma_0^2}{E} a(1 - a/t)h_1(P/P_L)^{n+1} \tag{7}$$

where α is obtained from the Ramberg-Osgood stress-strain relation and h_1 is a dimensionless term which is a function of a/t, a/ℓ, n and t/R_i. This latter constant is evaluated from finite element results.[8]

Combining all of the above terms into equation (3) results in an equation which when plotted has the shape shown in Figure 6-1. The DPFAD curve is unique for a given set of stress-strain parameters, flaw size and vessel geometry.

6.2.2. Assessment Point Evaluation

Having defined the DPFAD curve the beltline of the vessel can be evaluated for a given set of material properties. Assessment points are denoted by K_r', S_r' and are defined as follows:

$$K_r'(a_o + \Delta a) = \sqrt{\frac{J^e(a_o + \Delta a)}{J_R\ (\Delta a)}} \tag{8}$$

$$S_r'(a_o + \Delta a) = \frac{P}{P_L\ (a_o + \Delta a)} \tag{9}$$

where terms are as defined in section 6.2.1 with $J_R(\Delta a)$ being the material J_I-R resistance property. a_o is the initial assumed flaw.

6.2.3. Instability Pressure Prediction

To evaluate the structure, the applied pressure is held constant and successive points are calculated incrementing the crack size. The assessment point which is the minimum distance from the origin represents the maximum crack growth which the structure can sustain before becoming unstable. The corresponding point on the DPFAD curve then represents the instability pressure designated by P_{crit}. This process is illustrated in Figure 6-2.

If the initial point evaluated is $J_R(\Delta a) = J_{IC}$ then the pressure can be determined which corresponds to the initiation of ductile tearing. This pressure is designated P_{init} and is also illustrated on Figure 6.2.

6.3. Sample Calculation and Presentation of Data

For further clarification of the failure assessment diagram approach to predicting tearing pressure, a sample calculcation is presented of an ASME Section III, Appendix G flaw in a beltline weld in a reactor pressure vessel under a pressure of 2500 psi.

Figure 6-3 and Table 6-1 present the FAD format while Figure 6-4 presents a plot of the J_R (Δa) curve for the weld material. Figure 6-5 shows the resultant tearing pressure versus stable crack growth, Δ a, as well as the local plastic instability pressure calculated by the ratio 2500/S_r' (S_r' is given in Table 6-1 as a function of Δa). The critical pressure is the lower value of the two curves. In all the figures and the table, the numbers refer to the points plotted; initiation is the point numbered #1 ($J=J_{IC}$) while the instability or the critical point is numbered #5.

6.4. Thermal Stress

Thermal stresses are not considered when evaluating ductile tearing. Thermal stresses arising from radial gradients through the wall are self-limiting and will decrease with crack propagation. Furthermore, for the conditions being considered (i.e., normal and upset transients in the power operating range) the thermal contribution to the J applied is small calculated on an elastic basis.

6.5. Acceptance Criteria

The acceptance criteria for the evaluation are two fold. Although the flaw used in the evaluation is hypothetical it is necessary to demonstrate that ductile initiation will not occur-to preclude assuming incremental reference flaws throughout the life of the plant. Therefore, the first criteria is that the initiation pressure, P_{init}, must be greater than 3000 psi. This value is one-third above the nominal operating pressure and ten-percent above any normal or upset anticipated transients.

The second criteria is that the instability pressure, Pcrit, must exceed two times the highest level A or B operating pressure. For B&W designed nuclear vessels this corresponds to 5500 psi. These criteria will be reflected in the Owner's licensing document by specifying a maximum allowed pressure in the Technical Specification of 2750 psi for temperatures in the operating range.

7. SUMMARY AND CONCLUSIONS

B&W's methods of compliance with the material properties and operational limit requirements of Appendix G to 10 CFR 50 have been described. Since Appendix G specifies fracture toughness requirements for the ferritic materials used in the RCPB and provides guidelines for determining its operating limitations, the RCPB is described first.

Section 2.3 describes the operational parameters of each loading condition for which pressure-temperature limit curves are required; these conditions are as follows:

1. Normal operation, including bolt preloading, heatup, and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic test.
4. Reactor core operations.

Section 3 describes the methods of compliance with these material requirements. Section 3.1 covers ferritic materials other than bolting and type 403 stainless steels. As required by Appendix G to 10 CFR 50, the RT_{NDT}s of these materials must be established in order to determine the pressure-temperature limit curves for the RC system. For later plants (ordered according to the Summer 1972 Addenda to ASME Section III or subsequent editions or addenda), the RT_{NDT}s were obtained as required by the applicable ASME Code. The RT_{NDT}s of the other ferritic materials in the older plants were conservatively estimated using the fracture toughness data obtained on low-alloy steel forgings, plates, carbon steel plates, weld metals, HAZs and piping.

Appendix G (10 CFR 50) also requires full Charpy test curves on the beltline region materials to determine the USEs for the more recent reactor vessels. B&W has specified complete Charpy test curves (normal and parallel to the principal working direction) on the base metals; for weld metals, only one curve is needed. These curves were also obtained for the HAZs of the beltline region base metals(s) selected to be monitored by the reactor vessel surveillance program. For older reactor vessels, Charpy test curves (both directions) were obtained on the materials from the surveillance program. Where enough material was available for testing, Charpy test curves for specimens oriented normal to the principal working direction were obtained for materials not included in the program. For any beltline region materials for which no test material was available, the Charpy USE was conservatively estimated from data obtained on beltline region low-alloy steel forgings, plates, and weld metals.

The fracture toughness properties of bolting materials are discussed in section 3. The requirements and acceptance criteria of Appendix G to 10 CFR 50 are compared to those of ASME Section III. Since the bolting materials ordered before August 16, 1973, meet only the requirements of the applicable ASME Code, it is demonstrated that these materials have adequate toughness for protection against nonductile failure.

The fracture toughness of type 403 modified steel is covered in section 3.3. The test results demonstrate that the material has adequate fracture toughness for protection against failure at 40F. However, B&W specifies a minimum service temperature of 100F for the CRDM, which provides appropriate conservatism.

Section 3.4 describes the supplemental material properties generated to assess the reactor vessel for resistance to ductile tearing instability. These properties are the stress-strain characteristics and the materials resistance to ductile tearing as a function of crack growth. This section also discusses the method of merging the LEFM criteria of section 4 with the EPFM criteria of section 6.

Section 4 describes the basis for the methods used by B&W to obtain the pressure-temperature limit curves. The calculational procedures are primarily based on ASME Appendix G and WRC Bulletin 175. The method of determining the pressure-temperature limit is described for each loading condition of interest.

In section 5, the methods presented and described in sections 3 and 4 are applied to a typical 177 FA plant. Figures in section 5 illustrate (1) the development of the composite limit curves for each loading condition of interest, (2) the development of the reactor criticality limit curve, (3) the limit curves appearing in the Technical Specifications for a typical plant, and (4) the pressure-temperature limits for the preservice system hydrostatic test.

In section 6 the methods are described for qualifying the reactor vessel in the event that a Charpy upper shelf energy of 50 ft-lbs is not obtained. This section determines the pressure limits for ductile tearing instability and states the acceptance criteria. The basis of this analysis is the J-integral and the supplemental fracture toughness data described in paragraph 3.4.

As described in this report, the fracture toughness requirements imposed on the ferritic materials of pressure-retaining components of the RCPB of B&W reactor coolant systems are in compliance with the fracture toughness requirements of Appendix G to 10 CFR 50. In addition, the report demonstrates that the ferritic materials ordered before the effective date of Appendix G to 10 CFR 50 have adequate toughness for protecting against nonductile failure when the system is operated in compliance with the pressure-temperature limit curves developed by B&W. The analytical method employed by B&W to calculate the maximum allowable pressure of the RC system as a function of fluid temperature includes all the margins of safety required by Appendix G.

8. REFERENCES

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