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PSEG
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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
PRESSURE/TEMPERATURE LIMITS
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications for the Hope Creek Generating Station. In accordance with 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed amendment would revise the reactor pressure vessel (RPV) pressure-temperature (P-T) limits and extend the validity of the limits to 32 effective full power years (EFPY). The current P-T limits expire at the end of the current operating cycle (defined as the end of the next refueling outage). The updated P-T limits are based on an RPV neutron fluence calculated using an NRC staff-accepted neutron fluence methodology for boiling water reactors.

PSEG has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2. Attachment 3 provides a description of inputs, methodology and results for the revised pressure-temperature curves.

A001

PSEG requests approval of the proposed License Amendment by September 30, 2004 to be implemented within 60 days of the issue date. The License Amendment is required to permit restart after completion of the Hope Creek Fall 2004 refueling outage.

Should you have any questions regarding this request, please contact Mr. Paul Duke at 856-339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 31 March 2004
(date)

Michael H. Brothers
Vice President - Site Operations

Attachments (3)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS PRESSURE/TEMPERATURE LIMITS

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1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station (HCGS). The proposed amendment would revise the reactor pressure vessel (RPV) pressure-temperature (P-T) limits in Technical Specification (TS) Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-.3 and extend the validity of the limits to 32 effective full power years (EFPY). The updated P-T limits are based on an RPV neutron fluence calculated using an NRC staff-accepted neutron fluence methodology for boiling water reactors. The revised P-T limit curves satisfy the requirements of Appendix G to 10CFR Part 50.

2. PROPOSED CHANGE

The marked up pages for the proposed changes to the Technical Specifications are included in Attachment 2 of this submittal.

The current TS Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3, which establish P-T limitations for the reactor coolant system, will be replaced by the figures in Attachment 2. The updated P-T curves are valid through the end of the 40-year operating license and are based on a re-calculation of neutron fluence using an NRC staff-accepted neutron fluence methodology for boiling water reactors (BWRs) which is consistent with the requirements of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 1).

Changes to the TS Bases would also be made to reflect the changes to the P-T limits. The marked up Bases pages are also included in Attachment 2 of this submittal.

PSEG plans to implement the proposed change to support the next refueling outage (i.e., Fall 2004) and subsequent restart. Because the current set of P-T curves expire at the end of the current operating cycle (defined as the end of the next refueling outage), a license amendment is required before the end of the refueling outage. The next refueling outage is currently scheduled to begin in October 2004.

3. BACKGROUND

10 CFR 50.60 requires that light-water nuclear power reactors meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in Appendix G to 10 CFR 50. Appendix G is the regulatory basis for P-T curves for light water reactors. Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of

normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Appendix G also requires that the reference temperature and Charpy upper-shelf energy for reactor vessel beltline materials account for the embrittlement caused by neutron fluence over the life of the vessel.

RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 2) describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels.

The P-T curves approved in Amendment No. 131 (Reference 3) were developed using the methodology specified in American Society of Mechanical Engineers (ASME) Code Cases N-588 and N-640, as well as the 1989 ASME Code, Section XI, Appendix G, and 10 CFR Part 50, Appendix G. Adjusted reference temperatures at the nil ductility transition values were developed for the reactor pressure vessel materials in accordance with RG 1.99, Revision 2. In the Safety Evaluation for Amendment No. 131 the NRC staff concluded that, while there was ample margin to allow use of the revised P-T curves through Cycle 11, the revised P-T curves could not be approved for the 32 EFPY for which they were intended until the fluence values were recalculated using the guidance of RG 1.190.

In Amendment 139 (Reference 4), the NRC staff approved use of the existing P-T limit curves for one additional cycle (through the end of Cycle 12). The estimated fluence at the end of Cycle 12 is less than half of the calculated fluence for 32 EFPY upon which the current P-T limits are based.

10 CFR 50 Appendix G requires reactor vessel beltline materials to be tested in accordance with the surveillance program requirements of 10 CFR 50 Appendix H. In References 5 and 6, PSEG requested a change the HCGS reactor vessel material surveillance program to incorporate the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) into the HCGS licensing basis. The updated fluence calculations described in Section 4 of this Attachment meet the condition for use of compatible neutron fluence methodologies acceptable to the NRC staff.

4. TECHNICAL ANALYSIS

Revised pressure-temperature (P-T) curves were developed for pressure test, core not critical, and core critical conditions. A report describing the inputs, methodology and results for the revised curves is provided in Attachment 3. The

revised curves are applicable for 32 effective full power years (EFPY). Curves applicable for 48 EFPY are included in the report for information only.

The curves were developed using the methodology specified in American Society of Mechanical Engineers (ASME) Code Cases N-588 and N-640 as well as the 1989 ASME Code, Section XI, Appendix G, and 10 CFR Part 50, Appendix G.

ASME Code Case N-640 permits application of the lower bound static initiation fracture toughness value equation (K_{IC} equation) as the basis for establishing the P-T curves in lieu of the lower bound crack arrest fracture toughness value equation (i.e., the K_{IA} equation). The K_{IA} equation is based on conditions needed to arrest a dynamically propagating crack and is the method invoked by Appendix G to Section XI of the 1989 ASME Code. Use of the K_{IC} equation in determining the lower bound fracture toughness in the development of the P-T operating limits curve is more technically correct than the use of the K_{IA} equation because the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 13, January 2004 (Reference 7) states that Code Case N-640 is acceptable to the NRC for application in licensees' Section XI inservice inspection programs.

ASME Code Case N-588 permits postulation of a circumferentially oriented flaw (in lieu of an axially-oriented flaw) for the evaluation of the circumferential welds in RPV P-T limit curves. RG 1.147 states that Code Case N-588 is acceptable to the NRC for application in licensees' Section XI inservice inspection programs.

Neutron Fluence Calculations

The neutron fluence calculations were updated using the NRC-approved General Electric Nuclear Energy (GENE) methodology documented in GENE's Licensing Topical Report NEDC-32983P-A (Reference 8). The NRC-accepted proprietary methodology is fully described in NEDC-32983P-A and is not repeated herein. In general, GENE's methodology is consistent with the guidance in RG 1.190 for neutron flux calculations and is based on a two-dimensional discrete ordinates code.

The fluence is based upon operation for 32 EFPY, with 12 EFPY at 3293 MWt, the original licensed thermal power, 3 EFPY at 3339 MWt, the current licensed thermal power, and the remaining 17 EFPY at 3952 MWt which bounds operation at the current licensed thermal power through the end of the 40-year operating license. The updated fluence evaluation (at 3952 MWt) was performed using the GENE methodology. The calculated RPV flux value for 3293 MWt is based on the neutron transport calculation described in Reference 9 in which the jet pumps were not modeled. Results from the updated fluence evaluation demonstrated that the neutron transport calculation in Reference 9 provided sufficient

conservatism to permit its use for operation at 3293 MWt. The peak flux for 3339 MWt includes a 1.4% uprate from 3293 MWt.

The peak calculated RPV inside surface fluxes are provided below:

Power Level (MWt)	Flux (n/cm^2-s)
3293	9.3×10^8
3952	1.22×10^9

The calculated fast neutron fluences at the end of plant life (32 EFPY) are provided below:

Parameter	Fluence (n/cm^2)
Peak Surface	1.1×10^{18}
Peak 1/4 T	7.6×10^{17}
Limiting Beltline Material Peak Surface	5.3×10^{17}
Limiting Beltline Material Peak 1/4 T	3.7×10^{17}

Regulatory Guide 1.99 and Adjusted Reference Temperature

Adjusted reference temperature (ART_{NDT}) values were recalculated in accordance with Regulatory Guide 1.99, Revision 2 based on the new calculated fluence values.

Upper shelf energy (USE) calculations were performed and confirmed that all USE values are greater than 50 ft-lb throughout RPV life as required by 10 CFR 50 Appendix G. For the limiting USE material, the USE at 32 EFPY decreased from 61 to 60 ft-lb.

Pressure-Temperature Curve Evaluation

Three regions of the reactor pressure vessel (RPV) were evaluated to develop the revised P-T curves: (1) the beltline region, (2) the bottom head region, and (3) the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

The methodology used to generate the P-T curves in this submittal is similar to the methodology used to generate the curves approved in HC TS Amendment No. 131. In this update, however, the estimate of the RPV neutron fluence was based on a new fluence methodology that follows the guidance of Regulatory Guide 1.190.

Non-Beltline Regions

Non-beltline regions are defined as the vessel locations that are remote from the active fuel and where the end-of-life (EOL) neutron fluence is not sufficient (i.e., <

10^{17} n/cm^2) to cause any significant embrittlement. Non-beltline components include nozzles, closure flanges, some shell plates, the top and bottom head plates, and the control rod drive penetrations.

For the feedwater nozzles, an updated ASME Code stress and fatigue analysis demonstrated acceptable fatigue life through the end of the 40-year operating license, including the effects of operation at 3952 MWt which bounds operation at the current licensed thermal power. Calculated stresses from the updated analysis were used to develop pressure-temperature limits for the upper vessel.

The methodologies used to develop the proposed P-T limit curves satisfy the requirements of the regulations (as modified by application of ASME Code Cases N-588 and N-640). The revised P-T curves and outputs from the Integrated Surveillance Program (which when approved by NRC for use at HCGS, will be used, as appropriate, for future adjustments to P-T limits) ensure that adequate RPV safety margins against non-ductile failure will continue to be maintained during normal operations, anticipated operational occurrences, and hydrostatic testing. Together, these measures ensure that the integrity of the reactor coolant system will be maintained for the life of the plant.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The revised curves are based on uprated fluence projections and are applicable for the service period up to 32 effective full power years (EFPY). There are no changes being made to the reactor coolant system (RCS) pressure boundary or to RCS material, design or construction standards. The proposed heatup and cooldown curves define limits that continue to ensure the prevention of nonductile failure of the RCS pressure boundary. The design-basis events that were evaluated have not changed. The modification of the heatup and cooldown curves does not alter any assumptions previously made in the radiological consequence evaluations since the integrity of the RCS pressure boundary is

unaffected. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Revisions to the heatup and cooldown curves do not involve any new components or plant procedures. The proposed changes do not create any new single failure or cause any systems, structures or components to be operated beyond their design bases. Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed figures define the limits for ensuring prevention of nonductile failure for the reactor coolant system based on the methods described in 1989 ASME Code Section XI Appendix G, 10CFR 50 Appendix G, and ASME Code Cases N-640 and N-588. The effect of the change is to permit plant operation within different pressure-temperature limits, but still with adequate margin to assure the integrity of the reactor coolant system pressure boundary. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.60 requires that light-water nuclear power reactors meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in Appendix G to 10 CFR 50. Appendix G is the regulatory basis for P-T curves for light water reactors. Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation,

including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Appendix G also requires that the reference temperature and Charpy upper-shelf energy for reactor vessel beltline materials account for the embrittlement caused by neutron fluence over the life of the vessel. Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 contains the U.S. Nuclear Regulatory Commission (NRC) staff's guidance on how to determine neutron fluence. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 2) describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7. REFERENCES

1. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Revision 0. March 2001

2. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988

3. Hope Creek Generating Station, Issuance of Amendment Re: 1.4% Increase In Licensed Power Level (TAC No. MB0644), July 30, 2001

4. "Hope Creek Generating Station - Issuance of Amendment Re: Use of Existing Pressure-Temperature Curves through Cycle 12 (TAC No. MB4685)," August 13, 2002.

5. LR-N02-0406, "Request for Change to Reactor Material Surveillance Program," dated December 23, 2002

6. LRN-03-0344, "Request for Additional Information on Reactor Pressure Vessel Material Surveillance Program," dated August 14, 2003

7. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 13, January 2004

8. NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," Rev. 1, December 2001

9. GE-NE-523-A164-1294R1, Hope Creek 1 Generating Station, RPV Surveillance Materials Testing and Fracture Toughness Analysis," December 1997.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REQUEST FOR LICENSE AMENDMENT

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
Index	xviii
Figure 3.4.6.1-1	3/4 4-23
Figure 3.4.6.1-2	3/4 4-23a
Figure 3.4.6.1-3	3/4 4-23b
Bases 3/4.4.6	B 3/4 4-5
Bases Table B 3/4.4.6-1	B 3/4 4-7
Bases Figure B 3/4.4.6-1	B 3/4 4-8
Bases Table B 3/4.4.6-2	B 3/4 4-9 (New Page) B 3/4 4-10 (New Page)

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BASES

SECTION PAGE

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Accident Monitoring Instrumentation ..... B 3/4 3-5
Source Range Monitors ..... B 3/4 3-5
Traversing In-Core Probe System ..... B 3/4 3-5

3/4.3.8 DELETED ..... B 3/4 3-7

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION ..... B 3/4 3-7

Figure B3/4 3-1 Reactor Vessel Water Level ..... B 3/4 3-8

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION ..... B 3/4 3-9

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM ..... B 3/4 4-1

3/4.4.2 SAFETY/RELIEF VALVES ..... B 3/4 4-2

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems ..... B 3/4 4-3
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3/4.4.4 CHEMISTRY ..... B 3/4 4-3

3/4.4.5 SPECIFIC ACTIVITY ..... B 3/4 4-4

3/4.4.6 PRESSURE/TEMPERATURE LIMITS ..... B 3/4 4-5

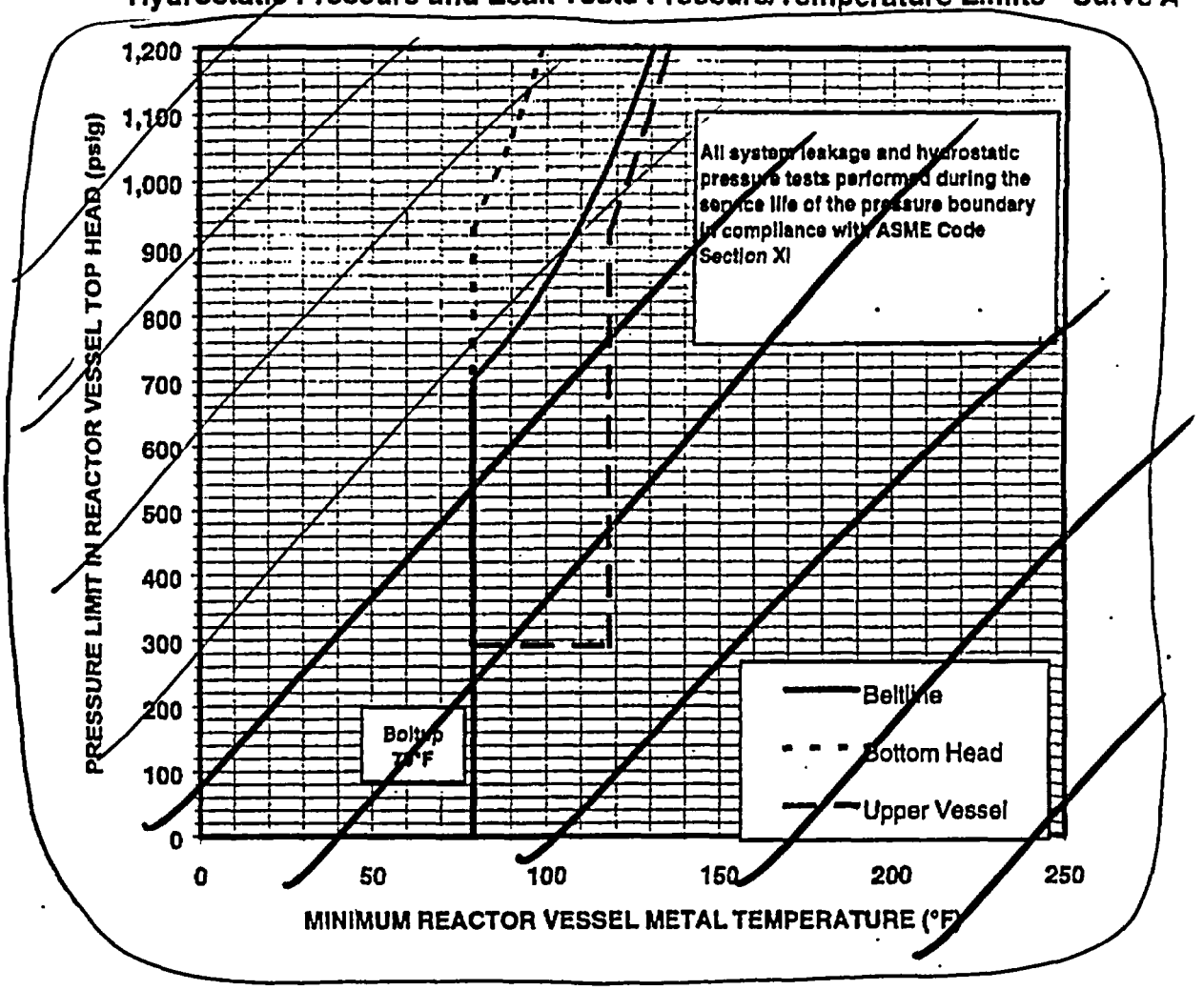
Table B3/4.4.6-1 Reactor Vessel Toughness ..... B 3/4 4-7

Figure B3/4.4.6-1 Fast Neutron Fluence (E>1Mev) at (1/4)T as a Function of Service life ..... B 3/4 4-8

Table B3/4.4.6-2 Numeric Values for Pressure/Temperature Limits ..... B 3/4 4-9

Figure 3.4.6.1-1

Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits - Curve A

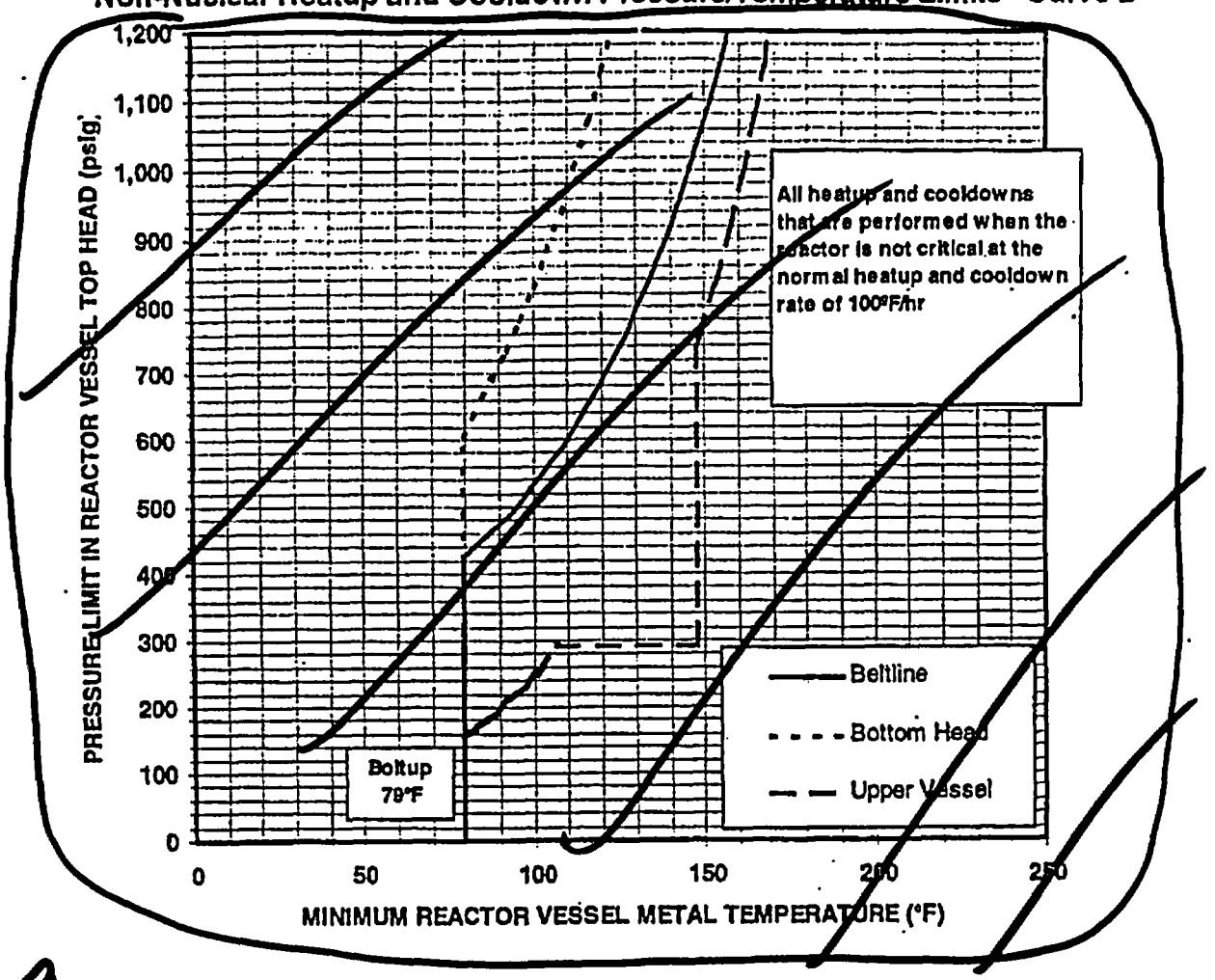


Note: This figure is valid through Cycle 12 Operation in accordance with NRC Safety Evaluation Report supporting Amendment No. 139

INSERT CURVE A

Figure 3.4.6.1-2

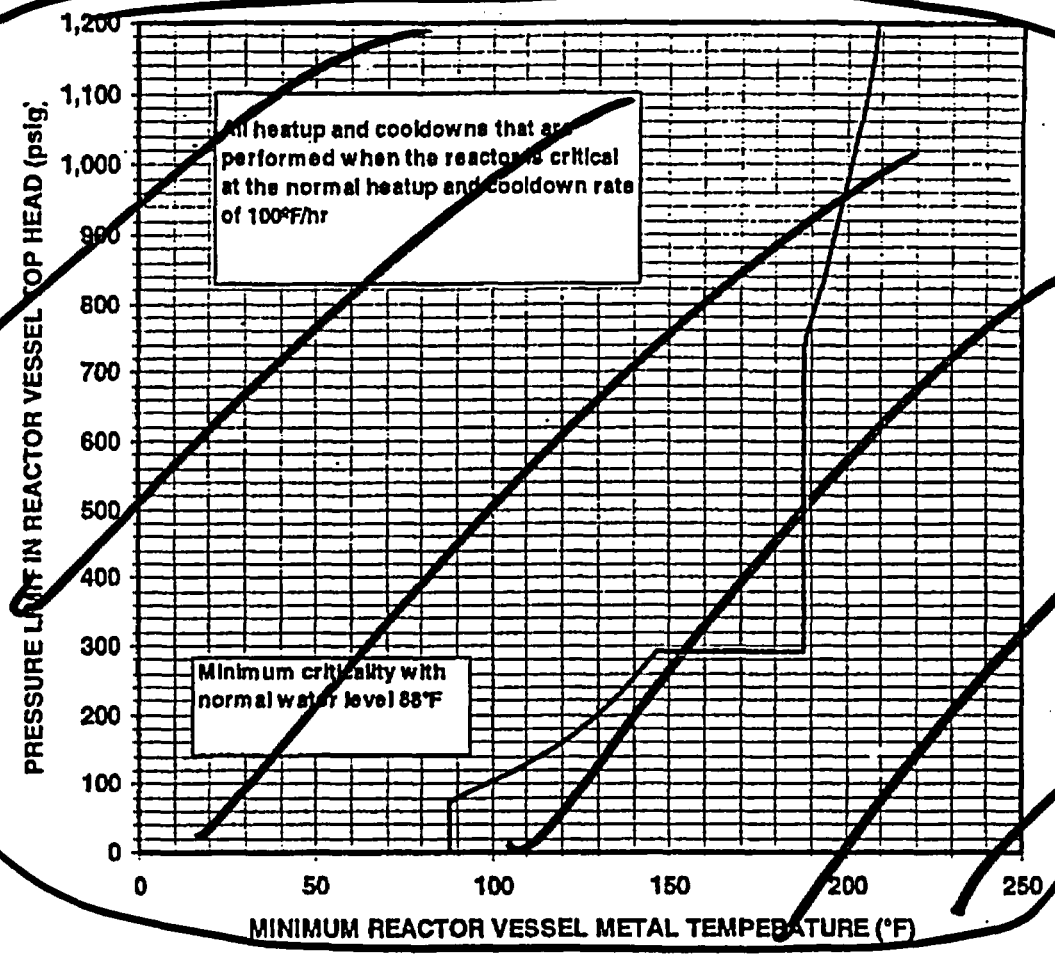
Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits - Curve B



Note: This figure is valid through Cycle 12 Operation with NRC Safety Evaluation Report supporting Amendment No. 139

INSERT CURVE B

Figure 3.4.6.1-3
Core Critical Heatup and Cooldown Pressure/Temperature Limits - Curve C



Note: This figure is valid through Cycle 12 Operation in accordance with NRC Safety Evaluation Report supporting Amendment No. 139.

INSERT CURVE C

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (3.9) of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Specifically the average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any 1-hour period.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G and ASME Code Cases N-588 and N-640. . The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.

INSERT

Tabulated values for the P-T curves are shown in Table B 3/4.4.6-2

TO BE CHANGED UPON IMPLEMENTATION OF BWROG VIP INTEGRATED SURVEILLANCE PROGRAM (LR-N02-0406)

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

ΔRT_{NDT} + MARgin

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni(%)	HIGHEST RT_{NDT}(°F)	ART_{NDT}(°F)	PREDICTED EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	0.71	+19	56 ~~53.8~~	~~67~~ 66	75 ~~72.8~~
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	~~.08~~ 0.081	~~0.59~~ 0.611	-30	~~63.1~~ 78	~~120~~ 118	~~33.1~~ 48

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

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, C1.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, C1.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, C1.1	All Heats	+30
LPCI Nozzles(1)	SA 508, C1.2,	All Heats	-20
Top Head Torus	SA 533, GR.B, C1.1	All Heats	+19
Top Head Flange	SA 508, C1.2	All Heats	+10
Vessel Flange	SA 508, C1.2	All Heats	+10
Feedwater Nozzle	SA 508, C1.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

(1) The design of the ~~Hope Creek~~ vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of ~~2.83~~ 3.3 x 10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of ~~+24.6°F~~ +29 °F.

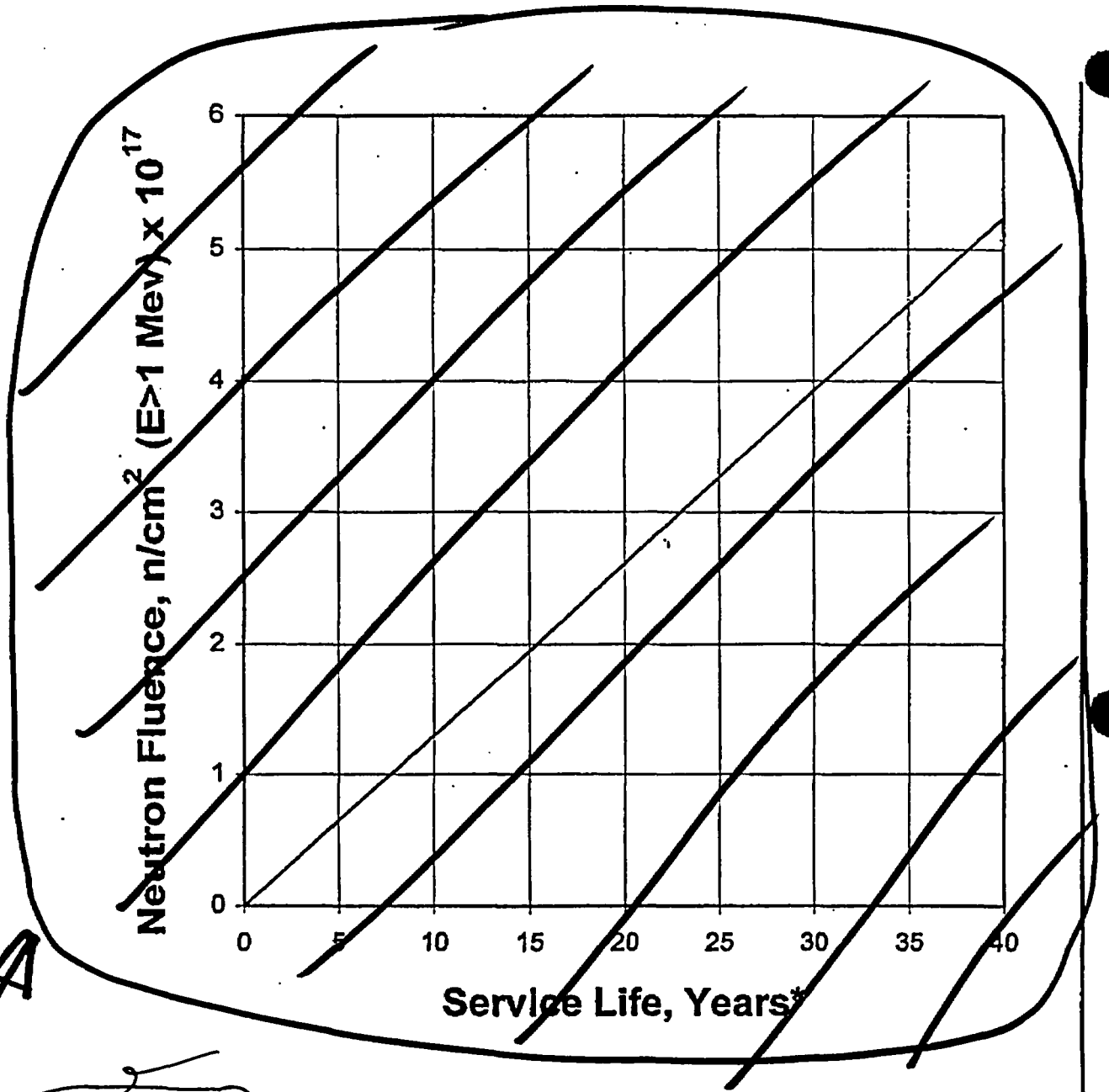


FIGURE B 3/4 4.6-1 LOWER - INTERMEDIATE SHELL FAST NEUTRON FLUENCE (E>1 MeV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

*At 80% capacity factor (40 years = 32 EFPY)

INSERT FIGURE B 3/4.4.6-1

INSERT TABLE B 3/4.4.6-2

BASES TABLE B 3/4.4.6-2

Numeric Values for Pressure/Temperature Limits

Figure 3.4.6.1-1, Curve A

Bottom Head	
Temperature (°F)	**Pressure (psig)**
79	0
79	929
88	1040
90	1068
92	1097
94	1126
96	1157
98	1190
100	1223

Upper Vessel	
Temperature (°F)	**Pressure (psig)**
79.0	0
79.0	292
118.0	292
118.0	925
123.0	996
128.0	1074
133.0	1161
138.0	1257

Beltline	
Temperature (°F)	**Pressure (psig)**
79.0	0
79.0	691
88.0	743
93.0	777
98.0	814
103.0	855
108.0	900
113.0	950
118.0	1,005
123.0	1,065
128.0	1,133
133.0	1,207

Figure 3.4.6.1-2, Curve B

Bottom Head	
Temperature (°F)	**Pressure (psig)**
79	0
79	606
88	690
92	732
96	778
100	827
104	881
108	939
112	1002
116	1070
120	1144
124	1224

Upper Vessel	
Temperature (°F)	**Pressure (psig)**
79.0	0
79.0	50
79.0	75
79.0	90
79.0	100
79.0	125
79.8	175
86.6	202
90.6	220
96.6	250
98.4	260
102.6	285
103.7	292
148.0	292
148.0	740
148.0	745
148.0	750
151.6	830
155.8	910
159.7	990
163.3	1070
165.5	1150
167.5	1230

Beltline	
Temperature (°F)	**Pressure (psig)**
79.0	0
79.0	416
88.0	455
93.0	480
98.0	508
103.0	538
108.0	572
113.0	610
118.0	651
123.0	697
128.0	747
133.0	803
138.0	864
143.0	932
148.0	1,008
153.0	1,091
158.0	1,183
163.0	1,284

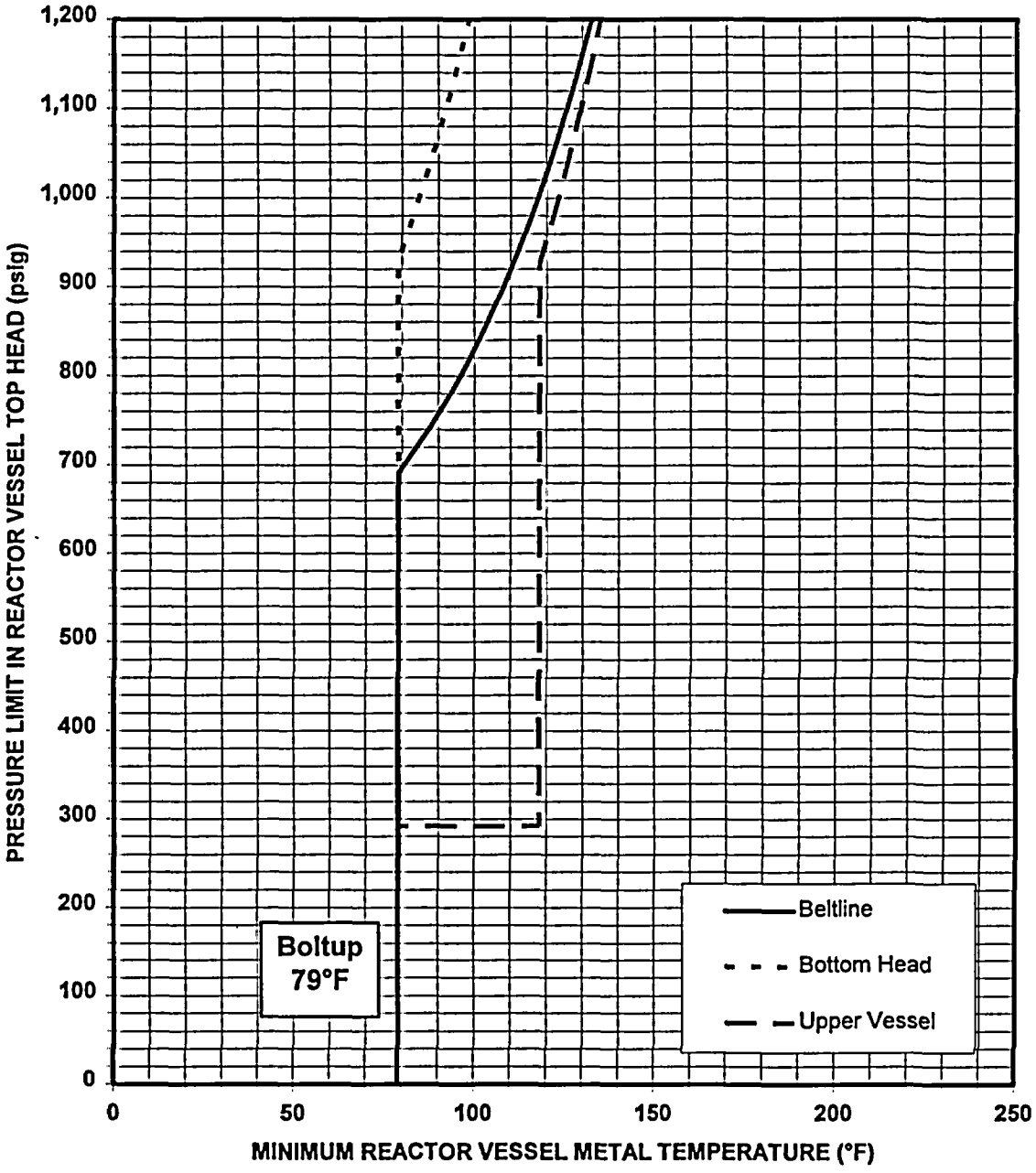
BASES TABLE B 3/4.4.6-2 (continued)

Numeric Values for Pressure/Temperature Limits

Figure 3.4.6.1-3, Curve C

Temperature (°F)	Pressure (psig)
88.0	0
88.0	50
88.0	75
88.0	90
92.0	100
103.4	125
119.8	175
126.6	202
130.6	220
136.6	250
138.4	260
142.6	285
143.7	292
188.0	292
188.0	740
188.0	745
188.0	750
191.6	830
195.8	910
199.7	990
203.3	1070
205.5	1150
207.5	1230

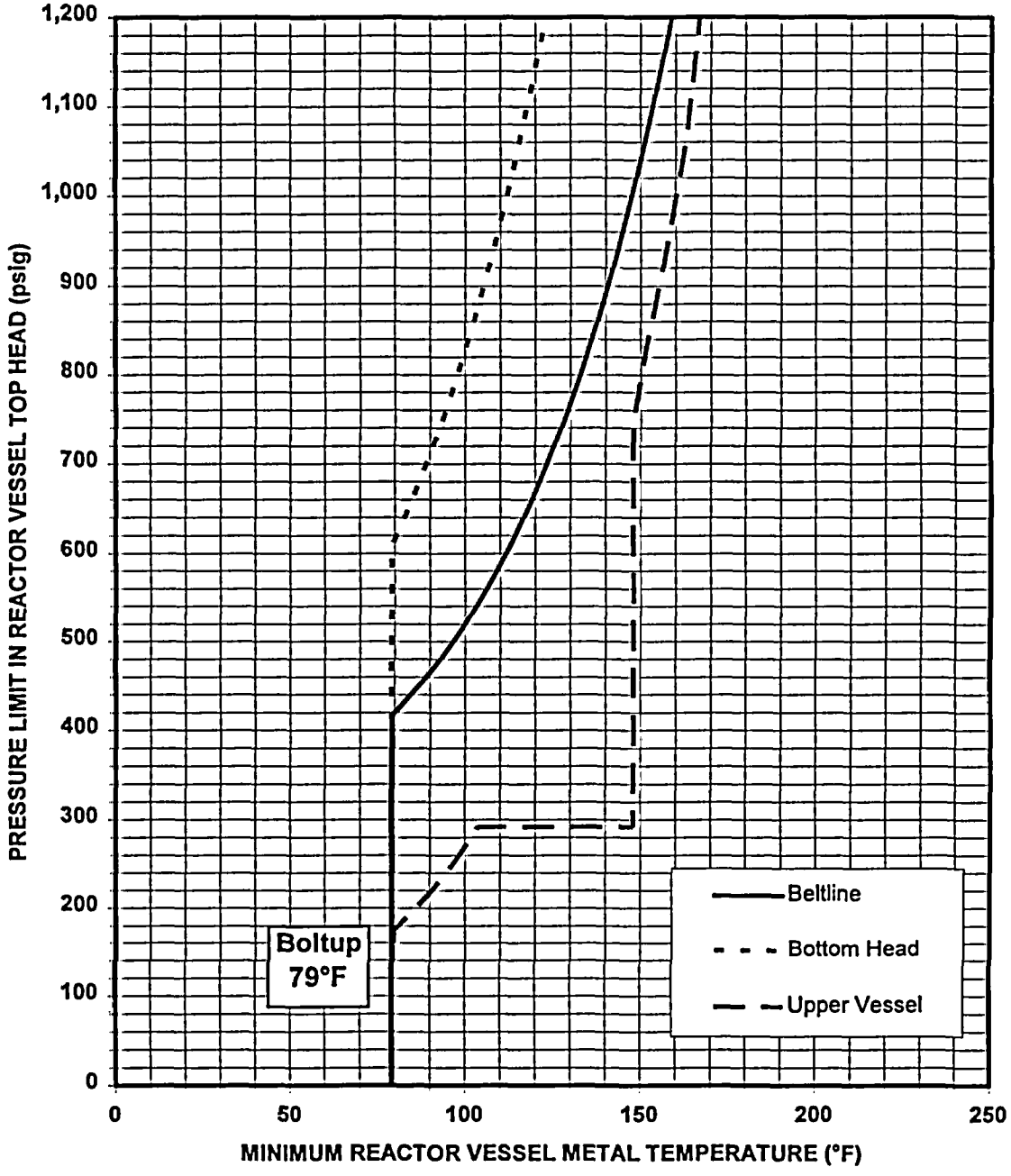
INSERT CURVE A



All system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with ASME Code Section XI.

This figure is valid for 32 EFPY of operation.

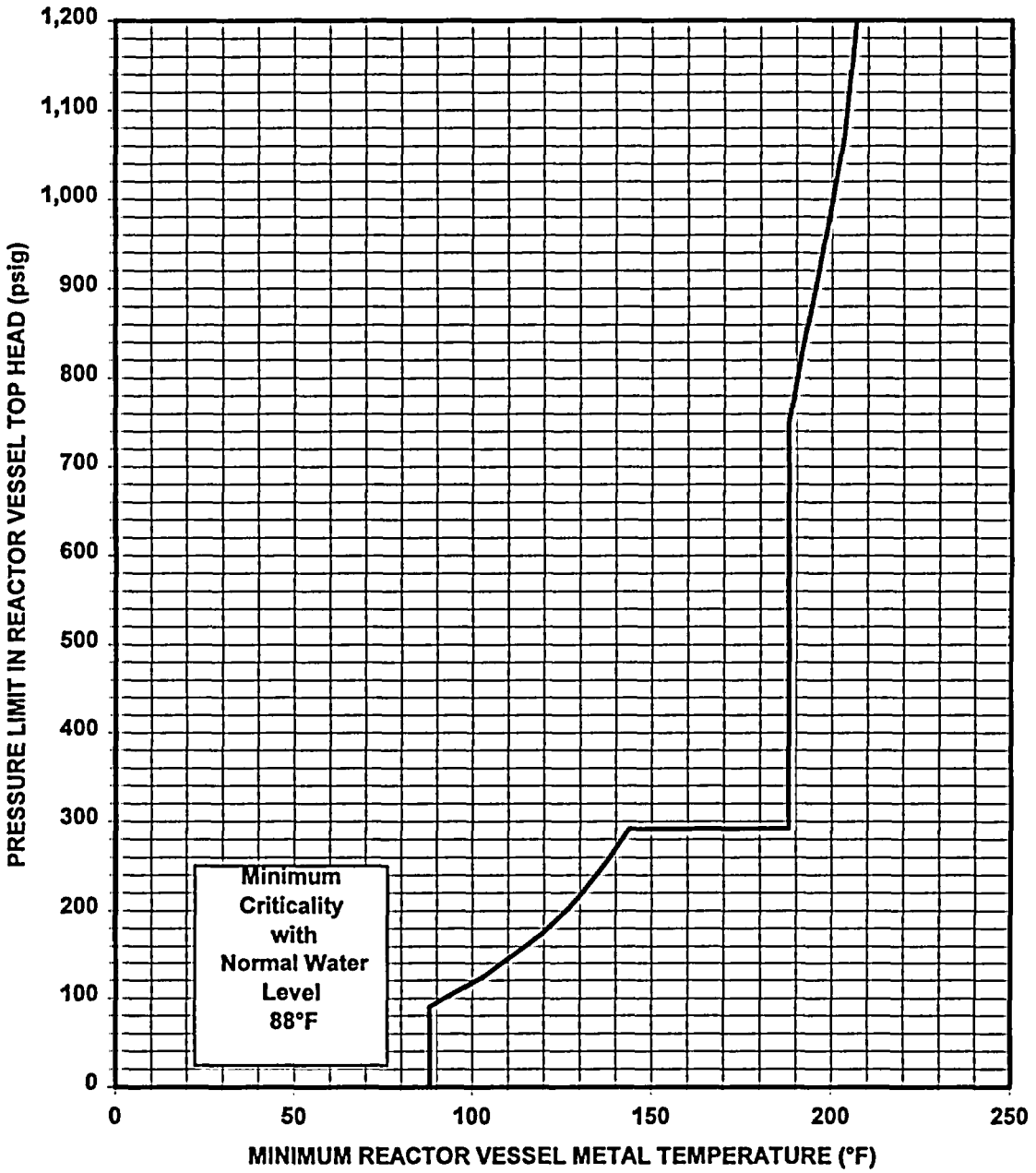
INSERT CURVE B



All heatup and cooldowns that are performed when the reactor is not critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

INSERT CURVE C



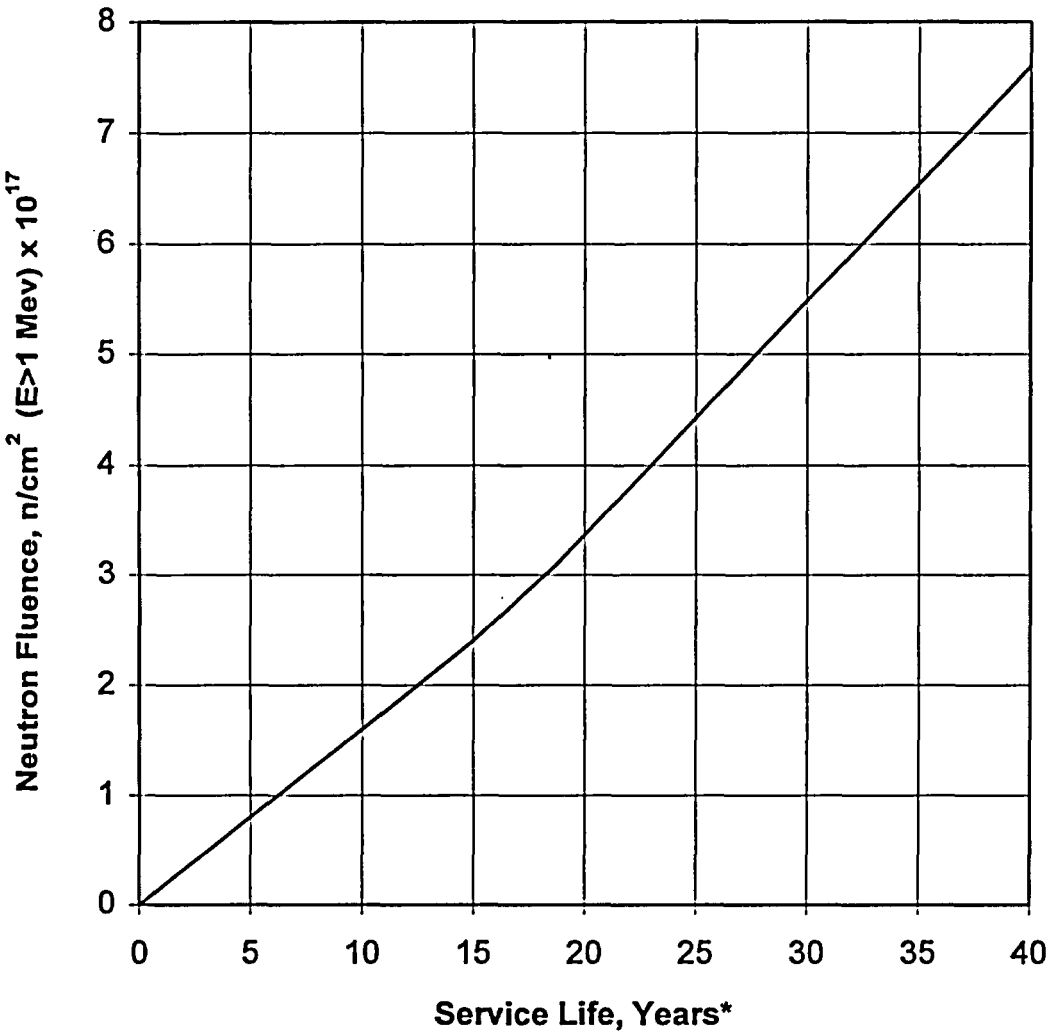
All heatup and cooldowns that are performed when the reactor is critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

INSERT TS BASES 3/4.4.6

The fluence in Bases Figure B 3/4.4.6-1 was determined using methodology described in NRC-approved General Electric Nuclear Energy Licensing Topical Report NEDC-32983P-A. This methodology is consistent with the guidance in Regulatory Guide 1.190, Rev. 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

INSERT FIGURE B 3/4.4.6-1



HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Revised Pressure-Temperature Curves for Hope Creek
SIR-00-136, Rev. 1

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March 23, 2004
BPT-04-005
SIR-00-136, Rev. 1

Mr. Randal J. Schmidt
PSEG Nuclear
Hope Creek Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038

Subject: Revised Pressure-Temperature (P-T) Curves for Hope Creek

Reference: PSEG Purchase Order No. 4500204466 dated 6/27/03

Dear Randy:

The attachment to this letter documents the revised set of pressure-temperature (P-T) curves developed for the Hope Creek Generating Station, in accordance with SI's Quality Assurance Program. This work was performed in accordance with the referenced contract, and includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for 32 and 48 effective full power years (EFPY). The curves were developed in accordance with 1989 ASME Code Section XI Appendix G, U.S. 10CFR 50 Appendix G, and ASME Code Cases N-588 and N-640.

The inputs, methodology, and results for this effort are summarized in the attachment. The calculations for this work (PSEG-10Q-301 and –302) are also attached.

Please don't hesitate to call me if you have any questions.

Prepared By: Bryan P. Templeton
Engineer

Reviewed By: Carl R. Limpus
Engineer

Approved By: Gary L. Stevens, P. E.
Associate

Attachments
cc: PSEG-10Q-401

ATTACHMENT

Revised P-T Curves for Hope Creek

1.0 Introduction

This attachment documents the revised set of pressure-temperature (P-T) curves developed for the Hope Creek Generating Station. This work includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for 32 and 48 effective full power years (EFPY). The curves were developed using the methodology specified in ASME Code Cases N-588 [2] and N-640 [3], as well as 1989 ASME Code Section XI Appendix G [4], 10CFR50 Appendix G [5], and WRC-175 [6]. The improvement realized from the Code Case methodology is as much as 60°F, and is primarily obtained from using the critical fracture toughness, K_{IC}, in accordance with Code Case N-640. This revision of the previously issued report incorporates revised RT_{NDT} values [17], effects of a potential Extended Power Uprate (EPU) of 20% increase [18], and an updated feedwater nozzle thermal stress and fatigue evaluation [16, 19] (which supersedes the work in Reference 20) . The new fluence estimates for projected EPU conditions [17] only affect beltline components which are exposed to high fluence, and therefore [1] is still the controlling document for the initial RT_{NDT} values for non-beltline components.

2.0 RT_{NDT} Values

Adjusted reference temperature (ART_{NDT}) values were developed for the Hope Creek reactor pressure vessel (RPV) materials in accordance with NRC Regulatory Guide 1.99, Revision 2 [13] based on the revised fluence data contained in Reference [17]. An EXCEL spreadsheet was set up to perform the RT_{NDT} calculations for the different EFPY levels required for this work, and is shown in Table 1 for 32 EFPY and Table 2 for 48 EFPY. The ART_{NDT} values calculated in Table 1 match the values from Reference [17], with only minor variation due to differences in the number of significant figures used in the calculations. To calculate the 48 EFPY ART_{NDT} values shown in Table 2, the fluence values from [17] were conservatively multiplied by a factor of 1.5 corresponding to a linear increase with time. The most limiting beltline material is the Intermediate Plate, Heat No. 5K3025/1.

Table 1: Hope Creek RPV Material ART_{NDT} 32 EFPY Calculations

Part Name & Material	Heat No.	Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	EFPY	Adjustments For 1/4t ΔRT_{NDT} (°F)	Margin Terms σ_A (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
(Lower Plates)	5K3230/1	-10	0.07	0.56	44	32.0	16	8.0	0.0	22
	6C35/1	-11	0.09	0.54	58	32.0	21	10.6	0.0	31
	6C45/1	1	0.08	0.57	51	32.0	19	9.3	0.0	38
(Lower Intermediate Plates)	5K2963/1	-10	0.07	0.58	44	32.0	16	8.0	0.0	22
	5K2530/1	19	0.08	0.56	51	32.0	19	9.3	0.0	56
	5K3238/1	7	0.09	0.64	58	32.0	21	10.6	0.0	49
(Intermediate Plates)	5K3025/1	19	0.15	0.71	113	32.0	28	13.9	0.0	75
	5K2608/1	19	0.09	0.58	58	32.0	14	7.2	0.0	48
	5K2698/1	19	0.10	0.58	65	32.0	16	8.0	0.0	51
(LPCI Nozzle)	19468/1	-20	0.12	0.80	86	32.0	20	9.9	0.0	20
	10024/1	-20	0.14	0.82	105	32.0	24	12.1	0.0	28
Vertical Welds 3:										
Shell 3: SMAW / W13	510-01205	-40	0.09	0.54	109	32.0	27	13.5	0.0	14
Shell 3: SAW / W13	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	26	13.2	0.0	23
Grith Welds 3/4:										
Shells 3/4: SMAW / W6	519-01205	-49	0.01	0.53	20	32.0	5	2.5	0.0	-39
Shells 3/4: SMAW / W6	504-01205	-31	0.01	0.51	20	32.0	5	2.5	0.0	-21
Shells 3/4: SMAW / W6	510-01205	-40	0.09	0.54	109	32.0	27	13.5	0.0	14
Shells 3/4: SAW / W6	D53040 / 1810-02205	-49	0.081	0.611	107	32.0	26	13.2	0.0	4
Shells 3/4: SAW / W6	D55733 / 1810-02205	-40	0.10	0.68	126	32.0	31	15.6	0.0	22
LPCI Nozzle Welds:										
SMAW / W179	001-01205	-40	0.02	0.51	27	32.0	6	3.1	0.0	-28
SMAW / W179	519-01205	-49	0.01	0.53	20	32.0	5	2.3	0.0	-40
SMAW / W179	504-01205	-31	0.01	0.51	20	32.0	5	2.3	0.0	-22
Vertical Welds 4&5:										
Shells 4&5: SMAW / W14&15	510-01205	-40	0.09	0.54	109	32.0	40	19.9	0.0	40
Shells 4&5: SMAW / W14&15	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	39	19.5	0.0	48
Girth Welds 4/5:										
Shells 4/5: SMAW / W7	510-01205	-40	0.09	0.54	109	32.0	40	19.9	0.0	40
Shells 4/5: SAW / W7	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	39	19.5	0.0	48

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation @ 1/4t $e^{-0.34t}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
(Lower Plates)	6.100	1.525	32.0	1.10E+18	0.694	7.63E+17	0.365
(Lower Intermediate Plates)			32.0	1.10E+18	0.694	7.63E+17	0.365
(Intermediate Plates)			32.0	5.30E+17	0.694	3.68E+17	0.247
(LPCI Nozzle)			32.0	4.70E+17	0.694	3.26E+17	0.230

Table 2: Hope Creek RPV Material ART_{NDT} 48 EFPY Calculations

Part Name & Material	Heat No.	Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	EFPY	Adjustments For 1/4t ΔRT_{NDT} (°F)	Margin Terms σ_A (°F)	Margin Terms σ_I (°F)	ART_{NDT} (°F)
(Lower Plates)	5K3230/1	-10	0.07	0.56	44	32.0	20	9.8	0.0	29
	6C35/1	-11	0.09	0.54	58	32.0	26	12.9	0.0	41
	6C45/1	1	0.08	0.57	51	32.0	23	11.3	0.0	46
(Lower Intermediate Plates)	5K2963/1	-10	0.07	0.58	44	32.0	20	9.8	0.0	29
	5K2530/1	19	0.08	0.56	51	32.0	23	11.3	0.0	64
	5K3238/1	7	0.09	0.64	58	32.0	26	12.9	0.0	59
(Intermediate Plates)	5K3025/1	19	0.15	0.71	113	32.0	35	17.4	0.0	89
	5K2608/1	19	0.09	0.58	58	32.0	18	8.9	0.0	55
	5K2698/1	19	0.10	0.58	65	32.0	20	10.0	0.0	59
(LPCI Nozzle)	19468/1	-20	0.12	0.80	86	32.0	25	12.4	0.0	30
	10024/1	-20	0.14	0.82	105	32.0	30	15.2	0.0	41
Vertical Welds 3:										
Shell 3: SMAW / W13	510-01205	-40	0.09	0.54	109	32.0	34	16.8	0.0	27
Shell 3: SAW / W13	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	33	16.5	0.0	36
Grith Welds 3/4:										
Shells 3/4: SMAW / W6	519-01205	-49	0.01	0.53	20	32.0	6	3.1	0.0	-37
Shells 3/4: SMAW / W6	504-01205	-31	0.01	0.51	20	32.0	6	3.1	0.0	-19
Shells 3/4: SMAW / W6	510-01205	-40	0.09	0.54	109	32.0	34	16.8	0.0	27
Shells 3/4: SAW / W6	D53040 / 1810-02205	-49	0.081	0.611	107	32.0	33	16.5	0.0	17
Shells 3/4: SAW / W6	D55733 / 1810-02205	-40	0.10	0.68	126	32.0	39	19.4	0.0	38
LPCI Nozzle Welds:										
SMAW / W179	001-01205	-40	0.02	0.51	27	32.0	8	3.9	0.0	-24
SMAW / W179	519-01205	-49	0.01	0.53	20	32.0	6	2.9	0.0	-37
SMAW / W179	504-01205	-31	0.01	0.51	20	32.0	6	2.9	0.0	-19
Vertical Welds 4&5:										
Shells 4&5: SMAW / W14&15	510-01205	-40	0.09	0.54	109	32.0	48	20.0	0.0	48
Shells 4&5: SMAW / W14&15	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	48	20.0	0.0	58
Girth Welds 4/5:										
Shells 4/5: SMAW / W7	510-01205	-40	0.09	0.54	109	32.0	48	20.0	0.0	48
Shells 4/5: SAW / W7	D53040 / 1125-02205	-30	0.081	0.611	107	32.0	48	20.0	0.0	58

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation @ 1/4t $e^{-0.34t}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
(Lower Plates)	6.100	1.525	32.0	1.65E+18	0.694	1.14E+18	0.444
(Lower Intermediate Plates)			32.0	1.65E+18	0.694	1.14E+18	0.444
(Intermediate Plates)			32.0	7.95E+17	0.694	5.51E+17	0.308
(LPCI Nozzle)			32.0	7.05E+17	0.694	4.89E+17	0.289

3.0 P-T Curve Methodology

The P-T curve methodology is based on the requirements of References [2] through [6]. The supporting calculations for the curves are contained in References [7] and [8]. There are three regions of the reactor pressure vessel (RPV) that are evaluated: (1) the beltline region, (2) the

bottom head region, and (3) the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

The approach used for the beltline and bottom head (all curves), and upper vessel (Curve A only) includes the following steps:

a. Assume a fluid temperature, T. The temperature of the metal at the assumed flaw tip, $T_{1/4t}$ (i.e., 1/4t into the vessel wall) is conservatively assumed equal to fluid temperature. The assumed temperature also must account for an instrument uncertainty of 9°F [14].

b. Calculate the allowable stress intensity factor, K_{IC}, based on $T_{1/4t}$ using the relationship from Code Case N-640 [3], as follows:

$$K_{IC} = 20.734\, e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2 \qquad \text{(eqn. from Ref. [9])}$$

where: $T_{1/4t}$ = metal temperature at assumed flaw tip (°F)
ART_{NDT} = adjusted reference temperature for location under consideration and desired EFPY (°F)
K_{IC} = allowable stress intensity factor (ksi√inch)

c. Calculate the thermal stress intensity factor, K_{IT} from Code Case N-588 [2] for the beltline and bottom head regions, or from finite element results for the feedwater nozzle/upper vessel region.

d. Calculate the allowable pressure stress intensity factor, K_{IP}, using the following relationship:

$$K_{IP} = (K_{IC} - K_{IT})/SF$$

where: K_{IP} = allowable pressure stress intensity factor (ksi√inch)
SF = safety factor
= 1.5 for pressure test conditions (Curve A)
= 2.0 for heatup/cooldown conditions (Curves B and C)

e. Compute the allowable pressure, P, from the allowable pressure stress intensity factor, K_{IP}.

f. Subtract any applicable adjustments for pressure from P. The beltline and bottom head include a pressure adjustment of 20.3 psig to account for the static pressure head of a full vessel. An instrument error of 20.5 psig was also assumed [15].

g. Repeat steps (a) through (f) for other temperatures to generate a series of P-T points.

The approach used for the upper vessel (Curves B & C) includes the following steps:

a. Assume a fluid pressure, P. The assumed pressure includes an instrument uncertainty of 20.5 psig [15].

b. Calculate the thermal stress intensity factor, K_{IT}, based on finite element stresses. The feedwater nozzle stresses were obtained from the finite element analysis results contained in Reference [16]. The highest linearized (membrane and membrane + bending) thermal stresses for all of the design basis transients were selected to encompass all expected operating conditions.

σ_{ys} = 43.975 ksi @ 575°F for SA-508 Cl. 2 [11, 10]

Calculate $t^{1/2}$. The resulting M_m value is obtained from G-2214.1 [2].

K_{Im} is calculated from the equation in Paragraph G-2214.1 [4]:

$$K_{Im} = M_m * \sigma_{sm}$$

K_{Ib} is calculated from the equation in Paragraph G-2214.2 [4]:

$$K_{Ib} = (2/3)\, M_m * \sigma_{sb}$$

The total K_{IT} is therefore:

$$K_{IT} = R*SF*(K_{Im} + K_{Ib})$$

where:	R	=	correction factor, calculated to consider the nonlinear effects in the plastic region based on the assumptions and recommendations of WRC Bulletin 175 [6].
		=	$[\sigma_{ys} - \sigma_{pm} + ((\sigma_{total} - \sigma_{ys}) / 30)] / (\sigma_{total} - \sigma_{pm})$
	SF	=	Safety Factor for K_{IT}
		=	1.3 (conservatively used based on the recommendation in WRC-175 [6])

c. Compute the allowable pressure stress intensity factor, K_{IP}, is as follows:

$$K_{IP} = F(a/r_n)\sqrt{\pi a}\ \sigma_{pm}$$

where:	r_i	=	actual inner radius of nozzle
	r_c	=	nozzle corner radius [7]
	r_n	=	apparent radius of nozzle = $r_i + 0.29r_c$
	t'	=	nozzle corner thickness
	a	=	crack depth (inches)
		=	1/4 t'
	$F(a,r_n)$	=	nozzle stress factor, from Figure A5-1 of [6]

K_{IP} = allowable pressure stress intensity factor (ksi√inch).

σ_{pm} = primary membrane stress, PR/t (primary bending stresses are conservatively treated as membrane stresses, so $\sigma_{pb} = 0$)

d. Calculate the allowable stress intensity factor, K_{IC}, using the following relationship for a heatup/cooldown P-T curve:

$$K_{IP} = \frac{K_{IC} - K_{IT}}{2.0}$$

$$\text{thus:} \quad K_{IC} = 2.0K_{IP} + K_{IT}$$

e. Calculate the temperature, T, using the relationship from Code Case N-640 [3], as follows:

$$K_{IC} = 20.734\, e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2 \quad \text{(eqn. from Ref. [9])}$$

where: $T_{1/4t}$ = metal temperature at assumed flaw tip (°F), assumed equal to T, the temperature at the inner vessel wall

ART_{NDT} = adjusted reference temperature for location under consideration and desired EFPY (°F)

K_{IC} = allowable stress intensity factor (ksi√inch)

thus:

$$T = 50 * LN\left[\frac{K_{IC} - 33.2}{20.734}\right] + ART_{NDT}$$

f. The curve was generated by scaling the stresses used to determine the pressure and thermal stress intensity factors. The primary stresses were scaled based on pressure, while the secondary stresses were scaled based on temperature difference.

g. Repeat steps (a) through (f) for other pressures to generate a series of P-T points.

The following additional requirements were used to define the P-T curves. These limits are established in Reference [5]:

For Pressure Test Conditions (Curve A):

- If the pressure is greater than 20% of the pre-service hydro test pressure (312.6 psig), the temperature must be greater than ART_{NDT} of the limiting flange material + 90°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature is must be greater than or equal to the ART_{NDT} of the limiting flange material + 60°F. The instrument uncertainty of 9°F was not

applied since the 60°F is an additional margin above that recommended in Reference [10], and has been a standard recommendation for the BWR industry for non-ductile failure protection. Therefore, the 60°F is considered to adequately encompass instrument uncertainty.

For Core Not Critical Conditions (Curve B):

- If the pressure is greater than 20% of the pre-service hydro test pressure (312.6 psig), the temperature must be greater than RT_{NDT} of the limiting flange material + 120°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature must be greater than or equal to the ART_{NDT} of the limiting flange material + 60°F. The instrument uncertainty of 9°F was not applied since the 60°F is an additional margin above that recommended in Reference [10], and has been a standard recommendation for the BWR industry for non-ductile failure protection. Therefore, the 60°F is considered to adequately encompass instrument uncertainty.

For Core Critical Conditions (Curve C):

- Per the requirements of Table 1 of Reference [5], the core critical P-T limits must be 40°F above any Pressure Test or Core Not Critical curve limits. Core Not Critical conditions are more limiting than Pressure Test conditions, so Core Critical conditions are equal to Core Not Critical conditions plus 40°F.
- Another requirement of Table 1 of Reference [5] (or actually an allowance for the BWR), concerns minimum temperature for initial criticality in a startup. Given that water level is normal, BWRs are allowed initial criticality at the closure flange region temperature (ART_{NDT} + 60°F) if the pressure is below 20% of the pre-service hydro test pressure.
- Also per Table 1 of Reference [5], at pressures above 20% of the pre-service hydro test pressure, the Core Critical curve temperature must be at least that required for the pressure test (Pressure Test Curve at 1,100 psig). As a result of this requirement, the Core Critical curve must have a step at a pressure equal to 20% of the pre-service hydro pressure to the temperature required by the Pressure Test curve at 1,100 psig, or Curve B + 40°F, whichever is greater.

After accounting for instrument uncertainties, the resulting pressure and temperature series constitutes the P-T curve. The P-T curve relates the minimum required fluid temperature to the reactor pressure.

4.0 P-T Curves

Tabulated values for the P-T curves are shown in Tables 3 through 11. The resulting P-T curves are shown in Figures 1 through 5. Note that since the upper vessel (non-beltline) curve is limiting for core not critical conditions for both 32 and 48 EFPY, Curve C is identical for both EFPY levels (i.e., no fluence effects).

6.0 References

1. GE-NE-523-A164-1294R1, "Hope Creek 1 Generating Station RPV Surveillance Materials Testing and Fracture Toughness Analysis," December 1997, SI File No. PSEG-10Q-201 (PSEG VTD 323326 Rev. 1, including outstanding change 80010289, CD M-548).

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

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

4. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1989 Edition.

5. U. S. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements," 1-1-98 Edition.

6. WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," PVRC Ad Hoc Group on Toughness Requirements, Welding Research Council, August 1972.

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12. Deleted.

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15. Pressure Loop Uncertainty, PSEG Nuclear Calculation SC-BB-0355, SI File No. PSEG-10Q-103 (11/1/00 e-mail).

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17. GE-NE-000-0005-3385-01R0,"Project Task Report PSEG Nuclear LLC Hope Creek Generating Station Extended Power Uprate Task T0301: RPV Fracture Toughness Evaluation," July 2003, SI File No. PSEG-10Q-211P (PSEG VTD 326202).

18. GE-NE-26A5958R2, "Reactor Vessel Extended Power Uprate Certified Design Specification," November 25, 2004, SI File No. PSEG-10Q-209P (PSEG VTD 326205).

19. Structural Integrity Associates Calculation No. HC-05Q-304, Revision 0, "Feedwater Nozzle Thermal and Stress Analysis," 3/22/04.

20. GE-NE-000-0006-0156-01R0, "Project Task Report PSEG Nuclear LLC Hope Creek Generating Station Extended Power Uprate Task T0302: Reactor Vessel Integrity – Stress Evaluation," November 2003, SI File No. PSEG-10Q-210P (PSEG VTD 326203).

Table 3
Tabulated Values for Beltline Pressure Test Curve (Curve A) for 32 EFPY

Pressure-Temperature Curve Calculation
(Pressure Test = Curve A)

Inputs:

Plant =	Hope Creek		
Component =	Beltline		
Vessel thickness, t =	6.1000	inches, so √t =	2.470 √inch
Vessel Radius, R =	126.5	inches	
ART_{NDT} =	75	°F ======>	32 EFPY
Cooldown Rate, CR =	0	°F/hr	
K_{IT} =	0.00	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)	
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)	
Safety Factor =	1.50	(for pressure test)	
M_m =	2.287	(From N-588, for inside surface axial flaw)	
Temperature Adjustment =	9.0	°F	
Height of Water for a Full Vessel =	562.5	inches	
Pressure Adjustment =	20.3	psig (hydrostatic pressure for a full vessel at 70°F)	
Pressure Adjustment =	20.5	psig (Instrument Uncertainty)	
Conversion factor from ksi to psi =	1000		
Hydro Test Pressure =	1,563	psig	
Flange RT_{NDT} =	19.0	°F	

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	55.80	37.20	0	79.0	0
79.0	70.0	52.07	34.72	732	79.0	691
88.0	79.0	55.80	37.20	784	88.0	743
93.0	84.0	58.17	38.78	818	93.0	777
98.0	89.0	60.80	40.53	855	98.0	814
103.0	94.0	63.70	42.47	895	103.0	855
108.0	99.0	66.91	44.61	941	108.0	900
113.0	104.0	70.45	46.97	990	113.0	950
118.0	109.0	74.37	49.58	1045	118.0	1,005
123.0	114.0	78.70	52.47	1106	123.0	1,065
128.0	119.0	83.49	55.66	1174	128.0	1,133
133.0	124.0	88.78	59.18	1248	133.0	1,207
138.0	129.0	94.62	63.08	1330	138.0	1,289
143.0	134.0	101.08	67.39	1421	143.0	1,380
148.0	139.0	108.22	72.15	1521	148.0	1,480
153.0	144.0	116.11	77.41	1632	153.0	1,591
158.0	149.0	124.83	83.22	1755	158.0	1,714
163.0	154.0	134.47	89.65	1890	163.0	1,849
168.0	159.0	145.12	96.75	2040	168.0	1,999

Table 4
Tabulated Values for Beltline Pressure Test Curve (Curve A) for 48 EFPY

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	Hope Creek		
Component =	Beltline		
Vessel thickness, t =	6.1000	inches, so √t =	2.470 √inch
Vessel Radius, R =	126.5	inches	
ART_{NDT} =	89	°F ======>	48 EFPY
Cooldown Rate, CR =	0	°F/hr	
K_{IT} =	0.00	ksi*inch$^{1/2}$ (From N-588, for cooldown rate above)	
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)	
Safety Factor =	1.50	(for pressure test)	
M_m =	2.287	(From N-588, for inside surface axial flaw)	
Temperature Adjustment =	9.0	°F	
Height of Water for a Full Vessel =	562.5	inches	
Pressure Adjustment =	20.3	psig (hydrostatic pressure for a full vessel at 70°F)	
Pressure Adjustment =	20.5	psig (Instrument Uncertainty)	
Conversion factor from ksi to psi =	1000		
Hydro Test Pressure =	1,563	psig	
Flange RT_{NDT} =	19.0	°F	

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	50.31	33.54	0	79.0	0
79.0	70.0	47.49	31.66	668	79.0	627
88.0	79.0	50.31	33.54	707	88.0	666
93.0	84.0	52.11	34.74	732	93.0	692
98.0	89.0	54.10	36.07	760	98.0	720
103.0	94.0	56.30	37.53	791	103.0	751
108.0	99.0	58.73	39.15	826	108.0	785
113.0	104.0	61.41	40.94	863	113.0	822
118.0	109.0	64.38	42.92	905	118.0	864
123.0	114.0	67.66	45.11	951	123.0	910
128.0	119.0	71.28	47.52	1002	128.0	961
133.0	124.0	75.29	50.19	1058	133.0	1,017
138.0	129.0	79.72	53.14	1121	138.0	1,080
143.0	134.0	84.61	56.40	1189	143.0	1,148
148.0	139.0	90.01	60.01	1265	148.0	1,224
153.0	144.0	95.99	63.99	1349	153.0	1,308
158.0	149.0	102.59	68.39	1442	158.0	1,401
163.0	154.0	109.89	73.26	1545	163.0	1,504
168.0	159.0	117.96	78.64	1658	168.0	1,617
173.0	164.0	126.87	84.58	1783	173.0	1,743
178.0	169.0	136.72	91.15	1922	178.0	1,881
183.0	174.0	147.61	98.41	2075	183.0	2,034

Table 5
Tabulated Values for Feedwater Nozzle/Upper Vessel Region Pressure Test Curve (Curve A)

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	Hope Creek			
Component =	Upper Vessel	(based on FW nozzle)		
ART_{NDT} =	40.0	°F ======>	All EFPYs	
Vessel thickness, t =	6.169	inches, so √t =	2.484	√inch
Vessel Radius, R =	126.5	inches		
Nozzle corner thickness, t' =	9.7	inches, approximate		
F(a/m) =	1.44	nozzle stress factor		
Crack Depth, a =	2.425	inches		
Safety Factor =	1.50			
Temperature Adjustment =	9.0	°F		
Pressure Adjustment =	20.5	psig (Instrument Uncertainty)		
Conversion factor from ksi to psi =	1000			
Unit Pressure =	1,563	psig		
Flange RT_{NDT} =	19.0	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	78.43	52.29	0	79.0	0
79.0	79.0	78.43	52.29	313	79.0	292
118.0	109.0	115.62	77.08	313	118.0	292
118.0	109.0	115.62	77.08	946	118.0	925
123.0	114.0	124.28	82.86	1017	123.0	996
128.0	119.0	133.86	89.24	1095	128.0	1074
133.0	124.0	144.45	96.30	1182	133.0	1161
138.0	129.0	156.15	104.10	1277	138.0	1257
143.0	134.0	169.08	112.72	1383	143.0	1363
148.0	139.0	183.37	122.25	1500	148.0	1479

Table 6
Tabulated Values for Bottom Head Pressure Test Curve (Curve A)

Pressure-Temperature Curve Calculation

(Pressure Test = Curve A)

Inputs:

Plant =	**Hope Creek**	
Component =	**Bottom Head**	(Penetrations Portion)
Vessel thickness, t =	**6.100**	inches, so √t = 2.470 √inch
Vessel Radius, R =	**126.5**	inches
ART_{NDT} =	**30.0**	°F ======> **All EFPYs**
Safety Factor =	**1.50**	
Safety Factor =	**2.30**	Bottom Head Penetrations
M_m =	**2.287**	(From N-588, for inside surface axial flaw)
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)
Height of Water for a Full Vessel =	**562.5**	inches
Pressure Adjustment =	**20.3**	psig (full vessel at 70°F)
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)
Conversion factor from ksi to psi =	**1000**	
Unit Pressure =	**1,563**	psig
Flange RT_{NDT} =	**19.0**	°F

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	88.44	58.96	0	79	0
79.0	70.0	79.34	52.90	970	79	929
88.0	79.0	88.44	58.96	1081	88	1040
90.0	81.0	90.70	60.47	1109	90	1068
92.0	83.0	93.05	62.03	1137	92	1097
94.0	85.0	95.49	63.66	1167	94	1126
96.0	87.0	98.03	65.35	1198	96	1157
98.0	89.0	100.68	67.12	1231	98	1190
100.0	91.0	103.43	68.95	1264	100	1223
102.0	93.0	106.30	70.86	1299	102	1258
104.0	95.0	109.28	72.85	1336	104	1295
106.0	97.0	112.38	74.92	1374	106	1333
108.0	99.0	115.62	77.08	1413	108	1372
110.0	101.0	118.98	79.32	1454	110	1413
112.0	103.0	122.48	81.65	1497	112	1456
114.0	105.0	126.12	84.08	1542	114	1501
116.0	107.0	129.92	86.61	1588	116	1547
118.0	109.0	133.86	89.24	1636	118	1595

Table 7
Tabulated Values for Beltline Core Not Critical Curve (Curve B) for 32 EFPY

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**		
Component =	**Beltline**		
Vessel thickness, t =	**6.1000**	inches, so √t =	2.470 √inch
Vessel Radius, R =	**126.5**	inches	
ART_{NDT} =	**75**	°F ======>	**32 EFPY**
Cooldown Rate, CR =	**100**	°F/hr	
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)	
M_T =	**0.285**	(From Figure G-2214-2)	
$\Delta T_{1/4t}$ =	**0.0**	°F = Conservatively assumed zero	
Safety Factor =	**2.00**		
M_m =	**2.287**	(From N-588, for inside surface axial flaw)	
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)	
Height of Water for a Full Vessel =	**562.5**	inches	
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)	
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)	
Conversion factor from ksi to psi =	**1000**		
Hydro Test Pressure =	**1,563**	psig	
Flange RT_{NDT} =	**19.0**	°F	

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	55.80	23.52	0	79.0	0
79.0	70.0	52.07	21.66	457	79.0	416
88.0	79.0	55.80	23.52	496	88.0	455
93.0	84.0	58.17	24.71	521	93.0	480
98.0	89.0	60.80	26.02	549	98.0	508
103.0	94.0	63.70	27.47	579	103.0	538
108.0	99.0	66.91	29.08	613	108.0	572
113.0	104.0	70.45	30.85	650	113.0	610
118.0	109.0	74.37	32.81	692	118.0	651
123.0	114.0	78.70	34.97	737	123.0	697
128.0	119.0	83.49	37.37	788	128.0	747
133.0	124.0	88.78	40.01	844	133.0	803
138.0	129.0	94.62	42.93	905	138.0	864
143.0	134.0	101.08	46.16	973	143.0	932
148.0	139.0	108.22	49.73	1049	148.0	1,008
153.0	144.0	116.11	53.68	1132	153.0	1,091
158.0	149.0	124.83	58.04	1224	158.0	1,183
163.0	154.0	134.47	62.86	1325	163.0	1,284

Table 8
Tabulated Values for Beltline Core Not Critical Curve (Curve B) for 48 EFPY

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1000**	inches, so √t =	2.470	√inch
Vessel Radius, R =	**126.5**	inches		
ART_{NDT} =	**89**	°F ======>	**48 EFPY**	
Cooldown Rate, CR =	**100**	°F/hr		
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)		
M_T =	**0.285**	(From Figure G-2214-2)		
$\Delta T_{1/4t}$ =	**0.0**	°F = Conservatively assumed zero		
Safety Factor =	**2.00**			
M_m =	**2.287**	(From N-588, for inside surface axial flaw)		
Temperature Adjustment =	**9.0**	°F		
Height of Water for a Full Vessel =	**562.5**	inches		
Pressure Adjustment =	**20.3**	psig (hydrostatic pressure for a full vessel at 70°F)		
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)		
Conversion factor from ksi to psi =	**1000**			
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**19.0**	°F		

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	50.31	20.78	0	79.0	0
79.0	70.0	47.49	19.37	408	79.0	368
88.0	79.0	50.31	20.78	438	88.0	397
96.0	87.0	53.28	22.26	469	96.0	429
104.0	95.0	56.77	24.00	506	104.0	465
112.0	103.0	60.85	26.05	549	112.0	508
120.0	111.0	65.65	28.45	600	120.0	559
128.0	119.0	71.28	31.26	659	128.0	618
136.0	127.0	77.89	34.57	729	136.0	688
144.0	135.0	85.65	38.44	811	144.0	770
152.0	143.0	94.75	42.99	906	152.0	866
160.0	151.0	105.42	48.33	1019	160.0	978
168.0	159.0	117.96	54.60	1151	168.0	1,110
176.0	167.0	132.66	61.95	1306	176.0	1,265
184.0	175.0	149.92	70.58	1488	184.0	1,447
192.0	183.0	170.17	80.71	1702	192.0	1,661

Table 9
Tabulated Values for Feedwater Nozzle/Upper Vessel Region Core Not Critical Curve (Curve B)

Pressure-Temperature Curve Calculation
(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**	
Component =	**Upper Vessel**	
ART_{NDT} =	**40.0**	°F
σ_{pm} =	**20.61**	ksi for a pressure of 1,005 psig
σ_{pb} =	**0.00**	ksi for a pressure of 1,005 psig
$\sigma_{sm\ (original)}$ =	**4.68**	ksi for a temperature of 547°F [16]
$\sigma_{sb\ (original)}$ =	**16.29**	ksi for a temperature of 547°F [16]
σ_{ys} =	**44.0**	ksi
M_m =	**2.88**	
$F(a/r_n)$ =	**1.44**	
Temperature Adjustment =	**9.0**	°F (Instrument Uncertainty)
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)
Hydro Test Pressure =	**1563**	psig
Flange RT_{NDT} =	**19.0**	°F

Adjusted Pressure for Calculation (psig)	Saturation Temperature (°F)	K_{IT} (ksi*inch$^{1/2}$)	K_{IP} ksi*inch$^{1/2}$	K_{IC} (ksi*inch$^{1/2}$)	Calculated Temperature (°F)	Adjusted Temperature for P-T Curve (°F)	Pressure for P-T Curve (psig)
0	212.1	19.7	0.0	19.7	—	79.0	0
70.5	316.1	31.7	5.7	43.2	3.4	79.0	50
95.5	334.7	33.8	7.8	49.4	27.6	79.0	75
110.5	344.3	34.9	9.0	52.9	37.5	79.0	90
120.5	350.2	35.6	9.8	55.2	43.0	79.0	100
145.5	363.6	37.1	11.9	60.8	54.4	79.0	125
195.5	386.1	39.7	15.9	71.6	70.8	79.8	175
222.5	396.5	40.9	18.1	77.2	77.6	86.6	202
240.5	403.0	41.6	19.6	80.9	81.6	90.6	220
270.5	413.0	42.8	22.0	86.9	87.6	96.6	250
280.5	416.1	43.2	22.9	88.9	89.4	98.4	260
305.5	423.7	44.0	24.9	93.8	93.6	102.6	285
312.4	425.7	44.3	25.5	95.2	94.7	103.7	292
312.5	425.7	44.3	25.5	95.2	94.8	148.0	292
760.5	514.8	54.5	62.0	178.4	137.3	148.0	740
765.5	515.5	54.6	62.4	179.3	137.6	148.0	745
770.5	516.2	54.6	62.8	180.2	137.9	148.0	750
850.5	527.4	55.9	69.3	194.6	142.6	151.6	830
930.5	537.7	57.1	75.8	208.8	146.8	155.8	910
1010.5	547.4	58.2	82.4	222.9	150.7	159.7	990
1090.5	556.5	59.3	88.9	237.0	154.3	163.3	1070
1170.5	565.2	55.6	95.4	246.4	156.5	165.5	1150
1250.5	573.3	51.2	101.9	255.0	158.5	167.5	1230

Table 10
Tabulated Values for Bottom Head Core Not Critical Curve (Curve B)

Pressure-Temperature Curve Calculation

(Heatup/Cooldown, Core Not Critical = Curve B)

Inputs:

Plant =	**Hope Creek**	
Component =	**Bottom Head**	(Penetrations Portion)
Vessel thickness, t =	**6.100**	inches, so √t = 2.470 √inch
Vessel Radius, R =	**126.5**	inches
ART_{NDT} =	**30.0**	°F ======> **All EFPYs**
Safety Factor =	**2.00**	
Stress Concentration Factor =	**2.30**	Bottom Head Penetrations
Cooldown Rate, CR =	**100**	°F/hr
M_m =	**2.287**	(From N-588, for inside surface axial flaw)
K_{IT} =	**8.76**	ksi*inch$^{1/2}$ (From N-588, for cooldown rate)
M_T =	**0.285**	(From Figure G-2214-2)
Temperature Adjustment =	**9.0**	°F, Instrument Uncertainty
Height of Water for a Full Vessel =	**562.5**	inches (FEM stresses include deadweight)
Pressure Adjustment =	**20.3**	psig (full vessel at 70°F)
Pressure Adjustment =	**20.5**	psig (Instrument Uncertainty)
Conversion factor from ksi to psi =	**1000**	
Unit Pressure =	**1,563**	psig
Flange RT_{NDT} =	**19.0**	°F

Gauge Temperature T (°F)	Adjusted Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{IP} (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
79.0	79.0	88.44	39.84	0	79	0
79.0	70.0	79.34	35.29	647	79	606
88.0	79.0	88.44	39.84	731	88	690
92.0	83.0	93.05	42.14	773	92	732
96.0	87.0	98.03	44.64	818	96	778
100.0	91.0	103.43	47.34	868	100	827
104.0	95.0	109.28	50.26	921	104	881
108.0	99.0	115.62	53.43	980	108	939
112.0	103.0	122.48	56.86	1043	112	1002
116.0	107.0	129.92	60.58	1111	116	1070
120.0	111.0	137.97	64.61	1185	120	1144
124.0	115.0	146.70	68.97	1265	124	1224
128.0	119.0	156.15	73.70	1351	128	1310
132.0	123.0	166.39	78.82	1445	132	1404
136.0	127.0	177.48	84.36	1547	136	1506

Table 11
Tabulated Values for Core Critical Curve (Curve C) for 32 & 48 EFPY

Pressure-Temperature Curve Calculation

(Core Critical = Curve C)

Inputs:

Plant =	**Hope Creek**	
EFPY =	**32 & 48**	
Curve A Leak Test Temperature =	**130.0**	°F (at 32 EFPY and 1,100 psig)
Curve A Leak Test Temperature =	**138.0**	°F (at 48 EFPY and 1,100 psig)
Hydro Test Pressure =	**1,563**	psig
Flange RT_{NDT} =	**19.0**	°F

Curve B Temperature Upper Vessel (°F)	Curve B Pressure for Upper Vessel (psig)	Curve C Temperature (°F)	Curve C Pressure (psig)
—	0.0	88.0	0
3.4	50.0	88.0	50
27.6	75.0	88.0	75
37.5	90.0	88.0	90
43.0	100.0	92.0	100
54.4	125.0	103.4	125
70.8	175.0	119.8	175
77.6	202.0	126.6	202
90.6	220.0	130.6	220
96.6	250.0	136.6	250
98.4	260.0	138.4	260
102.6	285.0	142.6	285
103.7	291.9	143.7	292
148.0	292.0	188.0	292
148.0	740.0	188.0	740
148.0	745.0	188.0	745
148.0	750.0	188.0	750
151.6	830.0	191.6	830
155.8	910.0	195.8	910
159.7	990.0	199.7	990
163.3	1070.0	203.3	1070
165.5	1150.0	205.5	1150
167.5	1230.0	207.5	1230

Figure 1
Pressure Test P-T Curve (Curve A) for 32 EFPY

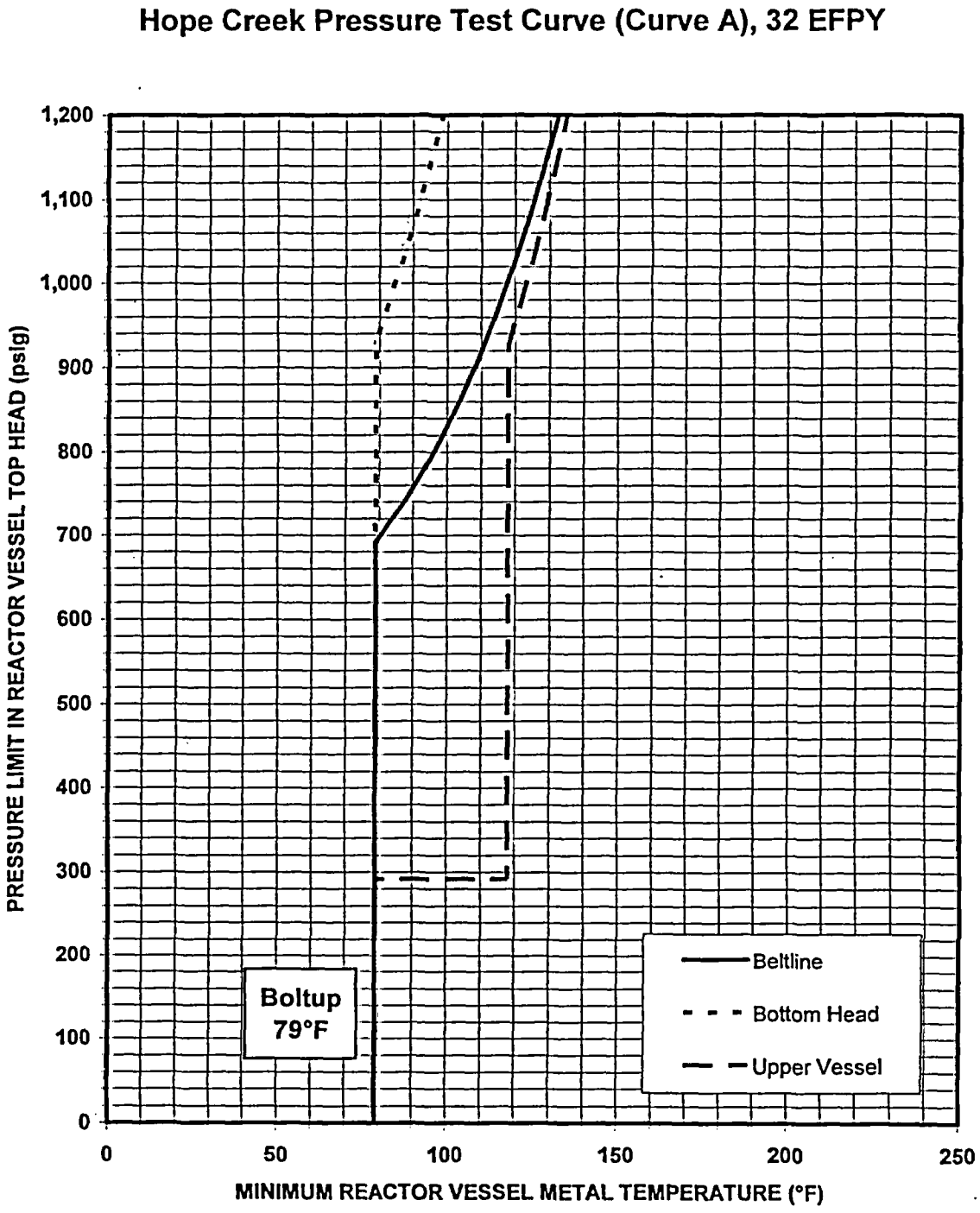

Figure 2
Pressure Test P-T Curve (Curve A) for 48 EFPY

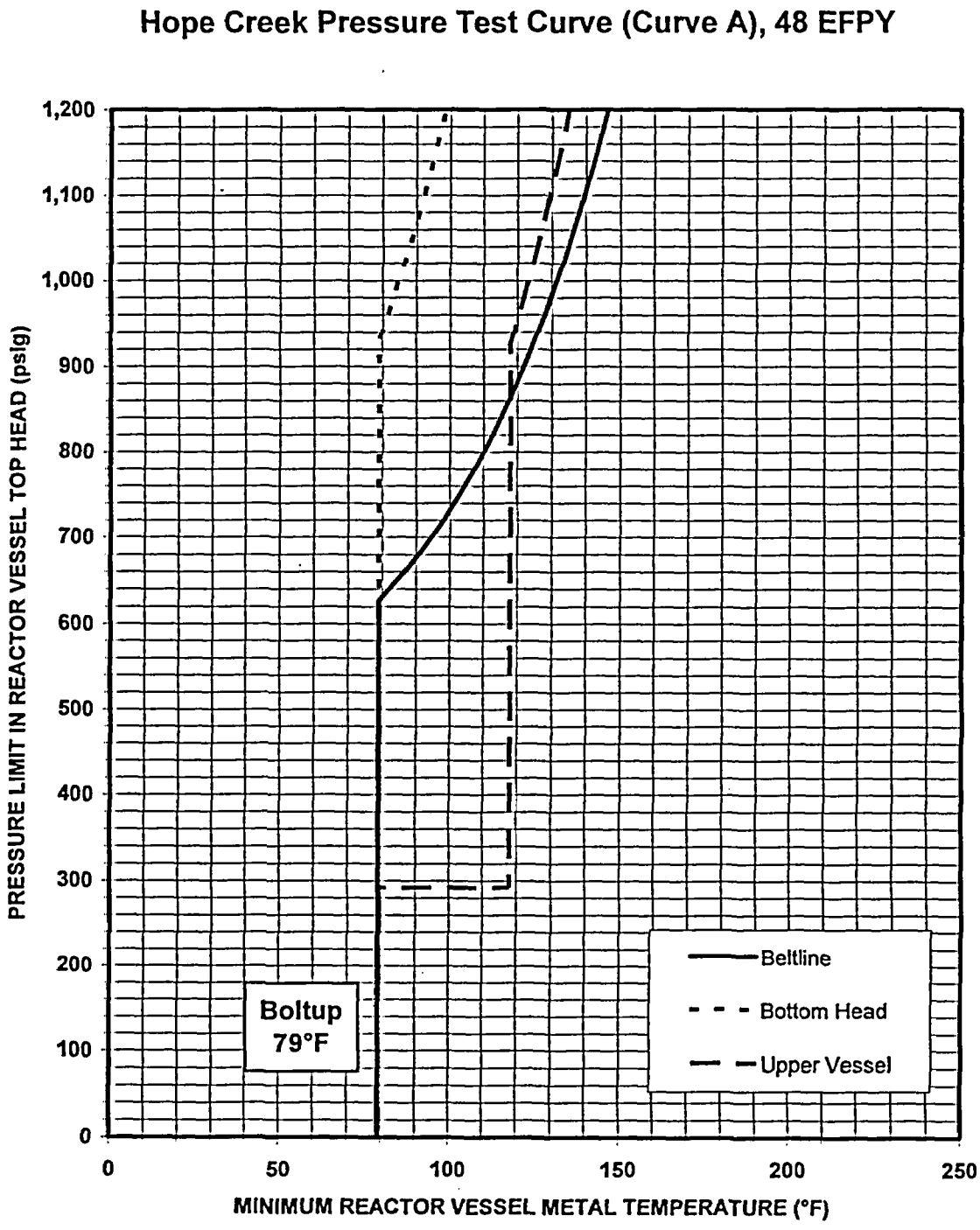

Figure 3
Core Not Critical Curve (Curve B) for 32 EFPY

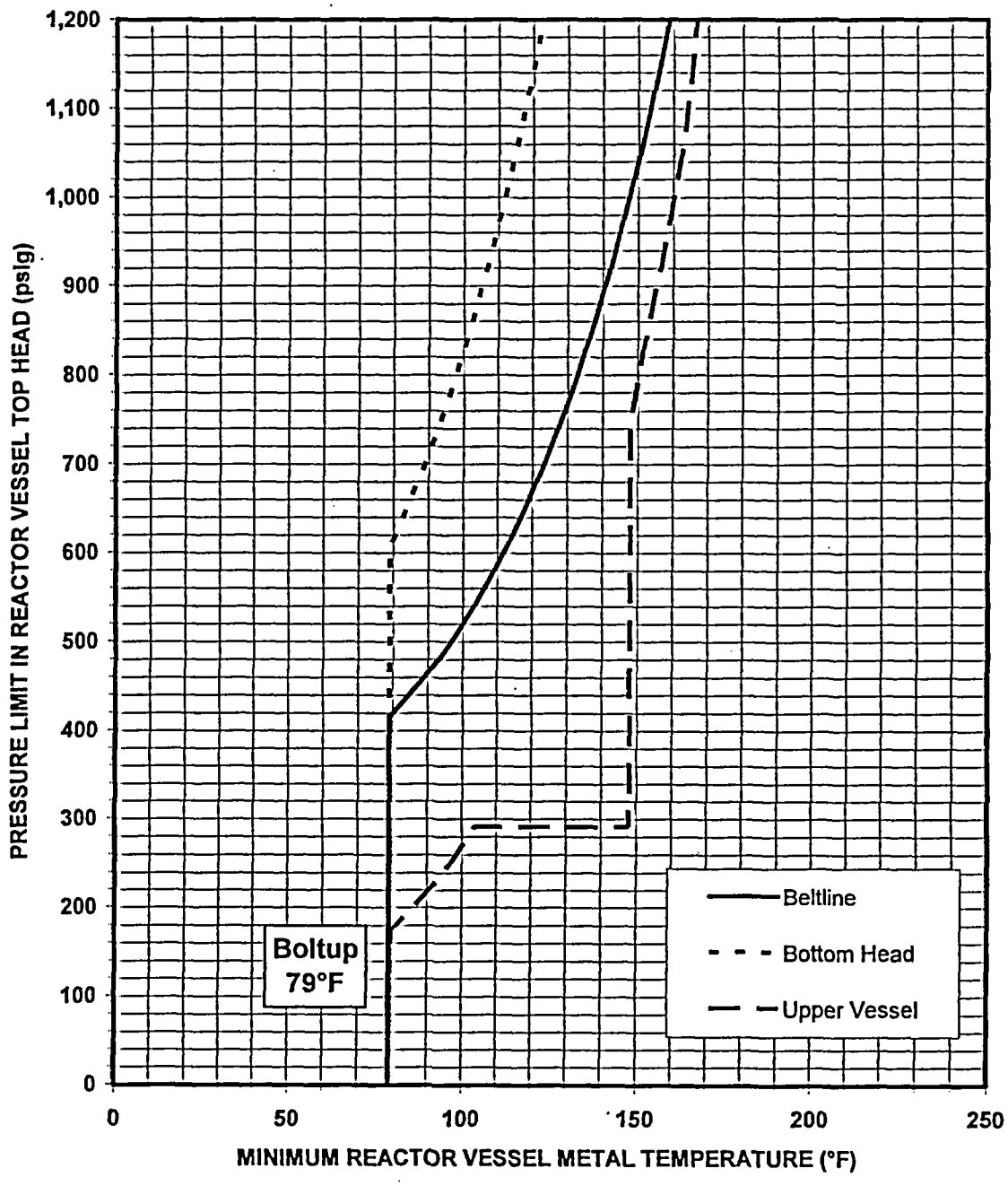

Figure 4
Core Not Critical Curve (Curve B) for 48 EFPY

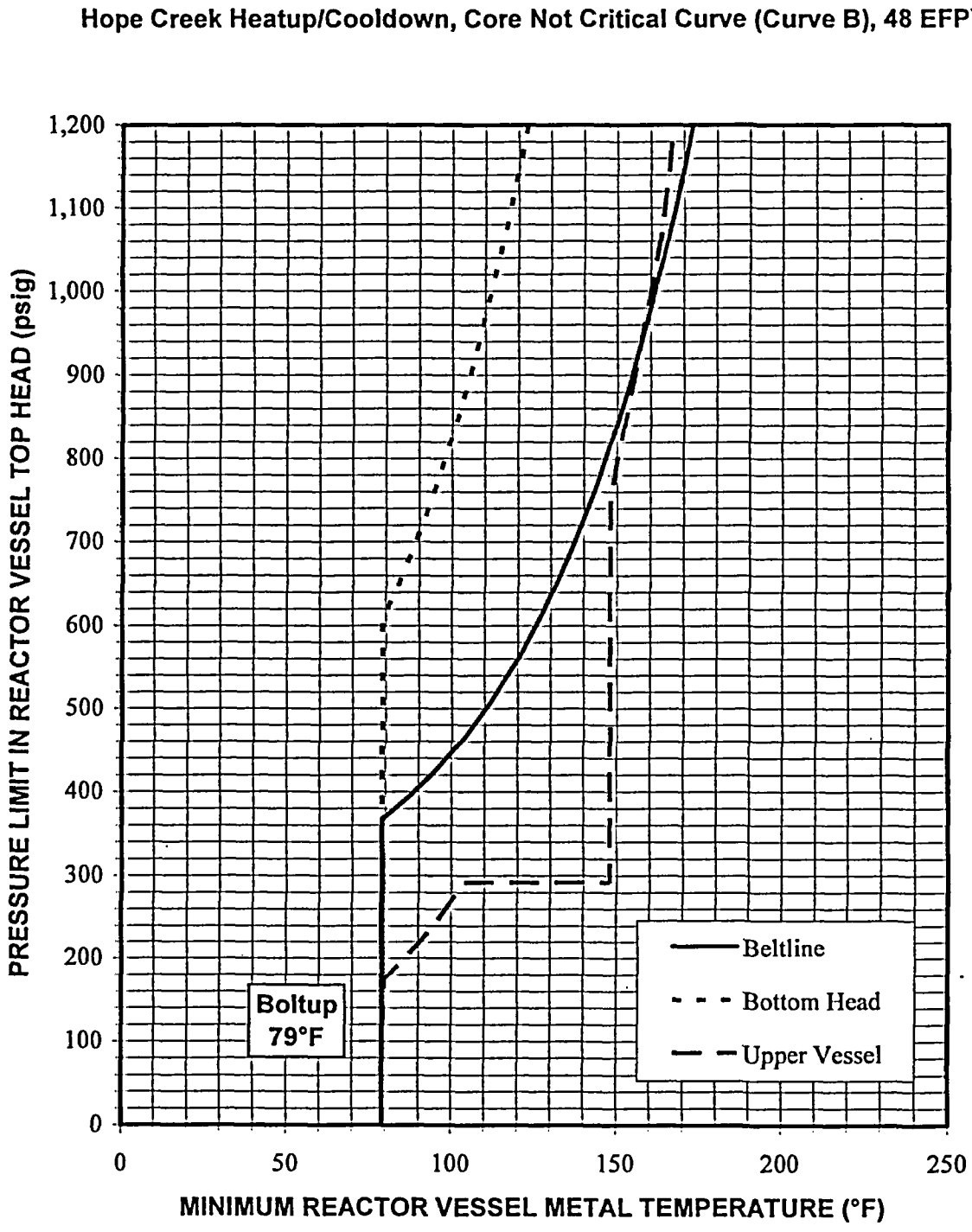

Figure 5
Core Critical Curve (Curve C) for 32 EFPY