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PRESSURE-TEMPERATURE LIMITS FOR 10 EFPY
THE TOLEDO EDISON COMPANY
DAVIS-BESSE NUCLEAR POWER STATION - UNIT 1

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CONTENTS

1. INTRODUCTION

This report presents the Davis-Besse Unit 1 reactor coolant pressure boundary pressure-temperature operating limits for 10 EFPY. The data used to develop these limitations are based on the analysis of Davis-Besse Nuclear Power Station Unit 1 reactor vessel surveillance capsule TE1-A as reported in BAW-1882.[1] The report contains data which supports the development of the pressure-temperature limits for normal operation, both heatup and cooldown, inservice leak and hydrostatic tests and reactor core operation for 21 EFPY. To minimize the impact of the effects of neutron radiation induced changes on the operating limitations, the pressure-temperature limitations, as defined in this report, were calculated for 10 EFPY. These limits are adequate for current operations and are justified by the data obtained from the first three surveillance capsules as presented in BAW-1882. New pressure-temperature limitations for additional EFPY operation will be developed before the reactor approaches the limits presented in this report.

In addition, the revised technical specifications and pressure-temperature limits as adjusted for Davis-Besse Nuclear Power Station Unit 1 through 10 EFPY are contained in Appendix A.

2. DETERMINATION OF REACTOR COOLANT PRESSURE BOUNDARY PRESSURE-TEMPERATURE LIMITS

The pressure-temperature limits of the reactor coolant pressure boundary (RCPB) of Davis-Besse Unit 1 are established in accordance with the requirements of 10CFR50, Appendix G.[2] The methods and criteria employed to establish operating pressure and temperature limits are described in topical report BAW-10046A.[3] The objective of these limits is to prevent nonductile failure during any normal operating condition, including anticipated operation occurrences and system hydrostatic tests. The loading conditions of interest include the following:

1. Normal operations, including heatup and cooldown.
2. Inservice leak and hydrostatic tests.
3. Reactor core operation.

The major components of the RCPB have been analyzed in accordance with 10CFR50, Appendix G. The closure head region, the reactor vessel outlet nozzle, and the beltline region have been identified as the only regions of the reactor vessel (and consequently of the RCPB) that regulate the pressure-temperature limits. Since the closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt preload), this region largely controls the pressure-temperature limits of the first several service periods. The reactor vessel outlet nozzle also affects the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle, which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} of the beltline region materials will be high enough that the beltline region of the reactor vessel will start to control the pressure-temperature limits of the RCPB. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, the outlet nozzle, and the beltline region. The maximum allowable pressure is taken to be the lowest of the three calculated pressures.

The limit curves for Davis-Besse Unit 1 are based on the predicted values of the adjusted reference temperatures of all the beltline region materials at the end of the tenth EFPY. The tenth EFPY was selected because it is estimated that the third surveillance capsule will be withdrawn at the end of the refueling cycle when the estimated capsule fluence corresponds to approximately the T/4 end-of-life value. The time difference between the withdrawal of the third and the currently available surveillance capsule data provides adequate time for re-establishing the operating pressure and temperature limits for the period of operation beyond the third surveillance capsule withdrawal.

The unirradiated impact properties were determined for the surveillance beltline region materials in accordance with 10CFR50, Appendixes G and H. For the other beltline region and RCPB materials for which the measured properties are not available, the unirradiated impact properties and residual elements, as originally established for the beltline region materials, are listed in Table 2-1. The adjusted reference temperatures are calculated by adding the predicted radiation-induced RT_{NDT} and the unirradiated RT_{NDT}. The predicted RT_{NDT} is calculated using the respective neutron fluence and copper and nickel contents. Figure 2-1 illustrates the calculated peak neutron fluence at several locations through the reactor vessel beltline region wall. The supporting information for Figure 2-1 are the predicted fluences that have been demonstrated in BAW-1882[1] to be conservative. The design curves of Regulatory Guide 1.99, Rev. 2,[4] were used to predict the radiation-induced RT_{NDT} values as a function of the material's copper and nickel content and neutron fluence.

The neutron fluences and adjusted RT_{NDT} values of the beltline region materials at the end of the tenth full-power year are listed in Table 2-1. The neutron fluences and adjusted RT_{NDT} values are given for the 1/4T and 3/4T vessel wall locations (T = wall thickness). The assumed RT_{NDT} of the closure head region and the outlet nozzle steel forgings is 60F, in accordance with BAW-10046A.

Figure 2-2 shows the reactor vessel's pressure-temperature limit curve for normal heatup. This figure also shows the core criticality limits as required by 10CFR50, Appendix G. Figures 2-3 and 2-4 show the vessel's pressure-temperature limit curve for normal cooldown and for heatup during inservice leak and hydrostatic tests, respectively. All pressure-temperature limit curves are applicable through ten EFPY. Protection against nonductile failure is ensured by maintaining the coolant pressure below the upper limits of the pressure-temperature limit curves. The acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the limit curve. The reactor is not permitted to go critical until the pressure-temperature combinations are to the right of the criticality limit curve. To establish the pressure-temperature limits for protection against nonductile failure of the RCPB, the limits presented in Figures 2-2 through 2-4 must be adjusted by the pressure differential between the point of system pressure measurement and the pressure on the reactor vessel controlling the limit curves. This is necessary because the reactor vessel is the most limiting component of the RCPB.

Table 2-1. Data for Preparation of Pressure-Temperature Limit Curves for Davis-Besse Unit 1 Reactor Vessel -- Applicable Through 10 EFPY

Material Heat No.	Identification Type	Beltline Region Location	Weldment Location: Core Midplane to Weld CL, cm	Weldment Location: Location from Major Axis, Degrees	Weldment Location: Weld T/4 Location	Inside Surface Fluence, n/cm^2 [d]	Chemical Composition[c]: Copper, w/o	Chemical Composition[c]: Nickel, w/o	Chemistry Factor	Initial RT_{NDT}, F	Radiation Induced ΔRT_{NDT} at End of 10 EFPY, F[a]: T/4[e]	Radiation Induced ΔRT_{NDT} at End of 10 EFPY, F[a]: 3T/4[e]	Margin, F	Adjusted RT_{NDT} at End of 10 EFPY, F[a]: T/4	Adjusted RT_{NDT} at End of 10 EFPY, F[a]: 3T/4
ADB 203	SA508, Cl 2	Nozzle Belt	--	--	--	9.01E17	0.04	0.68	26	+50	8	4	10/6	68	60
AKJ 233	SA508, Cl 2	Upper Shell	--	--	--	5.63E18	0.04	0.77	26	+20	18	12	18/12	56	44
BCC 241	SA508, Cl 2	Lower Shell	--	--	--	5.63E18	0.02	0.81	20	+50	14	9	14/8	78	67
WF 232	Weld	Upper Circum Seam (ID 9%)	+198	--	No	9.01E17	0.18	0.64	161	-6F[b]	--	--	--	--	--
WF 233	Weld	Upper Circum Seam (OD 91%)	+198	--	Yes	9.01E17	0.29	0.68	204	-6F[b]	62	35	68/52	124	81
WF 182-1	Weld	Middle Circum Seam (100%)	-24	--	Yes	5.63E18	0.24	0.63	178	+2	125	81	56	183	139
WF 232	Weld	Lower Circum Seam (ID 12%)	-247	--	No	3.15E16	0.18	0.64	161	-6F[b]	--	--	--	--	--
WF 233	Weld	Lower Circum Seam (OD 88%)	-247	--	Yes	3.15E16	0.29	0.68	204	-6F[b]	--	--	--	--	--
WF 182-1	Weld	Calculated RT_{NDT} shift of limiting weld metal based on use of surveillance capsule data per Reg. Guide 1.99, Rev. 2				5.63E18	0.24	0.63	167	+2	117	77	28	[147][f]	[107][f]

(a) RT_{NDT} calculated per draft Regulatory Guide 1.99, Revision 2, dated February 17, 1987 (to be published)

(b) Estimated initial RT_{NDT} of weld metals per BAW 1803, January, 1984 - One Standard Deviation = 19F.

(c) Materials chemical compositions per BAW 1820, December, 1984, and BAW 1799, July, 1983

(d) Fluence data per BAW 1882, September 1985

(e) Reactor Vessel Wall Thickness = 8.5 inches

(f) [ ] Controlling Values of RT_{NDT}

Figure 2-1. Predicted Fast Neutron Fluence at Various Locations Through Reactor Vessel Wall for 10 EFPY - Davis-Besse Unit 1

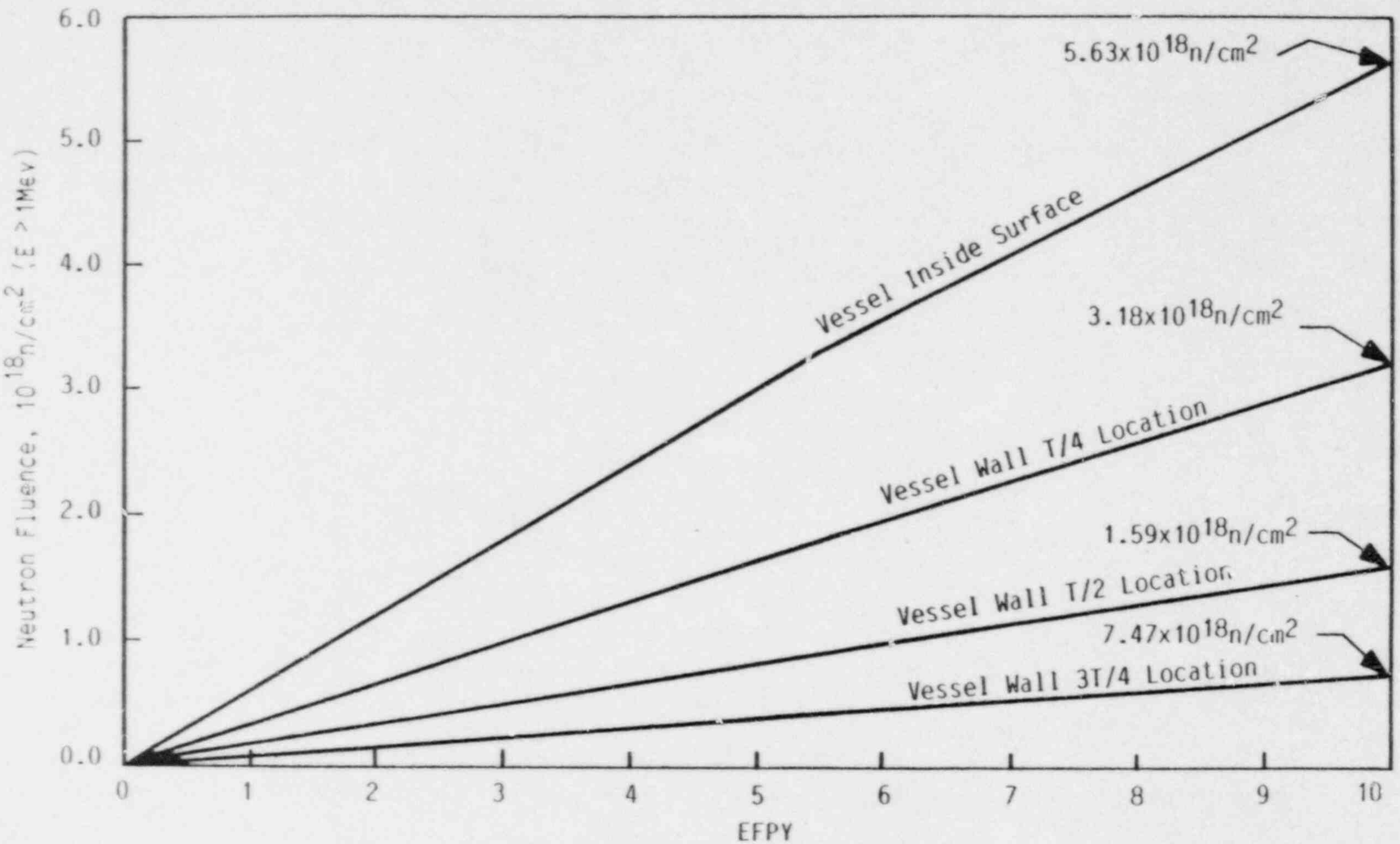


Figure 2-2. Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Heatup, Applicable for First 10 EFPY - Davis-Besse Unit 1

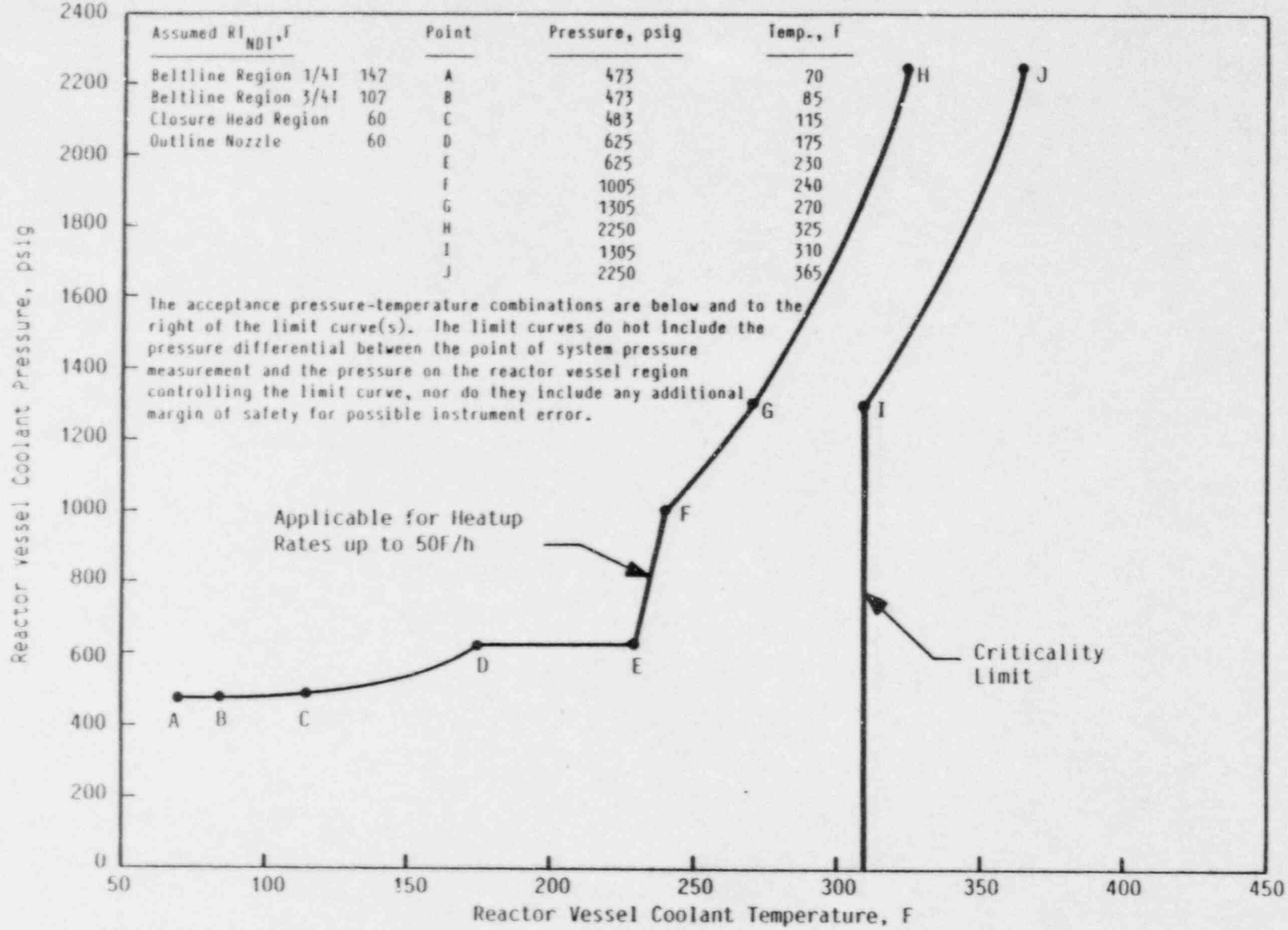


Figure 2-3. Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Cooldown, Applicable for First 10 EFPY - Davis-Besse Unit 1

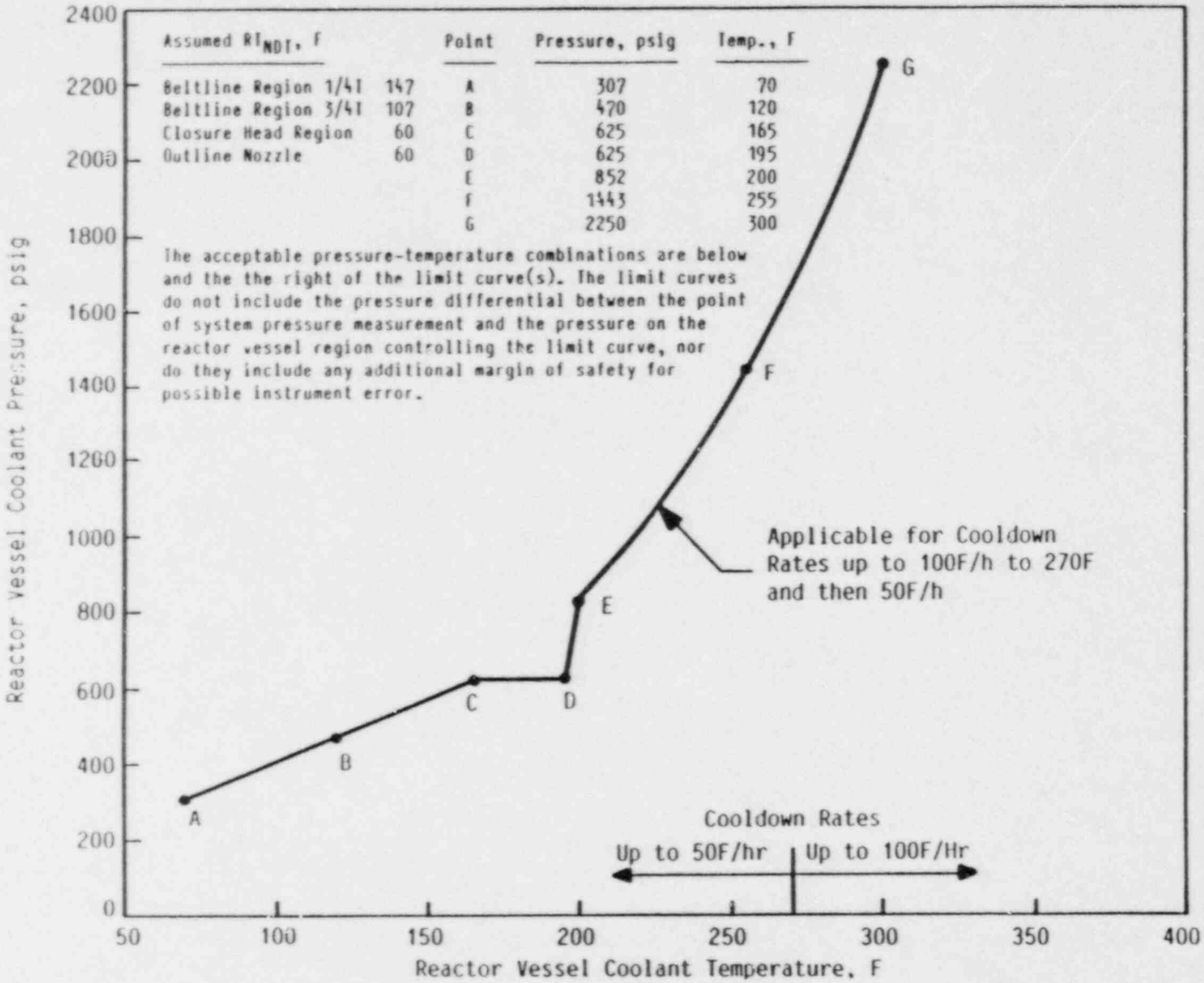
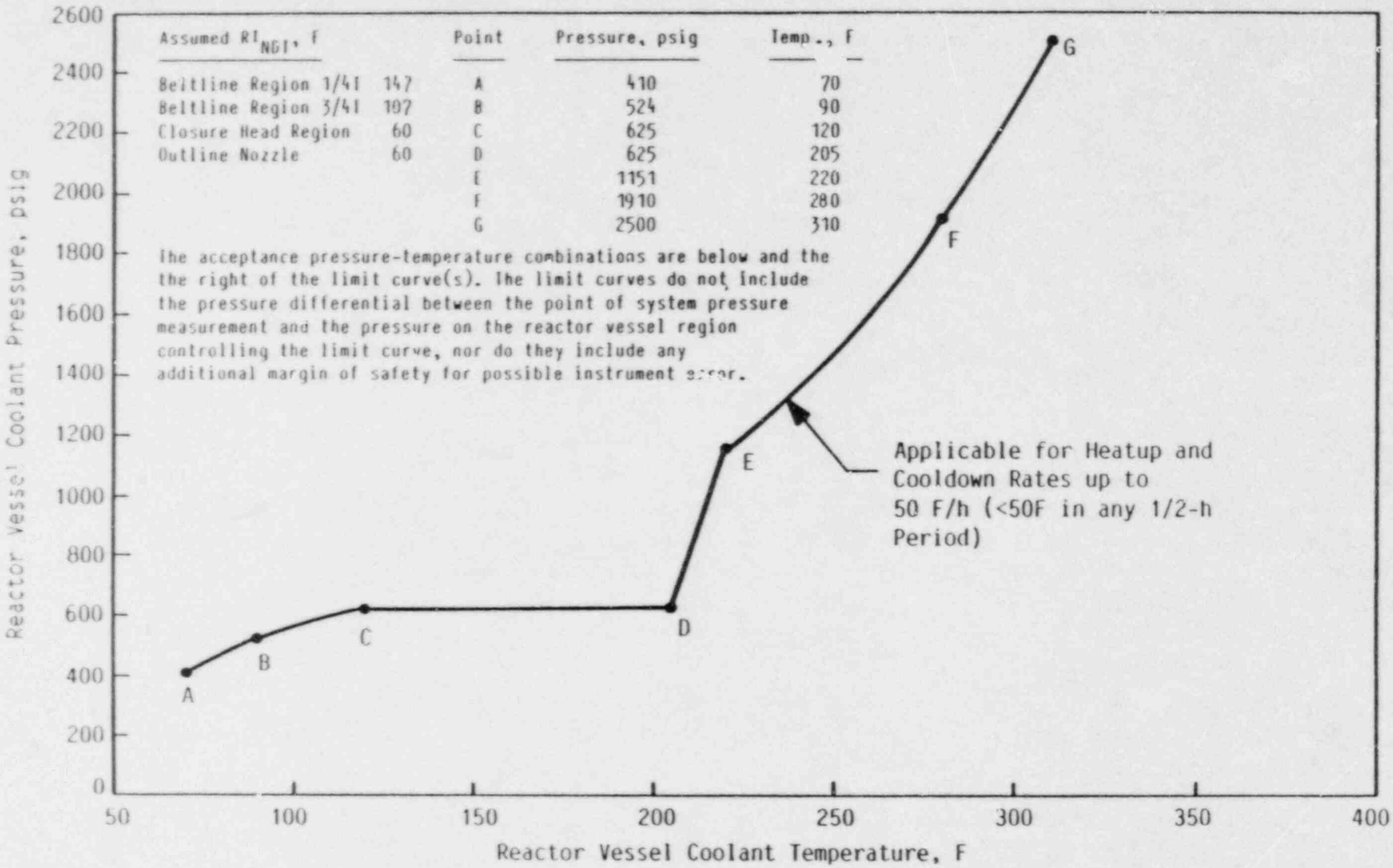


Figure 2-4. Reactor Vessel Pressure-Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 10 EFPY - Davis-Besse Unit 1



Assumed RT_{NDT}, F		Point	Pressure, psig	Temp., F
Beltline Region 1/4T	147	A	410	70
Beltline Region 3/4T	107	B	524	90
Closure Head Region	60	C	625	120
Outline Nozzle	60	D	625	205
		E	1151	220
		F	1910	280
		G	2500	310

3. DEVELOPMENT OF TECHNICAL SPECIFICATION PRESSURE-TEMPERATURE LIMITS

The pressure-temperature limits established for the technical specification were determined for selected heatup and cooldown rates by comparing the individual uncorrected pressure-temperature curves for the nozzle, beltline, and closure head over the operating temperature range of the reactor vessel (see Section 2). The limiting pressure (minimum) at each temperature was selected as the basis for developing the maximum pressure for setting the actual operating limitations. Differential pressure corrections were then applied to the resulting limiting curves to account for the pressure differential between the analyzed regions of the reactor vessel and the system pressure sensor on the reactor coolant system hot leg.

The resulting corrected data points were plotted to obtain a bounding technical specification curve for normal operations. Also, heatup and cooldown curves at various rates (°F/hr) over the various operating temperature ranges were combined into composite, bounding operating limit curves.

The resulting changes to the applicable technical specification section and the revised pressure-temperature curves for Davis-Besse are shown in Appendix A.

4. CERTIFICATION

The pressure-temperature operating limits for Davis-Besse Unit 1 reactor pressure vessel were calculated using approved procedures and established methods and techniques in accordance with the requirements of 10CFR50, Appendix G.

A L Lowe Jr P.E. 13 Nov. '87

A. L. Lowe, Jr., P.E. Date
Project Technical Manager

This report has been reviewed for ... cal content and accuracy.

L B Gross, PE Nov. 16, 1987

L. B. Gross, P.E. (material properties) Date
M&SA Unit

K K Yoon, PE Nov. 11 1987

K. K. Yoon, P.E. (fracture analysis) Date
M&SA Unit

Verification of independent review.

A. D. McKim Nov. 17, 1987

A. D. McKim, Manager Date
M&SA Unit

This report has been approved for release.

J F Walters 11/17/87

J. F. Walters Date
Program Manager

APPENDIX A

Revised Technical Specifications
Pressure-Temperature Operating Limitations

Intentionally
Omitted

APPENDIX B

References

1. A. L. Lowe, Jr., et al., Analyses of Capsule TE1-A, The Toledo Edison Company, Davis-Besse Nuclear Power Station Unit 1, Reactor Vessel Materials Surveillance Program, BAW-1882, Babcock & Wilcox, Lynchburg, Virginia, September, 1985.

2. Code of Federal Regulation, Title 10, Part 50, Fracture Toughness Requirements for Light-Water Nuclear Power Reactors, Appendix G, Fracture Toughness Requirements, Federal Register, Vol. 48, No. 104, May 17, 1983.

3. H. S. Palme, et al., Methods of Compliance With Fracture Toughness and Operational Requirements of Appendix G to 10CFR50, BAW-10046A, Rev. ~~1~~ 2, Babcock & Wilcox, Lynchburg, Virginia, July 1977. B 4/n/

4. U.S. Nuclear Regulatory Commission, Radiation Embrittlement of Reactor Vessel Material, Draft Regulatory Guide 1.99, Revision 2, Dated February 10, 1986.

5. A. L. Lowe, Jr., et al., Pressurized Thermal Shock Evaluations in Accordance With 10CFR50.61 for Babcock & Wilcox Owners Group Reactor Pressure Vessesl, BAW-1895, Babcock & Wilcox, Lynchburg, Virginia, January 1986.

6. A. S. Heller and A. L. Lowe, Jr., Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds, BAW-1803, Babcock & Wilcox, Lynchburg, Virginia, January 1984.

7. J. D. Aadland, Babcock & Wilcox Owner's Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information, BAW-1820, Babcock & Wilcox, Lynchburg, Virginia, December 1984.

8. K. E. Moore and A. S. Heller, BAW 177-FA Reactor Vessel Beltline Weld Chemistry Study, BAW-1799, Babcock & Wilcox, Lynchburg, Virginia, July 1983.

9. A. L. Lowe, Jr., et al., Integrated Reactor Vessel Material Surveillance Program (Addendum), BAW-1543A, Rev. 2, Addendum 1, Babcock & Wilcox, Lynchburg, Virginia, November 1987.