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JUL 26 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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SUSQUEHANNA STEAM ELECTRIC STATION
REVISION TO PLA – 5341
PROPOSED AMENDMENT NO. 240 TO LICENSE NFP-14 AND PROPOSED AMENDMENT NO. 205 TO LICENSE NFP-22: CHANGES TO REACTOR PRESSURE VESSEL PRESSURE-TEMPERATURE (P-T) LIMITS AND REQUEST FOR EXEMPTION FROM THE REQUIREMENTS OF 10CFR50 SECTION 50.60(a).
PLA-5345

Docket No. 50-387 and 50-388

Reference: 1) PLA – 5341, Letter from R. G. Byram to USNRC "Proposed Amendment No.240 to License NFP-14 and Proposed Amendment No. 205 to License NFP-22: Changes to Reactor Pressure Vessel Pressure - Temperature (P-T) Limits and Request for Exemption from the Requirements of 10CFR50 Section 50.60(a).

Please refer to the last page of Attachment 5 to PLA-5341 (Reference 1) titled Code Case N-640 and remove this page from the letter. The information contained on this page has been determined to be copyright protected and therefore cannot be reproduced for filing in the Public Document Room.

If you have any questions, please contact Mr. D. L. Filchner at (610) 774-7819.

Sincerely,

R. G. Byram

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. Schaaf, NRC Project Manager

A001

Robert G. Byram
Senior Vice President and
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**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 240 TO LICENSE
NFP-14 AND PROPOSED AMENDMENT NO. 205 TO
LICENSE NFP-22: CHANGES TO REACTOR PRESSURE VESSEL
PRESSURE-TEMPERATURE (P-T) LIMITS AND REQUEST FOR
EXEMPTION FROM THE REQUIREMENTS
OF 10CFR50 SECTION 50.60(a)**
PLA-5341

**Docket No. 50-387
and 50-388**

Reference: 1) *Letter from R. G. Schaaf (USNRC) to R. G. Byram (PPL) "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Re: 1.4 Percent Power Uprate (TAC Nos. MB04444 and MB04445), dated July 6, 2001.*

2) *Letter from B. C. Buckley (USNRC) to J. A. Hutton (PECO Nuclear) "Limerick Generating Station, Unit 1 – Issuance of Amendment – RE: Update Pressure-Temperature (P-T) Limit Curves (TAC No. MA8953)," dated September 15, 2000.*

3) *Letter from L. W. Rossbach (USNRC) to O. D. Kingsley (Commonwealth Edison) "Dresden – Issuance of Amendments – Revised Pressure Temperature Limits (TAC Nos. MA8346 and MA8347) dated September 19, 2000.*

4) *Letter from S. N. Bailey (USNRC) to O. D. Kingsley (Commonwealth Edison) "Quad Cities – Issuance of Amendments – Revised Pressure-Temperature Limit," dated February 4, 2000.*

5) *Letter from J. B. Hopkins (USNRC) to M. Reandeau (Clinton Power Station), "Clinton Power Station, Unit 1 – Issuance of Amendment", dated October 31, 2000.*

The purpose of this letter is to propose revisions to the Susquehanna Steam Electric Station Units 1 and 2 Technical Specifications Figure 3.4.10-1 "Reactor Vessel Pressure vs. Minimum Vessel Temperature", and the associated TS Surveillance Requirements SR 3.4.10.1 and 3.4.10.2. In addition, changes are also made to TS Bases B3.4.10 and B3.10.1.

In accordance with 10CFR50.12, "Specific Exemptions", PPL is also requesting exemption from the requirements of 10CFR50.60(a) to permit use of ASME Code, Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1" in lieu of 10CFR50, Appendix G, paragraph IV.A.2.b.

These proposed P-T limit changes rely on recently approved ASME Boiler and Pressure Vessel Code methodology for determining allowable P-T limits through incorporation of ASME B&PV Code Case N-640. This Code Case provides an alternate method for determining the fracture toughness of reactor pressure vessel materials when determining the RPV P-T limits. This Code Case was approved for use by the ASME on February 26, 1999 and its use produces results which are based on an alternate methodology that provides adequate margin in the prevention of brittle fracture.

On July 6, 2001 PPL received approved License Amendments 194 and 169 for Power Uprate. (Reference 1). As a result of the two operating cycle limitation identified in the SER for the amendments, PPL proposes these P-T curve limit changes be effective until May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1. At that time Reg. Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" will be utilized to recalculate the P-T limits and the curves will be revised as necessary.

Attachment 1 to this letter is the "Safety Assessment" supporting this change. Attachment 2 to this letter contains the "No Significant Hazards Considerations Evaluation" performed in accordance with the criteria of 10CFR 50.92 and the categorical exclusion for an Environmental Assessment as specified in 10CFR 51.22. Attachment 3 to this letter contains the current pages of the Susquehanna SES Units 1 and 2 Technical Specifications and Technical Specification Bases marked to show the proposed changes. Attachment 4 to this letter is the "camera ready" version of the revised Technical Specification pages. Attachment 5 is the Code Case N-640 exemption request, and Attachment 6 is the technical report SIR-00-167, Rev.0 prepared for PPL by Structural Integrity Associates (SIA). #

The Susquehanna SES Plant Operations Review Committee and the Susquehanna Review Committee have reviewed the proposed changes.

This proposed change is similar to granted license amendments for changes to P-T Limits at Limerick, Dresden, Quad Cities, and Clinton. (References 2 through 5)

Code Case removed from document per PPL 7/26/01 letter due to copyright protection concerns.

PPL plans to implement the proposed changes during the Unit 1 Refueling and Inspection Outage scheduled to begin March 2, 2002 and the Unit 2 outage in March 2003. Therefore, we request NRC to complete the review of this change request by December 31, 2001 to support our scheduled implementation dates.

If you have any questions, please contact Mr. D. L. Filchner at (610) 774-7819.

Sincerely,

R. G. Byram

Attachment

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Project Manager
Mr. D. J. Allard, PA DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

PROPOSED AMENDMENT NO. 240 TO LICENSE NPF-14:
CHANGES TO REACTOR PRESSURE VESSEL
PRESSURE-TEMPERATURE (P-T) LIMITS AND REQUEST FOR
EXEMPTION FROM THE REQUIREMENTS
OF 10CFR50 SECTION 50.60(a).
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files Proposed Amendment No. 240 in support of a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 16th day of July, 2001.

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

PROPOSED AMENDMENT NO. 205 TO LICENSE NPF-22:
CHANGES TO REACTOR PRESSURE VESSEL
PRESSURE-TEMPERATURE (P-T) LIMITS AND REQUEST FOR
EXEMPTION FROM THE REQUIREMENTS
OF 10CFR50 SECTION 50.60(a).
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files Proposed Amendment No. 205 in support of a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
By:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 16th day of July, 2001.

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

Attachment 1 to PLA-5341

Safety Assessment

SAFETY ASSESSMENT

SECTION I

SUMMARY OF PROPOSED CHANGE

In accordance with 10CFR 50.90, PPL Susquehanna, LLC (PPL) proposes to revise the Susquehanna Steam Electric Station Units 1 and 2 Technical Specifications requirements for reactor pressure vessel (RPV) pressure-temperature (P-T) limits. The revised P-T limits are based in part on an alternative methodology endorsed by the American Society of Mechanical Engineers (ASME) which allows the use of ASME Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," for calculating Reactor Coolant System (RCS) P-T limits. This code case provides adequate margin in the prevention of a brittle-type fracture of the RPV and was developed based upon the knowledge gained through years of industry experience.

The proposed changes are to the Units 1 and 2 Technical Specification Figure 3.4.10-1, which shows the P-T limit curves for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality, respectively.

These changes make use of ASME Code Case N-640. They also establish the validity of the P-T curves until May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1. The current industry issue related to calculation methodology for determination of neutron fluence requires this limitation on applicability (of the curves) as presented in this proposed License amendment. It is anticipated that during the next two operating cycles for Units 1 and 2, the fluence issues will be resolved by the industry and PPL will submit revised P-T curves to the NRC based on the new fluence calculations. The final NRC review of the GE Licensing Topical Report on fluence will determine whether the traditional fluence calculation methodology was conservative.

The number of curves has been expanded from one to three to separately address beltline, bottom head, and non-beltline regions. In addition to the curve changes, editorial changes to the Limiting Condition for Operation and Surveillance Requirements of Technical Specification 3.4.10 are also proposed.

The revised P-T limits, as proposed, yield several benefits. A primary effect of the revised limits is to allow required reactor vessel hydrostatic and leak tests to be performed at a significantly lower temperature. This can substantially reduce critical path time associated with such testing during refueling outages by reducing or

eliminating the heatup time required to achieve required test conditions. The safety benefits that result from this change include a reduction in the challenges to plant operators associated with maintaining the RCS at higher test temperatures and/or within a narrow temperature band. Other benefits include reduced challenges to personnel safety as inspectors can perform their duties in lower ambient drywell temperatures, reduced dose to inspectors due to increased inspection effectiveness at the lower ambient drywell temperatures, and increased availability of systems connected to the RCS (including the Residual Heat Removal System) because of a reduced heatup and test duration.

The primary reason for developing Code Case N-640 was to reduce the excess conservatism in the current Appendix G approach. The excess conservatism present in Appendix G reduces overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument exists that the proposed change will increase plant safety and reduce personnel exposure for BWRs.

The primary result of this change for SSES is the reduction in the hydrostatic test temperature. SSES uses recirculation pump heat to reach the required pressure test temperatures. The pressure test is performed at temperatures over 200°F under the current Appendix G criteria. It is difficult to control temperatures between 200°F and 212°F throughout the system. Additionally, these high temperatures pose safety hazards to personnel performing the inspections. By reducing the test temperature, these safety issues are eliminated without reducing overall fracture margin.

We request approval of this amendment prior to January 1, 2002 to support the Unit 1 12th Refueling and Inspection outage scheduled to begin March 2, 2002. The NRC has recently approved similar P-T curve changes based on application of Code Case N-640 for BWR's at Limerick 2 (3/21/01), Dresden (9/19/00), Clinton (10/31/00) and Quad Cities (2/4/00).

SECTION II

DESCRIPTION AND BASIS (BOTH LICENSING AND DESIGN) OF THE CURRENT REQUIREMENTS

During all modes of operation, reactor vessel pressure and temperature limits are imposed to ensure that, at the existing pressure, the vessel temperature will not approach the low temperature that could lead to brittle fracture, i.e. the nil ductility temperature. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," provides the requirement that the pressure and

temperature limits as well as the associated vessel surveillance program are consistent with 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

10CFR50 Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P-T limits. 10 CFR 50 Appendix G specifies fracture toughness and testing requirements for reactor vessel material in accordance with Section XI of the ASME B&PV Code, Appendix G. 10 CFR 50, Appendix G also requires the prediction of the effects of neutron irradiation on the vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," requires that the methods in Regulatory Guide 1.99, Revision 2, be used to predict the effect of neutron irradiation on the reactor vessel material. Appendix H of 10CFR 50 requires the establishment of a surveillance program to periodically withdraw surveillance capsules from the reactor vessel.

All components in the RCS 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. 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.

The heatup and cooldown process for SSES is controlled by P-T limit curves, which are developed based on fracture mechanics analysis. These limits are developed according to Appendix G of the ASME Boiler and Pressure Vessel Code, Section Xl, and incorporate a number of safety margins. A key safety margin is the lower bound fracture toughness curve, or K_{IA} (equivalent to K_{IR}). Section XI recently approved Code Case N-640 permitting the use of the lower bound static fracture toughness curve, K_{IC}, for calculating operating P-T limit curves. The same change appears in Appendix G of Section XI in the 1999 Addenda.

The present Units 1 and 2 Technical Specification Figure 3.4.10-1 represents the reactor pressure vs. minimum vessel temperature limits. Three curves are shown which are based upon 10 CFR 50 Appendix G requirements, representative of the lower bound fracture toughness curve K_{IA} for the system hydrotest limit with fuel in the vessel (designated A), the non-nuclear heating limit (designated B), and the nuclear (core critical) limit (designated C).

SAFETY ASSESSMENT

There are two lower bound fracture toughness curves available in Section XI: K_{IA}, which is a lower bound on all static, dynamic and arrests fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. Code Case N-640 changes the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged.

SECTION III

EVALUATION OF PROPOSED CHANGE AND BASIS

The proposed changes to the P-T curves utilize Code Case N-640 and represent the lower bound on static fracture toughness curve K_{IC}. These curves will be included in the Technical Specifications as revised Figure 3.4.10-1 and new Figures 3.4.10-2 and 3.4.10-3. Additionally, editorial changes to the Limiting Conditions for Operation and Surveillance Requirements of Technical Specification 3.4.10 are proposed to accommodate the expanded P-T curve set.

Following is the justification for the proposed changes to the P-T limits, developed in accordance with the technical requirements of ASME B&PV Code, Section XI, Appendix G as modified by ASME Code Case N-640:

Use of Code Case N-640

This code case allows use of the K_{IC} fracture toughness curve shown on ASME Code, Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure A-2210-1, as the lower bound for fracture toughness in the development of the P-T limit curves. The other margins involved with the ASME Code, Section XI, Appendix G, process of determining P-T limits remain the same.

Use of Code Case N-640 is applicable to Susquehanna and is justified based upon the knowledge gained in the industry since the fracture toughness curve was created in 1974. Since that time, additional knowledge of the fracture toughness of materials and their response to applied loads has been acquired. This additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. The K_{IC} curve provides an adequate margin of safety as discussed below.

SAFETY ASSESSMENT

Use of the K_{IC} fracture toughness curve in developing P-T limits provides additional operating margin for the P-T curves, thus realizing significant benefits primarily for the pressure test. Continued operation of Susquehanna Units 1 and 2 without the relief provided by ASME Code Case N-640 unnecessarily restricts the P-T operating window. For example, the lower temperature requirement significantly lowers the duration of the pressure test. Additionally, a personnel safety benefit is provided while conducting inspections of the primary containment at elevated temperatures. It is reasonable to expect that, during the pressure tests, the drywell inspectors are more effective due to the decreased ambient temperatures. (Further justification for exemption to use Code Case N-640 is provided in Attachment 5.)

Curves Valid Until May 1, 2005 (U2) and May 1, 2006 (U1)

The Units 1 and 2 P-T limit curves have been developed for use until May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1. One of the major considerations for extended life of the RPV is irradiation of the core region, or beltline. The effect of irradiation is to shift the reference nil-ductility transition temperature (RT_{NDT}) of the beltline materials. This shift must be evaluated to meet the requirements of 10 CFR 50, Appendix G. To encompass the effects of neutron fluence due to irradiation, a maximum lifetime of 32 EFPYs was originally established. However, resolution of the current industry issues related to fluence calculation methodology requires PPL to limit applicability of the curves for the above time periods.

Structural Integrity Associates Report No. SIR-00-167, Revision 0 (Attachment 6) provides the basis for the updated curves which incorporate the effect of Code Case N-640. The three P-T curves; i.e., pressure test, non-nuclear heatup and cooldown, and core critical operation, were updated to include these effects. The updated curves ensure the P-T limits are not exceeded during the currently licensed operating period.

The SIA report was prepared for a rated reactor power level of 3441 MWT. It did not develop the revised curves for uprated power conditions of 3489 MWT. The curves were calculated for 32 EFPY and PPL will only be using these revised curves for the limited time until 2005 and 2006 for Units 2 and 1, respectively. The increase in fluence due to operation at the uprated power level for this relatively short period of time is minimal (less than 1%) at the belt line region and has no effect on the upper vessel curve, which is limiting. The new fluence calculations that are developed will be based on a reactor power level of 3489 MWT.

Separate Curves for Different RPV Regions

The current Units 1 and 2 P-T limit curves have been developed as composite curves that encompass all regions of the RPV. Due to the different material properties associated with the different regions of the RPV, as well as different operating temperatures and pressures associated with the various regions of the RPV, the revised set of curves separately provides P-T limits for the beltline, bottom head, and non-beltline regions. This allows for improved surveillance and monitoring of the RPV during heatup, cooldown and hydrostatic leak test conditions since the more restrictive limits for beltline and discontinuity regions are not applied to the bottom head region where temperatures can be significantly cooler.

SECTION IV

CONCLUSIONS

The proposed change to the P-T limits in accordance with 10 CFR 50 Appendix G and ASME Code Case N-640 is justified because the following results will occur:

- Use of Code Case N-640 actually increases plant safety by reducing the excess conservatism which caused vessel hydrostatic testing to be performed at higher temperatures.
  - Challenges to operators are reduced while conducting pressure testing because they are not unnecessarily preoccupied with maintaining RCS temperatures above 200°F within a narrow temperature band.
  - Personnel safety will be enhanced because inspections will be conducted at lower coolant temperatures thereby eliminating hot water or steam vapor hazards.
  - Effectiveness of the inspections and potential dose savings are realized because the inspectors will spend less time in a hazardous environment and are more likely to find any problems more efficiently in the lower ambient temperature containment.
- Potential outage critical path schedule savings by the reduction of time to achieve RCS temperature and RPV pressure requirements for testing.
- Adequate margin of safety is provided by the new curves that are in accordance with ASME code requirements as required by 10CFR50 Appendix G.

Attachment 2 to PLA-5341

No Significant Hazards Considerations Evaluation

and

Environmental Assessment

NO SIGNIFICANT HAZARDS CONSIDERATION AND ENVIRONMENTAL ASSESSMENT

In 10 CFR 50.92(c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

> ...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 for a testing facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of an accident of a new or different kind from any previously evaluated; or (3) Involve a significant reduction in the margin of safety.

PPL Susquehanna, LLC (PPL) has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based upon the following discussion.

Basis for no significant hazards consideration determination

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

   The changes to the calculational methodology for the pressure and temperature (P-T) limits based upon Code Case N-640 continue to provide adequate margin in the prevention of a brittle-type fracture of the reactor pressure vessel (RPV). The code case was developed based upon the knowledge gained through years of industry experience. P-T curves developed using the allowances of Code Case N-640 indeed yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the existing assumptions used for the calculation of P-T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the subject Code Case in developing the P-T limit curves will continue to provide adequate protection against nonductile-type fractures of the RPV.

   The evaluation for the Unit 1 and Unit 2 P-T limit curves for 32 EFPYs was performed using the approved methodologies of 10 CFR 50, Appendix G. The curves generated from these methods ensure the P-T limits will not be exceeded during any phase of reactor operation. Resolution of the current industry issues

related to fluence calculation methodology requires PPL to limit applicability of the curves to May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed changes will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events. Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

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

   The proposed changes provide more operating margin in the P-T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with the benefits being primarily realizable during the pressure tests. Operation in the "new" regions of the newly developed P-T curves has been analyzed in accordance with the provisions of ASME Code, Section XI, Appendix G; 10 CFR 50 Appendix G, and ASME Code Case N-640, thus providing adequate protection against a nonductile-type fracture of the RPV. These proposed changes do not create the possibility of any new or different type of accident. Further, they do not result in any new or unanalyzed operation of any system or piece of equipment important to safety.

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

   As mentioned previously, the revised P-T curves provide more operating margin and thus, more operational flexibility than the current P-T curves. However, the industry experience since the inception of the P-T limits in 1974 confirms that some of the existing methodologies used to develop P-T curves is unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N-640 takes advantage of the acquired knowledge by establishing more realistic methodologies for the development of P-T curves.

   Use of Code Case N-640 to develop the revised P-T curves utilized the K_{IC} fracture toughness curve in lieu of the K_{IA} curve as the lower bound for fracture toughness. Use of the K_{IC} curve to determine lower bound fracture toughness is more technically correct than using the K_{IA} curve. P-T curves based on the K_{IC} fracture toughness limits enhance overall plant safety by expanding the P-T window in the

low-temperature operating region. The benefits which occur are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment with no decrease to the margin of safety.

Therefore, operational flexibility is gained without a reduction in the margin of safety to RPV brittle fracture.

The development of the P-T curves to 32 EFPYs was performed per the guidelines of 10 CFR 50, and thus, the margin of safety is not reduced as the result of the proposed changes. Resolution of the current industry issues related to fluence calculation methodology requires PPL to limit applicability of the curves to May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1.

NO SIGNIFICANT HAZARDS CONSIDERATION AND ENVIRONMENTAL ASSESSMENT

Environmental Assessment

10 CFR 51.22(c) (9) provides criterion for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment would not:

1. Involve a significant hazards consideration;

2. Result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site;

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

PPL has determined that the proposed Technical Specifications changes described in Attachment 1 Safety Assessment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c) (9).

Accordingly, pursuant to 10 CFR 51.22, no environmental impact statement needs to be prepared in connection with the issuance of the license amendment for the proposed changes. The basis for this determination using the above criteria follows:

1. As demonstrated in this enclosure, the proposed changes do not involve a significant hazards consideration.

2. The proposed changes do not result in a change to the types of effluents or in the amounts of effluents released offsite. These proposed changes involve the reactor vessel P-T limits. They do not involve changes to the radioactive waste processing systems or to radioactive waste effluent monitors. Accordingly, the changes do not require the radioactive waste processing systems to perform any different function than they are designed to perform nor do they change the operation or testing of any such system.

NO SIGNIFICANT HAZARDS CONSIDERATION AND ENVIRONMENTAL ASSESSMENT

3. The proposed changes do not result in a significant increase in occupational radiation exposure. Inspections of primary containment during pressure tests will continue to be done in accordance with as low as reasonably achievable (ALARA) principles. As discussed in Attachment 5 Request for Exemption from the Requirements of 10 CFR 50 Appendix G, the changes will result in lower temperatures in the primary containment for the inspectors, but will not result in additional time for inspection. In that respect, their exposure time will not be increased.

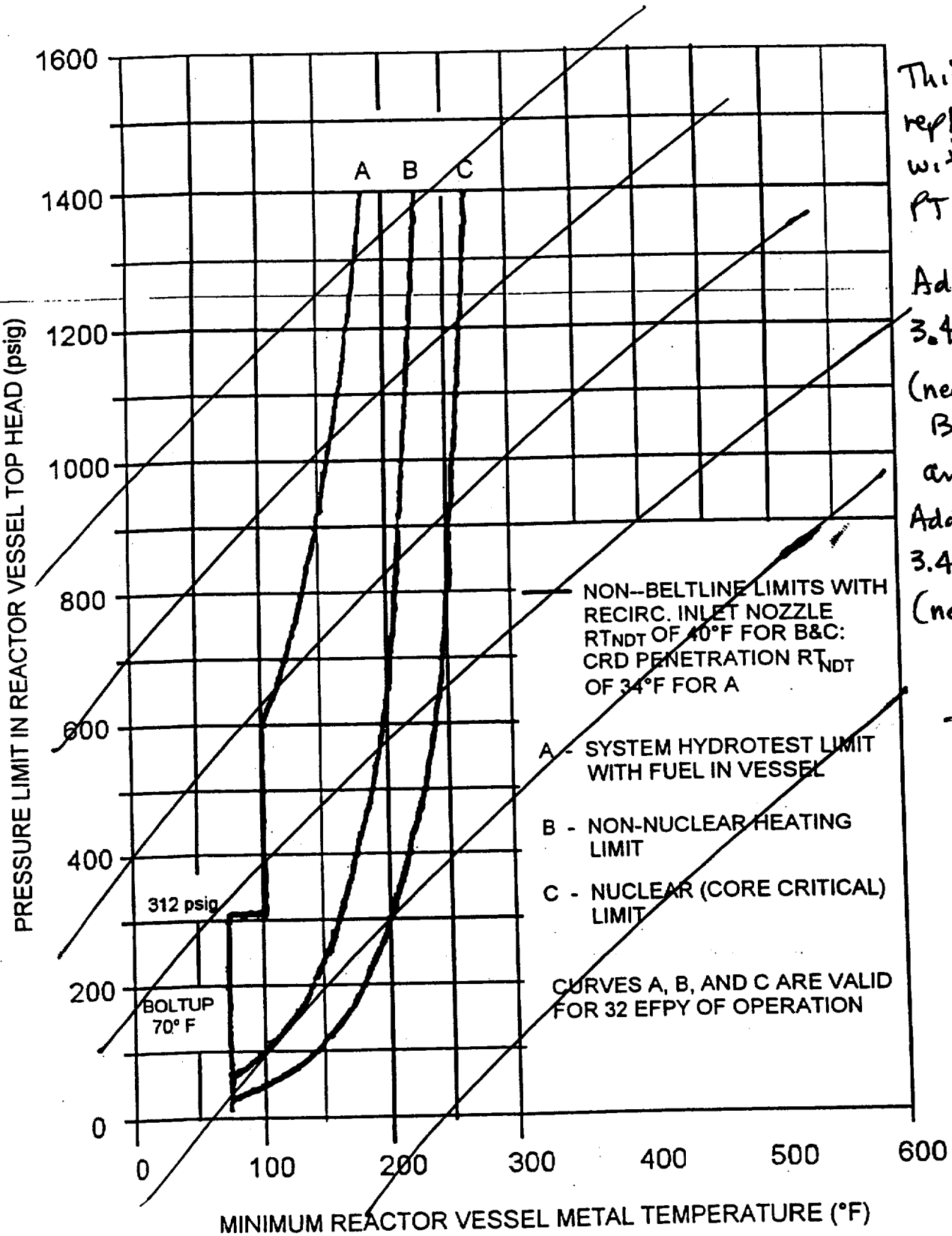
Attachment 3 to PLA-5341

Technical Specification Markups
Technical Specification Bases Markups
(Units 1&2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are to the right of the most limiting curve specified in Figures 3.4.10-1 through 3.4.10-3; and b. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-2, Curve B, and Figure 3.4.10-3, Curve C. ---------- RCS heatup and cooldown rates are ≤ 100°F in any one hour period; and c. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-1, Curve A. ---------- RCS heatup and cooldown rates are ≤ 20°F in any one hour period.	30 minutes
SR 3.4.10.2	Verify RCS pressure and RCS temperature are to the right of the criticality limit (Curve C) specified in Figure 3.4.10-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

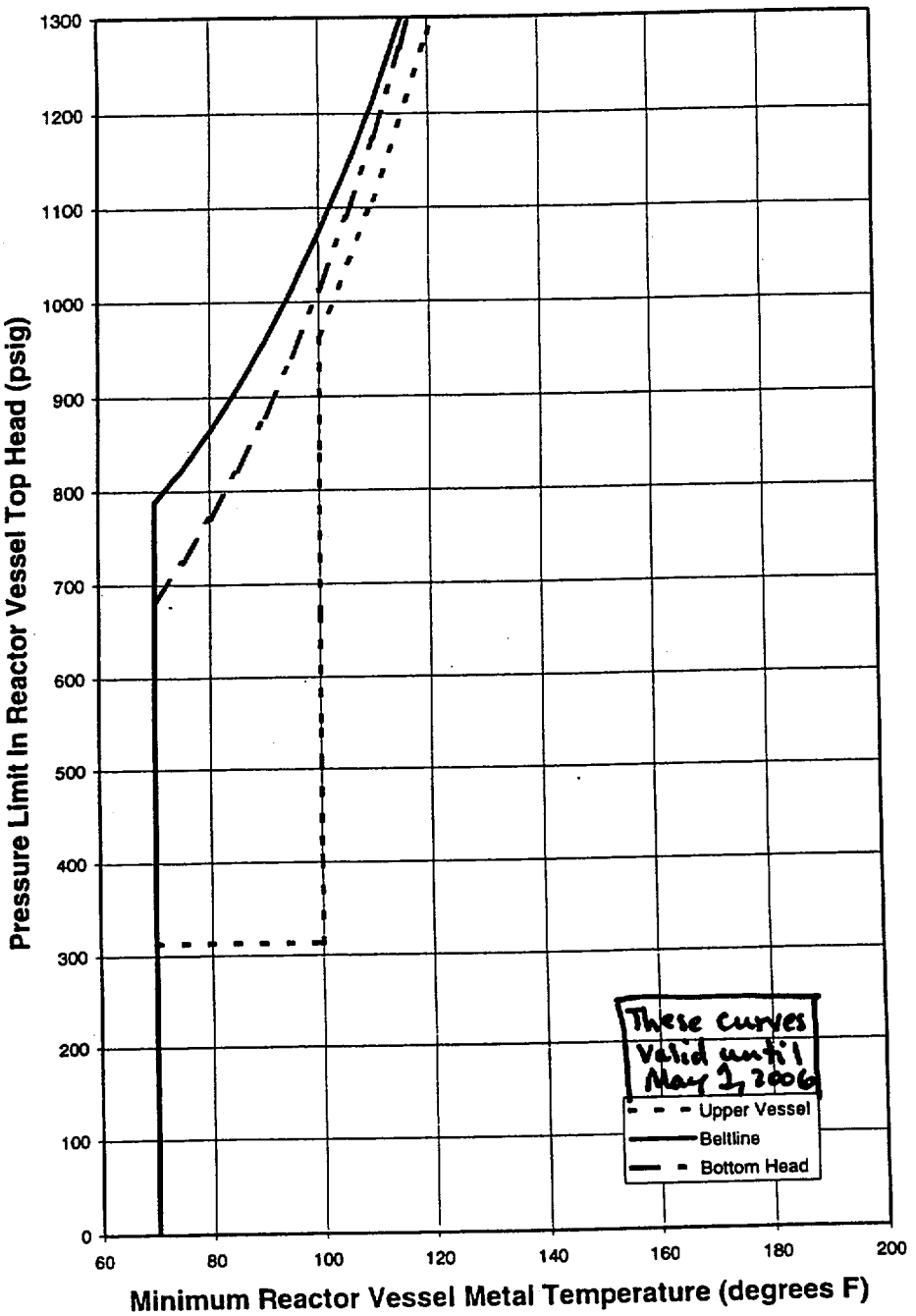
(continued)



This figure replaced with new PT Curve A, Add Figure 3.4.10-2 (new Curve B) and Add Figure 3.4.10-3 (new Curve C)

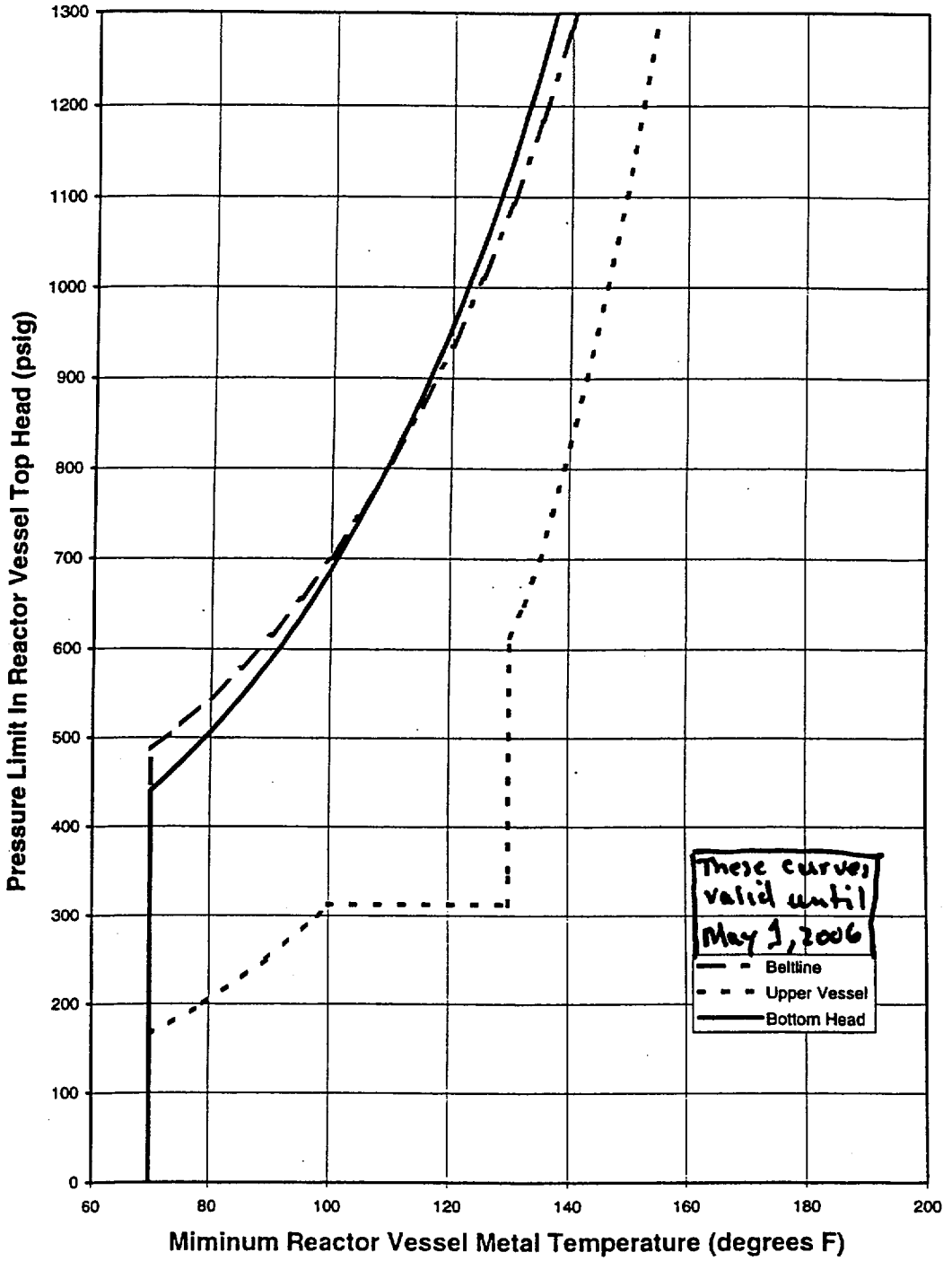
REACTOR VESSEL PRESSURE VS. MINIMUM VESSEL TEMPERATURE

Figure 3.4.10-1 ~~(page 1 of 1)~~



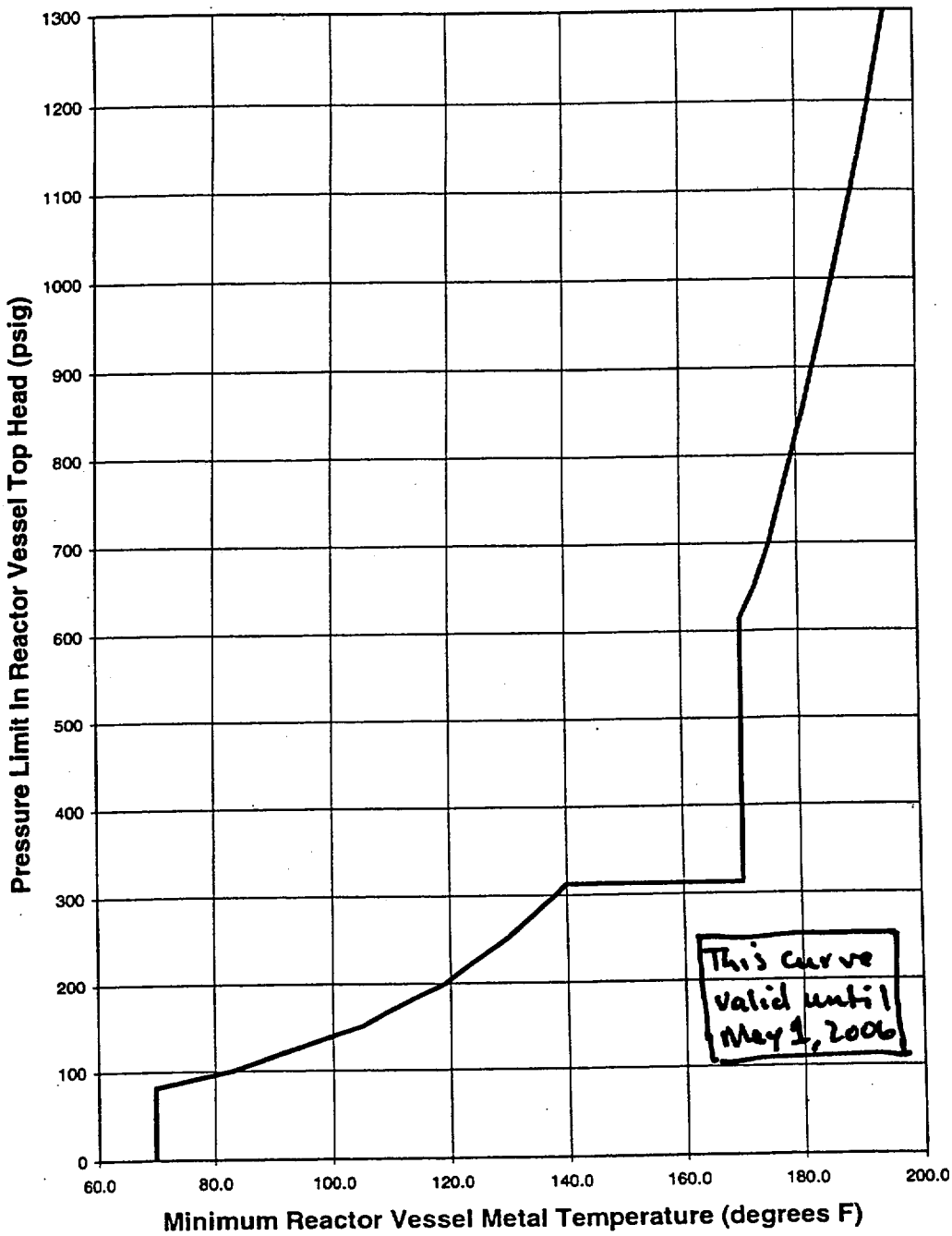
~~Figure 1~~ Figure 3.4.10-1

Pressure Test P-T Curve (Curve A) ~~for Unit 1~~



~~Figure 2~~ Figure 3.4.10-2

Core Not Critical Curve (Curve B) ~~for Unit 1~~



~~Figure 5~~ Figure 3.4.10-3

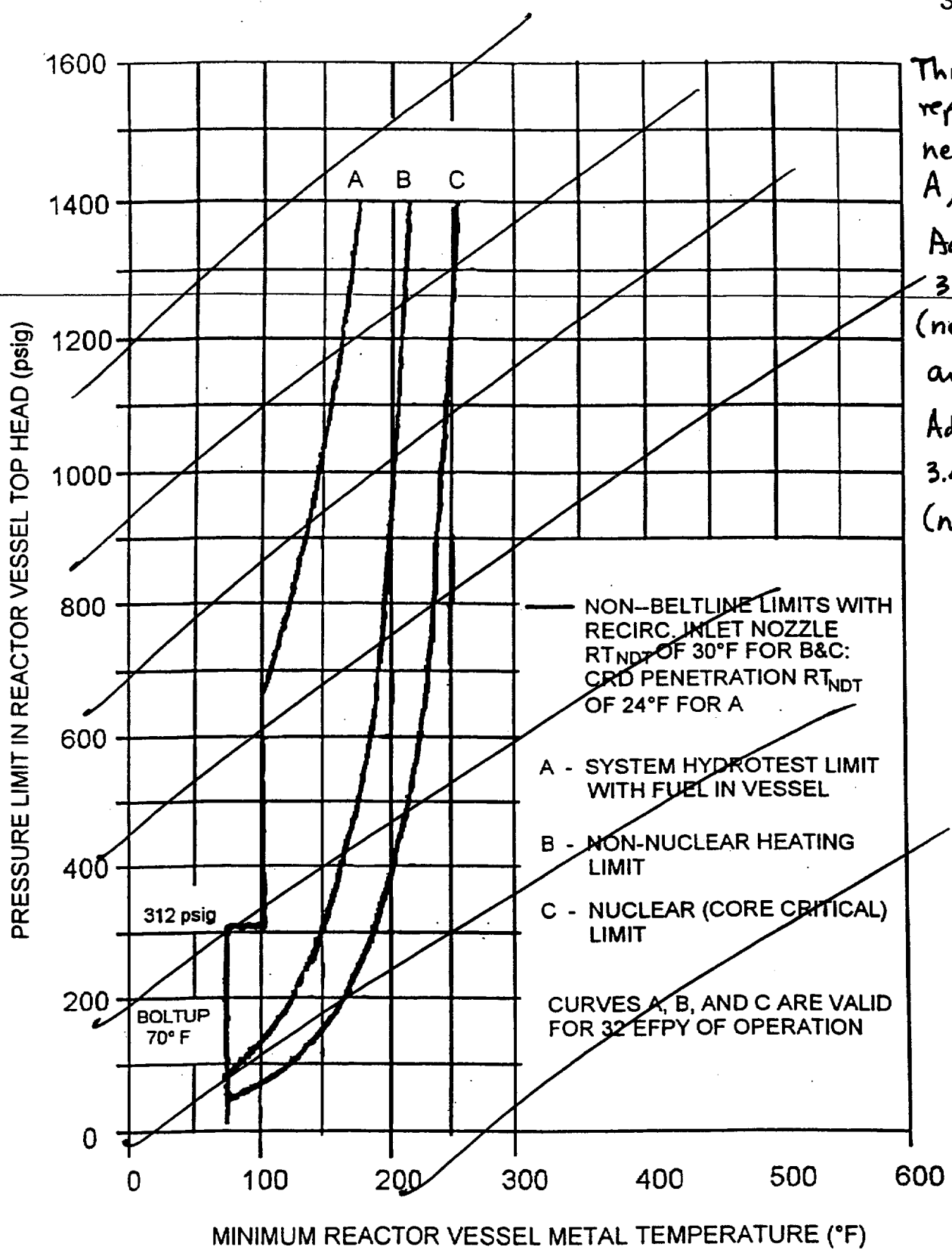
Core Critical Curve (Curve C) ~~for Unit 1~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify a. RCS pressure and RCS temperature are to the right of the ~~applicable limit~~ specified in ~~Figure 3.4.10-1~~ and most limiting curve Figures 3.4.10-1 through 3.4.10-3 b. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-~~1~~ 2, Curves B, ~~and C~~ and Figure 3.4.10-3, Curve C. ---------- RCS heatup and cooldown rates are ≤ 100°F in any one hour period; and c. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-1, Curve A. ---------- RCS heatup and cooldown rates are ≤ 20°F in any one hour period.	30 minutes
SR 3.4.10.2	Verify RCS pressure and RCS temperature are to the right of the criticality limit (Curve C) specified in Figure 3.4.10-~~1~~ 3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

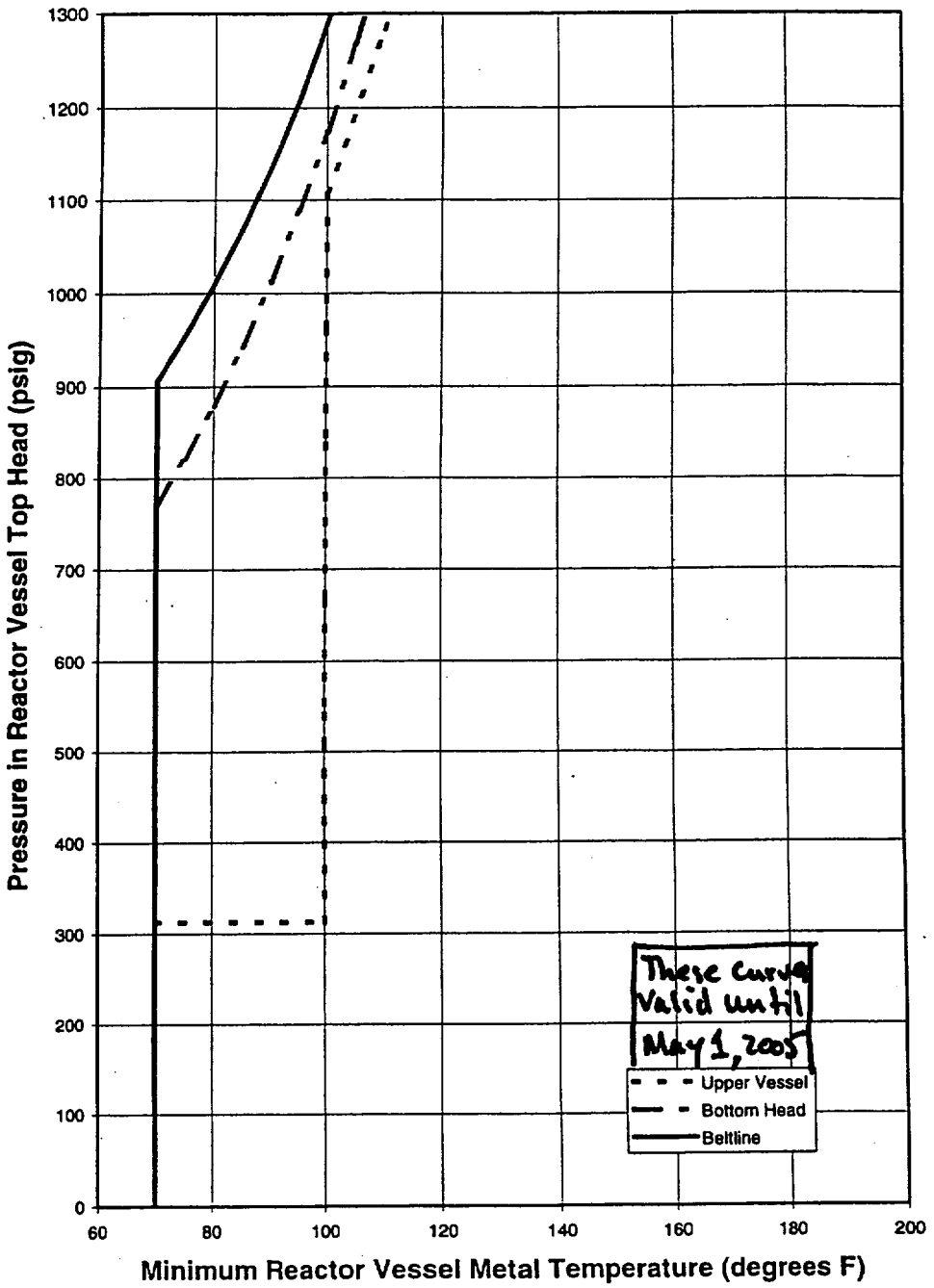
(continued)

This figure replaced with new PT curve A, Add Figure 3.4.10-2 (new Curve B) and Add Figure 3.4.10-3 (new Curve C)



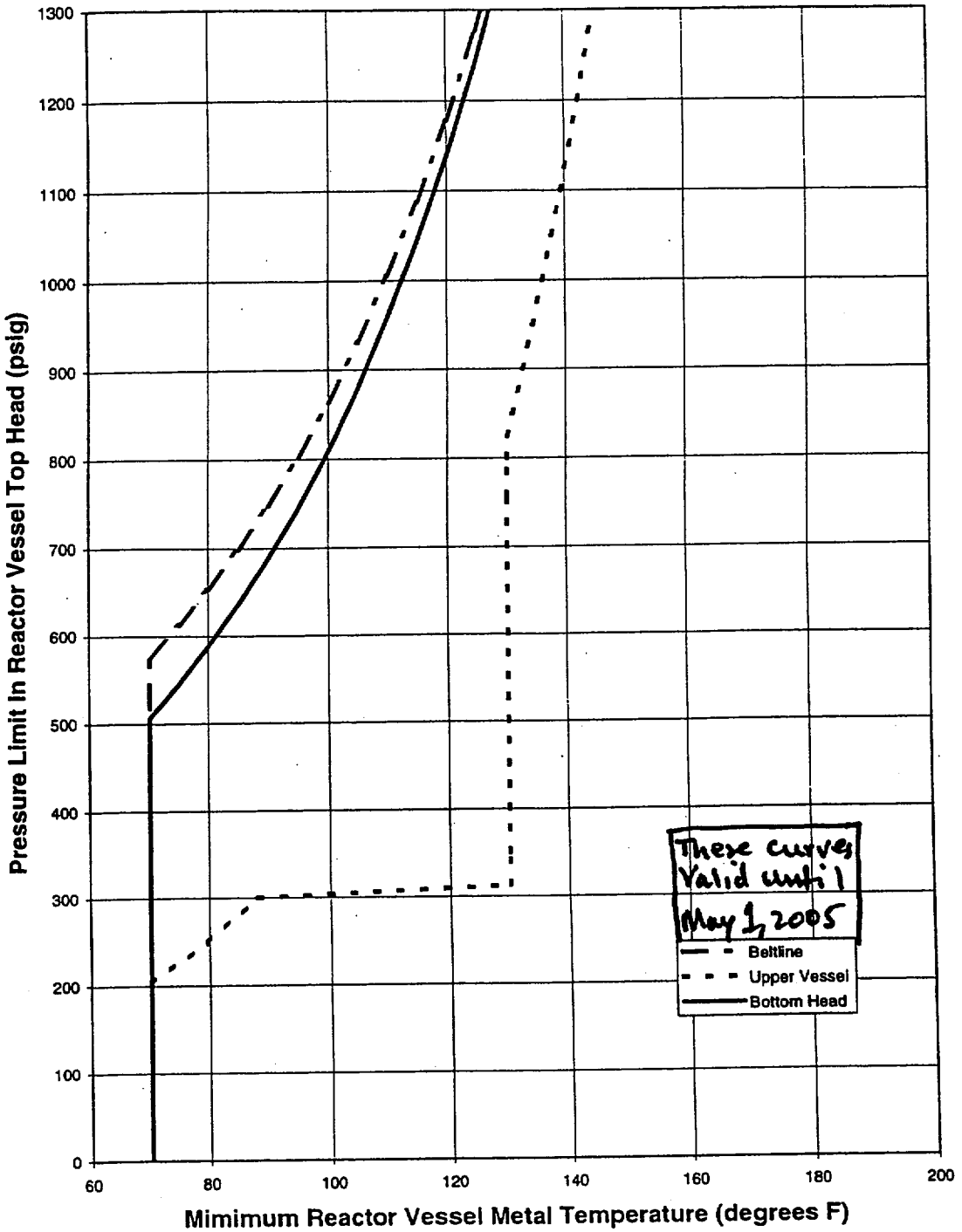
REACTOR VESSEL PRESSURE VS. MINIMUM VESSEL TEMPERATURE

Figure 3.4.10-1 ~~(page 1 of 1)~~



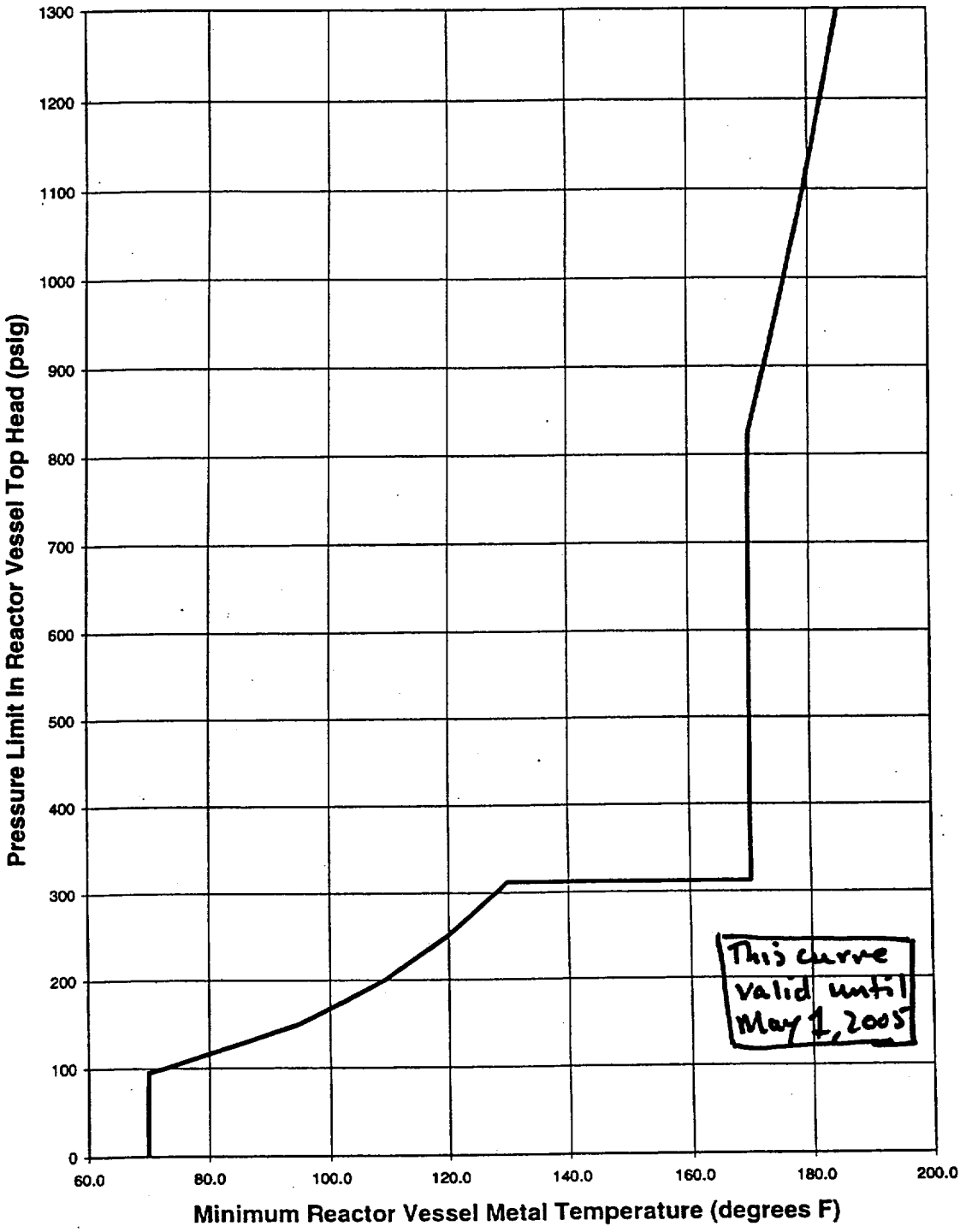
~~Figure 2~~ Figure 3.4.10-1

Pressure Test P-T Curve (Curve A) ~~for Unit 2~~



~~Figure 4~~ Figure 3.4.10-2

Core Not Critical Curve (Curve B) ~~for Unit 2~~



~~Figure 6~~ Figure 3.4.10-3
Core Critical Curve (Curve C) ~~for Unit 2~~

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and limits for the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section ~~III,~~ XI, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 9).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are to the right of the applicable curves specified in ~~Figure 3.4.10-1~~ Figures 3.4.10-1 through 3.4.10-3 and within the applicable heat-up or cool down rate specified in SR 3.4.10.1 during RCS heatup, cooldown, and inservice leak and hydrostatic testing;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant ≤ 145°F during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel are ≤ 50°F during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow;

d. RCS pressure and temperature are to the right of the criticality limits specified in Figure 3.4.10-~~1~~ 3 prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Figures 3.4.10-1 through 3.4.10-3

Verification that operation is within limits (i.e., to the right of the applicable curves in ~~Figure 3.4.10-1~~) is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1 (continued)

increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

Notes to the acceptance criteria for heatup and cooldown rates ensure that more restrictive limits are applicable when the P/T limits associated with hydrostatic and inservice testing are being applied.

SR 3.4.10.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits (i.e., to the right of the criticality curve in Figure 3.4.10-~~1~~3) before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of reactor criticality. Although no Surveillance Frequency is specified, the requirements of SR 3.4.10.2 must be met at all times when the reactor is critical.

SR 3.4.10.3 and SR 3.4.10.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits

(continued)

BASES

power levels > 30% of RTP and with single loop flow rate ≥ 21,320 gpm (50% of rated loop flow). Therefore, SR 3.4.10.5 and SR 3.4.10.6 have been modified by a Note that requires the Surveillance to be met only under these conditions. The Note for SR 3.4.10.6 further limits the requirement for this Surveillance to exclude comparison of the idle loop temperature if the idle loop is isolated from the RPV since the water in the loop can not be introduced into the remainder of the Reactor Coolant System.

SR 3.4.10.7, SR 3.4.10.8, and SR 3.4.10.9

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

REFERENCES

~~REFERENCES (continued)~~

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section ~~III~~ XI, Appendix G.
3. ASTM E 185-73
4. 10 CFR 50, Appendix H.

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Inservice Leak and Hydrostatic Testing Operation

BASES

BACKGROUND

The purpose of this Special Operations LCO is to allow certain reactor coolant pressure tests to be performed in MODE 4 with temperatures as high as 212°F when operational conditions or the metallurgical characteristics of the reactor pressure vessel (RPV) require the pressure testing at temperatures > 200°F (normally corresponding to MODE 3).

Inservice hydrostatic testing and system leakage pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) are performed prior to the reactor going critical after a refueling outage. Recirculation pump operation and a water solid RPV (except for an air bubble for pressure control) are used to achieve the necessary temperatures and pressures required for these tests. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.10, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence.

With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RPV P/T limit curves are performed as necessary, based upon the results of analyses of irradiated surveillance specimens removed from the vessel. Hydrostatic and leak testing may eventually be required with minimum reactor coolant temperatures > 200°F.

The hydrostatic test requires increasing pressure to 1035 (+10, -0) psig. ~~because of the expected increase in reactor vessel fluence.~~ The minimum allowable vessel temperature according to LCO 3.4.10 is ~~increased to 160°F~~ approximately 110°F for Unit 1. The hydrostatic test pressure does not exceed the Safety Limit of 1375 psig.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and limits for the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section ~~III,~~ XI, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 9).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are to the right of the applicable curves specified in ~~Figure 3.4.10-1~~ Figures 3.4.10-1 through 3.4.10-3 and within the applicable heat-up or cool down rate specified in SR 3.4.10.1 during RCS heatup, cooldown, and inservice leak and hydrostatic testing;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant ≤ 145°F during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel are ≤ 50°F during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow;

d. RCS pressure and temperature are to the right of the criticality limits specified in Figure 3.4.10-~~1~~ 3 prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that operation is within limits (i.e., to the right of the applicable curves in ~~Figure 3.4.10-1~~ Figures 3.4.10-1 through 3.4.10-3) is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1 (continued)

increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

Notes to the acceptance criteria for heatup and cooldown rates ensure that more restrictive limits are applicable when the P/T limits associated with hydrostatic and inservice testing are being applied.

SR 3.4.10.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits (i.e., to the right of the criticality curve in Figure 3.4.10-~~2~~3) before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of reactor criticality. Although no Surveillance Frequency is specified, the requirements of SR 3.4.10.2 must be met at all times when the reactor is critical.

SR 3.4.10.3 and SR 3.4.10.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section ~~III~~ XI, Appendix G.

3. ASTM E 185-73

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. NEDO-21778-A, December 1978.

8. FSAR, Section 15.4.4.

9. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Inservice Leak and Hydrostatic Testing Operation

BASES

BACKGROUND

The purpose of this Special Operations LCO is to allow certain reactor coolant pressure tests to be performed in MODE 4 with temperatures as high as 212°F when operational conditions or the metallurgical characteristics of the reactor pressure vessel (RPV) require the pressure testing at temperatures > 200°F (normally corresponding to MODE 3).

Inservice hydrostatic testing and system leakage pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) are performed prior to the reactor going critical after a refueling outage. Recirculation pump operation and a water solid RPV (except for an air bubble for pressure control) are used to achieve the necessary temperatures and pressures required for these tests. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.10, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence.

With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RPV P/T limit curves are performed as necessary, based upon the results of analyses of irradiated surveillance specimens removed from the vessel. Hydrostatic and leak testing may eventually be required with minimum reactor coolant temperatures > 200°F.

The hydrostatic test requires increasing pressure to 1035 (+10, -0) psig. ~~because of the expected increase in reactor vessel fluence;~~ The minimum allowable vessel temperature according to LCO 3.4.10 is ~~increased to 150°F~~ 100°F for Unit 2. The hydrostatic test pressure does not exceed the Safety Limit of 1375 psig.

(continued)

Attachment 4 to PLA-5341

"Camera Ready" Technical Specifications

(Units 1&2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are to the right of the most limiting curve specified in Figures 3.4.10-1 through 3.4.10-3; and b. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-2, Curve B, and Figure 3.4.10-3, Curve C. ---------- RCS heatup and cooldown rates are ≤ 100°F in any one hour period; and c. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-1, Curve A. ---------- RCS heatup and cooldown rates are ≤ 20°F in any one hour period.	30 minutes
SR 3.4.10.2	Verify RCS pressure and RCS temperature are to the right of the criticality limit (Curve C) specified in Figure 3.4.10-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

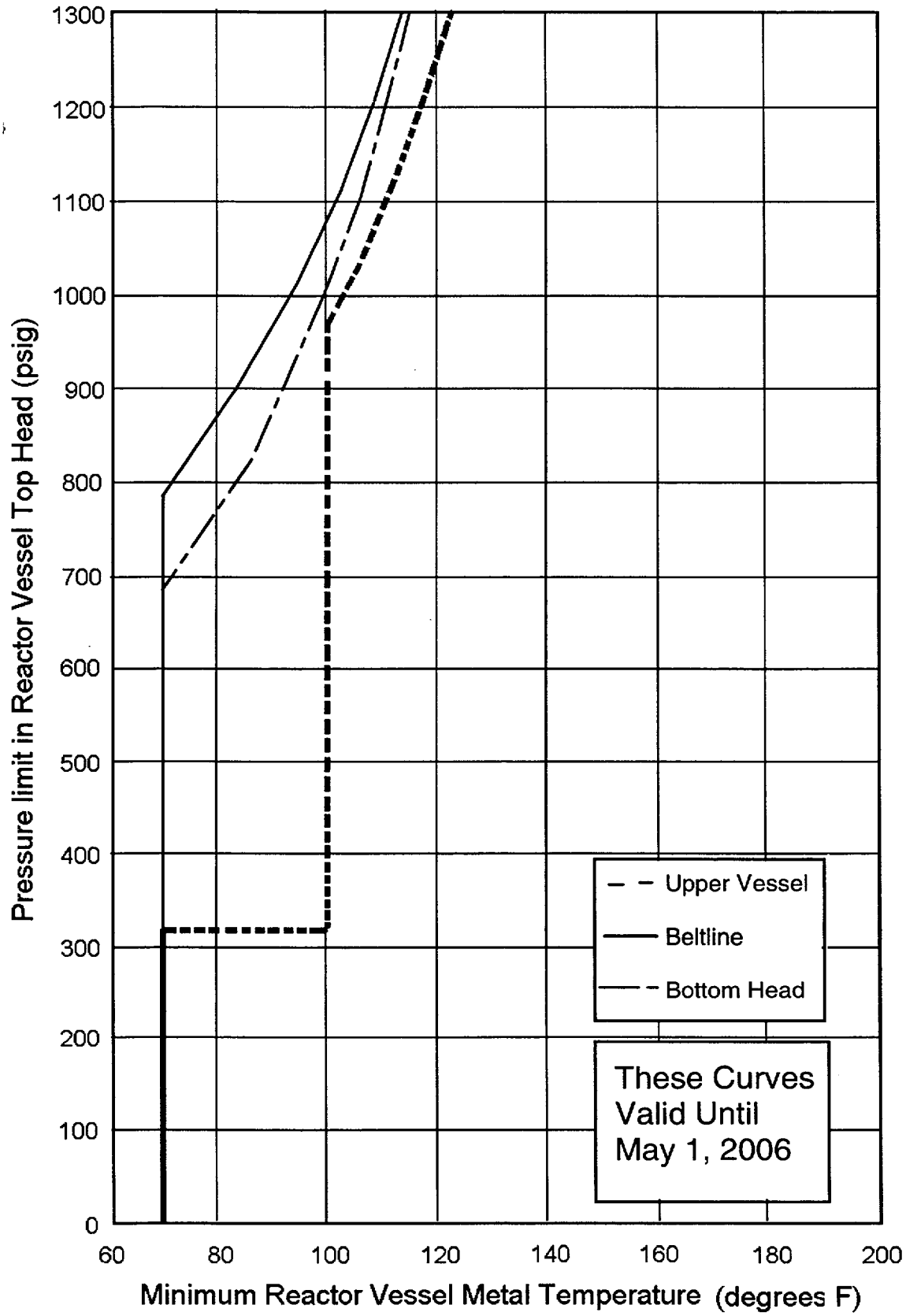


Figure 3.4.10-1
System Hydrotest Limit with Fuel in Vessel (Curve A)

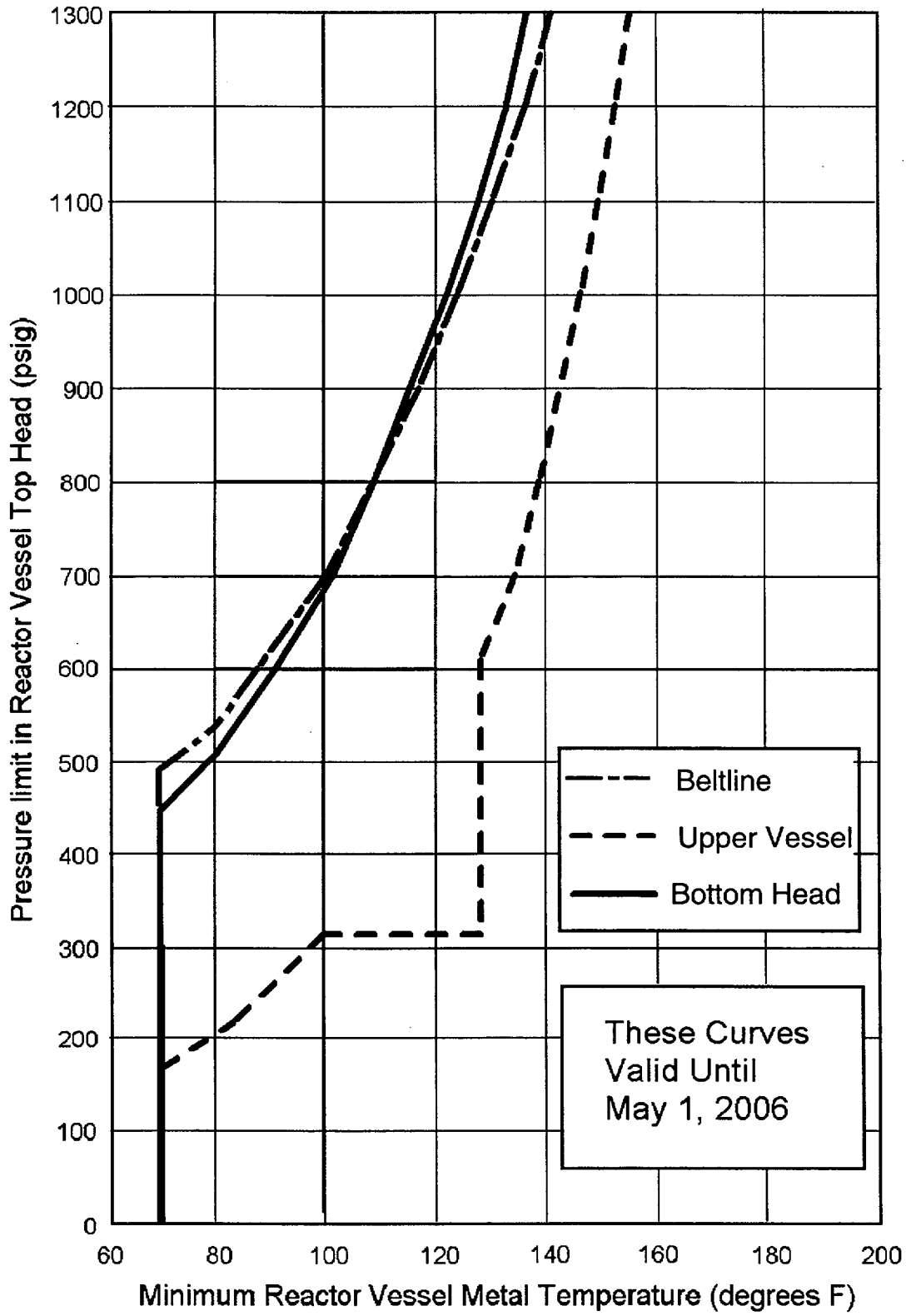


Figure 3.4.10-2
Non-Nuclear Heating Limit (Curve B)

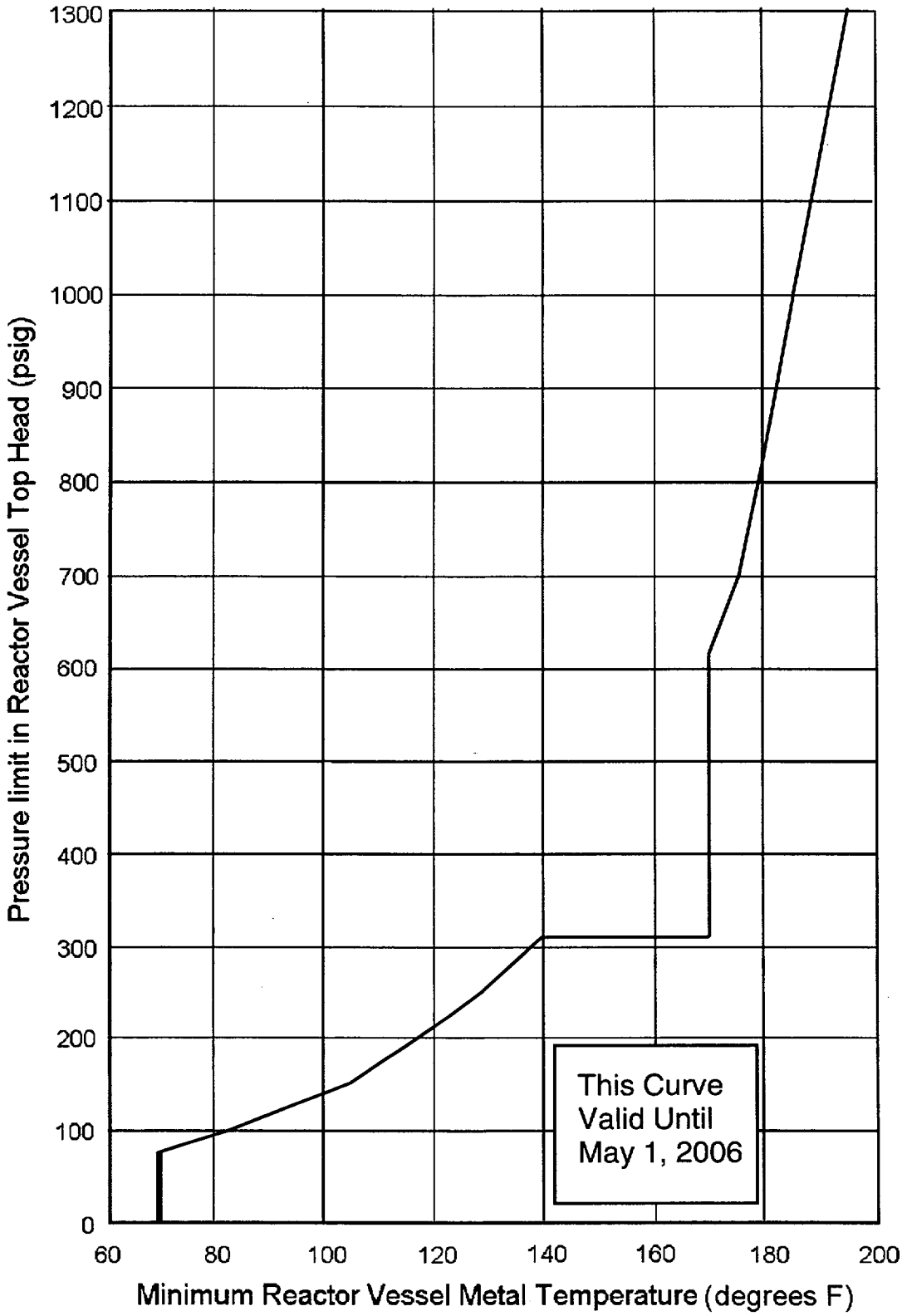


Figure 3.4.10-3
Nuclear (Core Critical) Limit (Curve C)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are to the right of the most limiting curve specified in Figures 3.4.10-1 through 3.4.10-3; and b. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-2, Curve B, and Figure 3.4.10-3, Curve C. ---------- RCS heatup and cooldown rates are ≤ 100°F in any one hour period; and c. ----------NOTE---------- Only applicable when governed by Figure 3.4.10-1, Curve A. ---------- RCS heatup and cooldown rates are ≤ 20°F in any one hour period.	30 minutes
SR 3.4.10.2	Verify RCS pressure and RCS temperature are to the right of the criticality limit (Curve C) specified in Figure 3.4.10-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

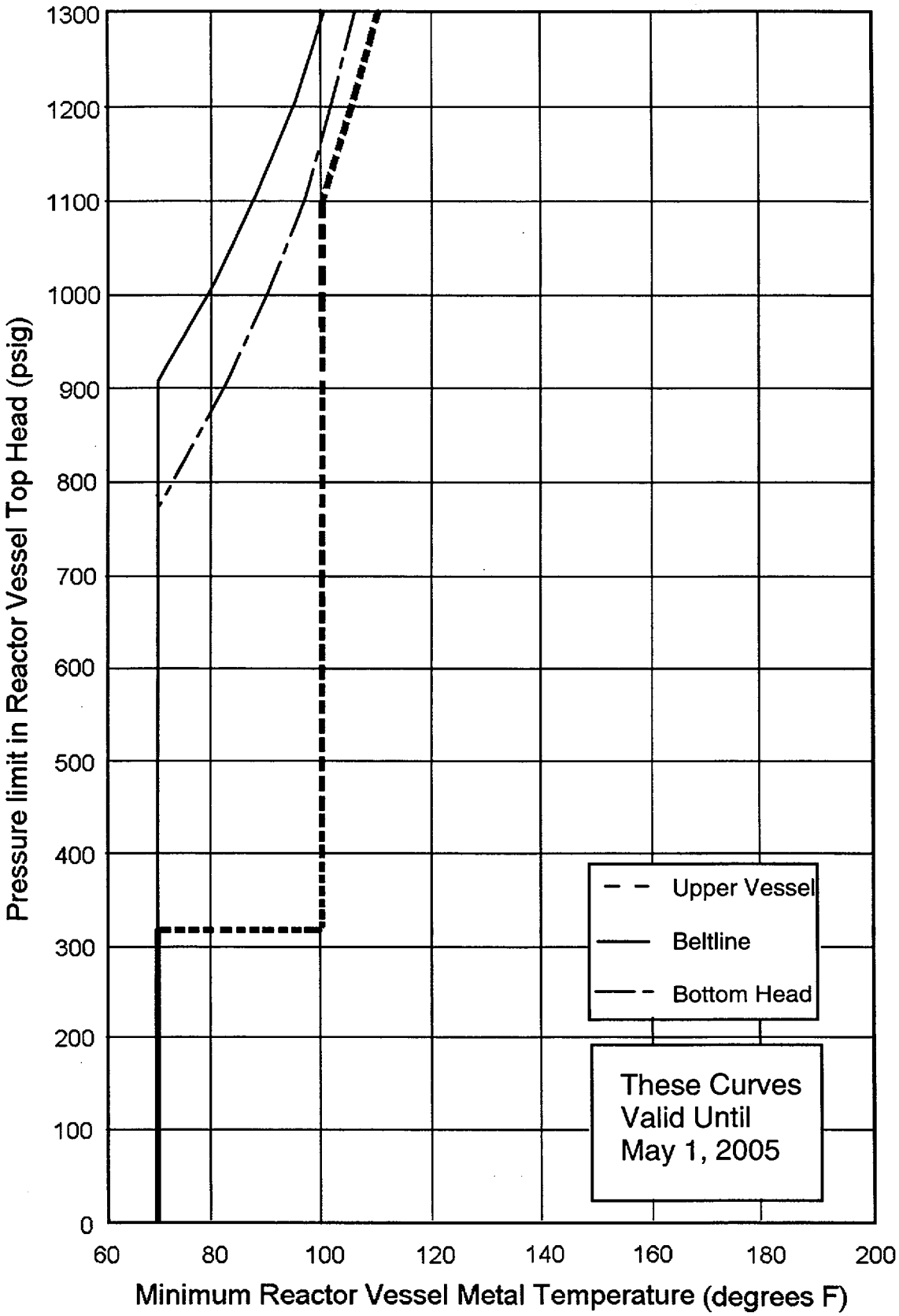


Figure 3.4.10-1
System Hydrotest Limit with Fuel in Vessel (Curve A)

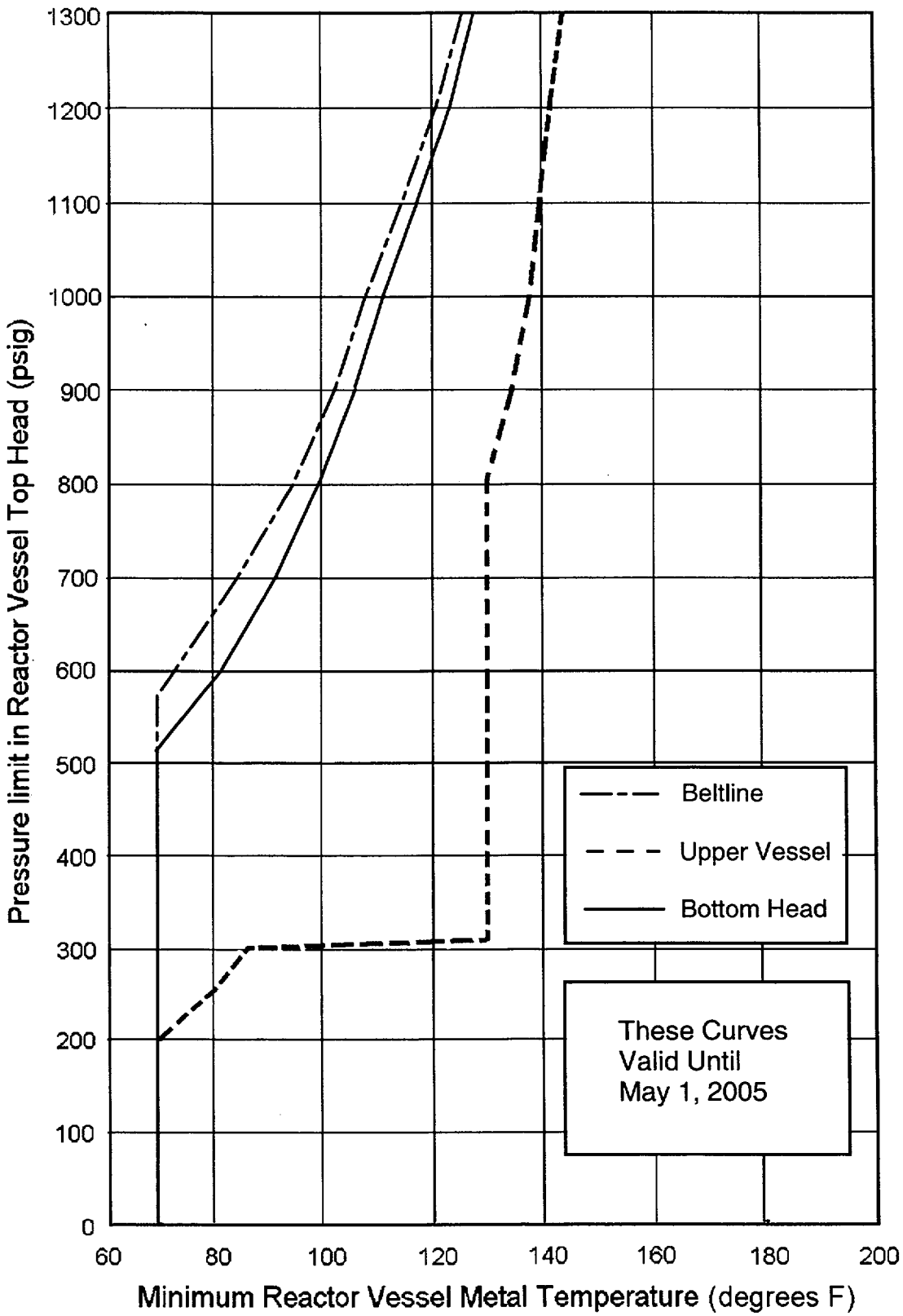


Figure 3.4.10-2
Non-Nuclear Heating Limit (Curve B)

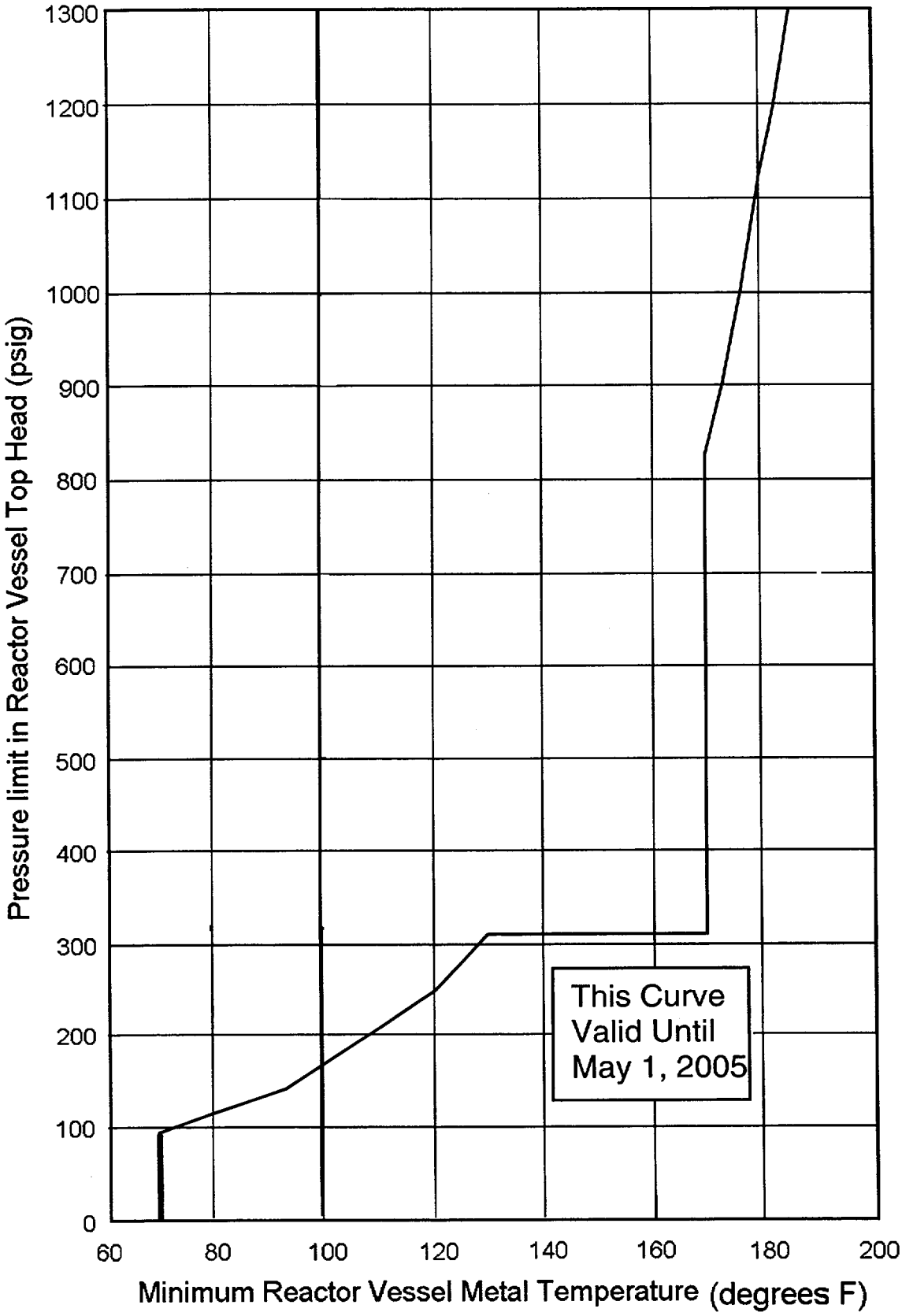


Figure 3.4.10-3
Nuclear (Core Critical) Limit (Curve C)

Attachment 5 to PLA-5341

Code Case N-640 Exemption Request

REQUEST FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50 APPENDIX G

In accordance with 10 CFR 50.12, PPL Susquehanna, LLC (PPL) is requesting an exemption from the requirements of 10 CFR 50.60(a) for Susquehanna Steam Electric Station Units 1 and 2. The exemption permits the use of ASME Code, Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1" in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b.

Justification for Use of Code Case N-640
(Code Case N-640 is provided at the end of this request)

10 CFR 50.12(a) Requirements

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME Code, Section XI, Appendix G, to determine the pressure-temperature (P-T) limits meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR50 provided:

1. *The requested exemption is authorized by law*: No law precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when the NRC grants an exemption under 10 CFR 50.12.

2. *The requested exemption does not present an undue risk to the public health and safety*: The proposed revision to the P-T limits rely, in part, on the requested exemption. The revised P-T limits were developed using the K_{IC} fracture toughness curve shown in ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME XI, Appendix G process of determining P-T limit curves remain unchanged.

Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor vessel. Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel (RPV) material fracture toughness.

Since 1974, additional knowledge about the fracture toughness of vessel materials and their fracture response to applied loads has been gained. The additional knowledge demonstrates the lower bound fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect against potential RPV failure. The lower bound K_{IC} fracture toughness provides an adequate margin of safety to protect against potential RPV failure and does not present an undue risk to public health and safety.

P-T curves based upon the K_{IC} toughness limits will enhance overall plant safety by opening the P-T operating window, especially in the region of low-temperature operations. The two primary benefits occurring during the pressure test are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

3. *The requested exemption will not endanger the common defense and security*: The common defense and security are not endangered by this exemption request.

4. *Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60*: In accordance with 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of the following paragraphs:

   - (2)(ii) - Demonstrates the underlying purpose of the regulation will continue to be achieved.

   - (2)(iii) - Will result in undue hardship or other costs that are significant if the regulation is enforced.

   - (2)(v) - Will provide only temporary relief from the applicable regulation and the licensee has made good-faith efforts to comply with the regulations.

   10 CFR 50.12(a)(2)(ii): ASME Code, Section XI, Appendix G provides procedures for determining allowable loading on the RPV and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfies the underlying requirement that:

1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; And

2) P-T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The ASME Code, Section XI, Appendix G procedure was conservatively developed based upon the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge concerning these topics has greatly expanded. This increased knowledge permits relaxation of the ASME Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety.

10 CFR 50.12(a)(2)(iii): The reactor coolant system (RCS) pressure-temperature operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME Code, Section XI, Appendix G procedure. Continued operation of Susquehanna Units 1 and 2 with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P-T operating window. This restriction requires the Operations staff to maintain a high temperature during pressure tests and also subjects inspection personnel to increased safety hazards while conducting inspections of systems with the potential for steam leaks in a primary containment at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P-T curves. Implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety.

10 CFR 50.12(a)(2)(v): The exemption provides only temporary relief from the applicable regulation; therefore, PPL requests that the exemption be granted until the NRC generically approves ASME Code Case N-640 for use by the nuclear industry.

Code Case N-640, Conclusion for Exemption Acceptability

Compliance with the specified requirement of 10 CFR 50.60(a) will result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the lower bound fracture toughness used by ASME Code, Section XI, Appendix G in the determination of reactor coolant P-T limits. This proposed alternative is acceptable, because the ASME Code case maintains the relative margin of safety commensurate with the margin of safety that existed at the time ASME Code, Section XI, Appendix G was approved in 1974. Therefore, application of ASME Code Case N-640 for Susquehanna Units 1 and 2 ensures an acceptable margin of safety and does not present an undue risk to the public health and safety.

Attachment 6 to PLA-5341

Technical report SIR-00-167, Rev.0 prepared for PPL by Structural Integrity Associates (SIA).

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TWK-01-002
SIR-00-167, Rev. 0

Mr. L. E. Willertz
PPL Susquehanna, LLC
Susquehanna Nuclear Power Station
Two North Ninth Street
Allentown, PA 18101-1179

Subject: Revised Pressure-Temperature Curves for Susquehanna Units 1 and 2

Reference: PPL Purchase Order No. 163739-C dated 12/11/2000.

Dear Mr. Willertz:

The attachment to this letter documents the revised set of pressure-temperature (P-T) curves developed for Susquehanna Steam Electric Station Units 1 and 2 (SSES-1 and SSES-2), in accordance with Structural Integrity'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 SSES-1 and SSES-2 for 32 effective full power years (EFPYs) and a power increase from 3293 MWT to 3441 MWT. The curves were developed in accordance with U.S. 10CFR 50 Appendix G, ASME Code Case N-640, and the 1995 Edition (Summer 1996 Addenda) of ASME Code, Section XI, Appendix G.

The inputs, methodology, and results for this effort are summarized in the attachment. The detailed calculations for this work (PPL-21Q-301 and PPL-21Q-302) are also attached.

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

Prepared By: Todd W. Keys
Todd W. Keys
Engineer

Reviewed By: Gary L. Stevens
Gary L. Stevens, P. E.
Technical Director

Approved By: Todd W. Keys
Todd W. Keys
Engineer

Attachment
cc: PPL-21Q-106 (ODL), -401

ATTACHMENT

REVISED P-T CURVES FOR SUSQUEHANNA UNITS 1 & 2

1.0 Introduction

This attachment documents the revised set of pressure-temperature (P-T) curves developed for the Susquehanna Steam Electric Station Units 1 and 2 (SSES-1 and SSES-2), in accordance with Structural Integrity's Quality Assurance Program. This work includes a full set of updated P-T curves (i.e., pressure test, core not critical, and core critical conditions) for SSES-1 and SSES-2 for 32 effective full power years (EFPYs) and a power increase from 3293 MWT to 3441 MWT. The curves were developed using the methodology specified in ASME Code Case N-640 [2], as well as 10CFR50 Appendix G [4], WRC-175 [5], 1995 Edition with Summer 1996 Addenda of ASME Code, Section XI, Appendix G [3]. The improvements realized from the Code Case methodology is as much as 60°F, and is primarily obtained from using the reference fracture toughness, K_{Ic}, in accordance with Code Case N-640.

2.0 RT_{NDT} Values

Adjusted reference temperature (ART_{NDT}) values were developed for the SSES-1 and SSES-2 reactor pressure vessel (RPV) materials in the Reference [1] and [9] reports. The values for a power increase from 3293 MWT to 3441 MWT were used in the current analysis.

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 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 method of generating the curves is primarily the same for each region for both Curves A and B. The exception is the method used to create the upper vessel/feedwater region Curve B. That method will be described separately.

The approach used for the Curve A beltline, bottom head, and upper vessel/feedwater nozzle regions, and the Curve B beltline and bottom head regions, includes the following steps:

a. Assume a fluid temperature, T. The temperature at the assumed flaw tip, $T_{1/4t}$ (i.e., 1/4t into the vessel wall) is determined by adding a temperature drop term, $\Delta T_{1/4t}$, to T. For the SSES evaluation, the temperature drop term was conservatively set to zero.

b. Calculate the reference fracture toughness, K_{Ic}, based on $T_{1/4t}$ using the relationship from Code Case N-640 [2], as follows:

$$K_{Ic} = 20.734\, e^{[0.02(T_{1/4t} - ART_{NDT})]} + 33.2 \quad [6, \text{A-4200}]$$

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}	=	reference fracture toughness (ksi√inch)

c. Calculate the thermal stress intensity factor, K_{It} from ASME Code, Section XI, Appendix G [3].

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}. For the bottom head region, a stress concentration factor of 3 is included to account for the bottom head penetrations, consistent with WRC-175 methodology [5] and other BWR P-T curve evaluations.

f. Apply any adjustments for temperature and/or pressure to T and P, respectively.

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

The approach used for the Curve B upper vessel/feedwater nozzle region includes the following steps:

a. Assume a pressure, P.

b. Calculate the thermal stress intensity factor, K_{It}, by combining the secondary membrane stress intensity factor, K_{Im} and the secondary bending stress intensity factor, K_{Ib}, per [3, G-2222] and including the correction factor, R, from Reference [5].

$$K_{Ip} = R(K_{Im} + K_{Ib})$$

where: R = correction factor, calculated to consider the nonlinear effects in the plastic region based on the assumptions and recommendations of WRC Bulletin 175 [5].

K_{Im} = secondary membrane stress intensity factor
= $M_m * \sigma_{sm}$

K_{Ib} = secondary bending stress intensity factor
= $(2/3) M_m * \sigma_{sb}$

The stress reports for the SSES feedwater nozzles did not provide sufficient detail for secondary stresses in the nozzle forging area. Therefore, the secondary stresses used in calculating the secondary membrane and bending stress intensity factors are those obtained from "generic" General Electric (GE) boiling water reactor P-T curve calculations (used to develop previous SSES P-T curves).

c. Calculate the allowable pressure stress intensity factor, K_{Ip}, based on the assumed P using the following relationship:

$$K_{Ip} = F(a/r_n)\,(\sigma_{pm} + R\sigma_{pb})\sqrt{(\pi a)}$$

where: $F(a/r_n)$ = nozzle stress factor, from Figure A5-1 of [5]
σ_{pm} = primary membrane stress
R = correction factor, defined above
σ_{pb} = primary bending stress
a = 1/4t crack depth for nozzle corner (inches)

d. Calculate the reference fracture toughness, K_{Ic}, using the following relationship:

$$K_{Ic} = K_{It} + K_{Ip} * SF$$

where: SF = safety factor = 2.0

e. Compute the fluid temperature, T (assumed equal to the 1/4t flaw temperature, $T_{1/4t}$), from the critical stress intensity factor K_{IC}, per the Code Case N-640 [2]. The K_{IC} equation [6] is manipulated to solve for $T_{1/4t}$ as follows:

$$T = 50 * \ln[(K_{IC} - 33.2)/20.734] + ART_{NDT}$$

f. Apply any adjustments for temperature and/or pressure to T and P, respectively.

g. Repeat steps (a) through (f) for other temperatures 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 [4]:

For Pressure Test Conditions (Curve A):

- If the pressure is greater than 20% of the pre-service hydro test pressure (1,563 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. This limit has been a standard recommendation for the BWR industry for non-ductile failure protection.

For Core Not Critical Conditions (Curve B):

- If the pressure is greater than 20% of the pre-service hydro test pressure, 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. This limit has been a standard recommendation for the BWR industry for non-ductile failure protection.

For Core Critical Conditions (Curve C):

- Per the requirements of Table 1 of Reference [4], 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 [4] (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 [4], 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.

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 1 through 14. The resulting P-T curves are shown in Figures 1 through 6.

5.0 References

1. GE Report No. GE-NE-523-169-1292, DRF B13-01666, "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," March 1993, SI File No. PPL-21Q-201.

2. 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.

3. 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," 1995 Edition with Summer 1996 Addenda.

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

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

6. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix A, "Analysis of Flaws," 1995 Edition with Summer 1996 Addenda.

7. Structural Integrity Associates Calculation No. PPL-21Q-301, Revision 0, "Development of Pressure Test (Curve A) P-T Curves," 02/20/01.

8. Structural Integrity Associates Calculation No. PPL-21Q-302, Revision 0, "Development of Heatup/Cooldown (Curves B & C) P-T Curves," 02/20/01.

9. GE Report No. GE-NE-523-107-0893, DRF 137-0010-6, Revision 1 "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," October 1993, SI File No. PPL-21Q-202.

Table 1
Tabulated Values for Unit 1 Beltline Pressure Test Curve (Curve A)

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

Inputs:

Plant =	**Susquehanna**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1875**	inches, so $\sqrt{t}$ =	2.487	$\sqrt{\text{inch}}$
Vessel Radius, R =	**126.6875**	inches		
ART_{NDT} =	**61.4**	°F ======>	**32 EFPY**	
K_R =	**0.0**	ksi*inch$^{1/2}$		
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)		
Safety Factor =	**1.5**	(for pressure test)		
M_m =	**2.303**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**30**	psig (hydrostatic pressure for a full vessel)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	57.83	38.55	817	70	787
75	75	60.42	40.28	854	75	824
80	80	63.28	42.18	894	80	864
85	85	66.44	44.29	939	85	909
90	90	69.94	46.62	989	90	959
95	95	73.80	49.20	1043	95	1,013
100	100	78.07	52.05	1104	100	1,074
105	105	82.79	55.19	1170	105	1,140
110	110	88.00	58.67	1244	110	1,214
115	115	93.77	62.51	1325	115	1,295
120	120	100.14	66.76	1416	120	1,386

Table 2
Tabulated Values for Unit 2 Beltline Pressure Test Curve (Curve A)

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

Inputs:

Plant =	**Susquehanna**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1875**	inches, so $\sqrt{t}$ =	2.487	$\sqrt{inch}$
Vessel Radius, R =	**126.6875**	inches		
ART_{NDT} =	**46.7**	°F ======>	**32 EFPY**	
K_{It} =	**0.0**	$ksi{\cdot}inch^{1/2}$		
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)		
Safety Factor =	**1.5**	(for pressure test)		
M_m =	**2.303**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**30**	psig (hydrostatic pressure for a full vessel)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc ($ksi{\cdot}inch^{1/2}$)	Klp ($ksi{\cdot}inch^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	66.24	44.16	936	70	906
75	75	69.72	46.48	986	75	956
80	80	73.56	49.04	1040	80	1,010
85	85	77.80	51.87	1100	85	1,070
90	90	82.49	55.00	1166	90	1,136
95	95	87.68	58.45	1239	95	1,209
100	100	93.41	62.27	1320	100	1,290
105	105	99.74	66.49	1410	105	1,380

Table 3
Tabulated Values for Unit 1 Feedwater Nozzle/Upper Vessel Region Pressure Test Curve (Curve A)

Inputs:				
Plant =	**Susquehanna**			
Component =	**Upper Vessel**	(based on FW nozzle)		
ART_{NDT} =	**40.0**	°F =====>	**All EFPYs**	
Vessel thickness, t =	**6.5**	inches, so √t	2.55	√inch
Vessel Radius, R =	**126.7**	inches		
F(a/m) =	**1.6**	nozzle stress factor		
Crack Depth, a =	**1.63**	inches		
Safety Factor =	**1.5**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**0.0**	psig		
Unit Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	KIc (ksi*inch$^{1/2}$)	KIp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
-	-	-	-	-	70	312.5
-	-	-	-	-	100	312.5
0	0	42.52	28.34	402	100	402
10	10	44.58	29.72	421	100	421
20	20	47.10	31.40	445	100	445
30	30	50.18	33.45	474	100	474
40	40	53.93	35.96	509	100	509
50	50	58.52	39.02	553	100	553
60	60	64.13	42.75	606	100	606
70	70	70.98	47.32	670	100	670
80	80	79.34	52.90	750	100	750
90	90	89.56	59.71	846	100	846
100	100	102.04	68.03	964	100	964
110	110	117.28	78.19	1108	110	1108
120	120	135.90	90.60	1284	120	1284
130	130	158.63	105.76	1498	130	1498

Table 4
Tabulated Values for Unit 2 Feedwater Nozzle/Upper Vessel Region Pressure Test Curve (Curve A)

Inputs:				
Plant =	**Susquehanna**			
Component =	**Upper Vessel**	(based on FW nozzle)		
ART_{NDT} =	**30.0**	°F =====>	**All EFPYs**	
Vessel thickness, t =	**6.5**	inches, so √t	2.55	√inch
Vessel Radius, R =	**126.7**	inches		
F(a/m) =	**1.6**	nozzle stress factor		
Crack Depth, a =	**1.63**	inches		
Safety Factor =	**1.5**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**0.0**	psig		
Unit Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	KIc (ksi*inch$^{1/2}$)	KIp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
-	-	-	-	-	70	312.5
-	-	-	-	-	100	312.5
0	0	44.58	29.72	421	100	421
10	10	47.10	31.40	445	100	445
20	20	50.18	33.45	474	100	474
30	30	53.93	35.96	509	100	509
40	40	58.52	39.02	553	100	553
50	50	64.13	42.75	606	100	606
60	60	70.98	47.32	670	100	670
70	70	79.34	52.90	750	100	750
80	80	89.56	59.71	846	100	846
90	90	102.04	68.03	964	100	964
100	100	117.28	78.19	1108	100	1108
110	110	135.90	90.60	1284	110	1284
120	120	158.63	105.76	1498	120	1498

Table 5
Tabulated Values for Unit 1 Bottom Head Pressure Test Curve (Curve A)

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

Inputs:

Plant =	**Susquehanna**		
Component =	**Bottom Head**		
Vessel thickness, t =	**6.1875**	inches, so √t :	2.487 √inch
Vessel Radius, R =	**126.6875**	inches	
ART_{NDT} =	**34.0**	°F ======>	**32 EFPY**
K_R =	**0.0**	ksi*inch$^{1/2}$	
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)	
Safety Factor =	**1.5**	(for pressure test)	
Stress Concentration Factor =	**3.0**	Bottom head penetrations	
M_m =	**2.303**		
Temperature Adjustment =	**0.0**	°F	
Height of Water for a Full Vessel =	**882.0**	inches	
Pressure Adjustment =	**31.85**	psig (hydrostatic pressure at bottom head for a full vessel at 70°F)	
Hydro Test Pressure =	**1,563**	psig	
Flange RT_{NDT} =	**10.0**	°F	

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	75.80	50.53	714	70	682
75	75	80.28	53.52	757	75	725
80	80	85.23	56.82	803	80	771
85	85	90.70	60.47	855	85	823
90	90	96.75	64.50	912	90	880
95	95	103.43	68.95	975	95	943
100	100	110.82	73.88	1044	100	1,012
105	105	118.98	79.32	1121	105	1,089
110	110	128.00	85.33	1206	110	1,174
115	115	137.97	91.98	1300	115	1,268
120	120	148.99	99.33	1404	120	1,372

Table 6
Tabulated Values for Unit 2 Bottom Head Pressure Test Curve (Curve A)

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

Inputs:

Plant =	**Susquehanna**			
Component =	**Bottom Head**			
Vessel thickness, t =	**6.1875**	inches, so √t =	2.487	√inch
Vessel Radius, R =	**126.6875**	inches		
ART_{NDT} =	**24.0**	°F ======>	**32 EFPY**	
K_{It} =	**0.0**	ksi*inch$^{1/2}$		
$\Delta T_{1/4t}$ =	**0.0**	°F (no thermal for pressure test)		
Safety Factor =	**1.5**	(for pressure test)		
Stress Concentration Factor =	**3.0**	Bottom head penetrations		
M_m =	**2.303**			
Temperature Adjustment =	**0.0**	°F		
Height of Water for a Full Vessel =	**882.0**	inches		
Pressure Adjustment =	**31.85**	psig (hydrostatic pressure at bottom head for a full vessel at 70°F)		
Hydro Test Pressure =	**1,563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	85.23	56.82	803	70	771
75	75	90.70	60.47	855	75	823
80	80	96.75	64.50	912	80	880
85	85	103.43	68.95	975	85	943
90	90	110.82	73.88	1044	90	1,012
95	95	118.98	79.32	1121	95	1,089
100	100	128.00	85.33	1206	100	1,174
105	105	137.97	91.98	1300	105	1,268
110	110	148.99	99.33	1404	110	1,372

Table 7
Tabulated Values for Unit 1 Beltline Core Not Critical Curve (Curve B)

Inputs:

Plant =	**Susquehanna**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1875**	inches, so √t	2.487	√inch
Vessel Radius, R =	**126.6875**	inches		
ART_{NDT} =	**61.4**	°F ======>	**32 EFPY**	
Cooldown Rate =	**100.0**	°F/hr		
K_{It} =	**9.08**	ksi*inch$^{1/2}$		
Safety Factor =	**2.0**			
M_m =	**2.303**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure for a full vessel)		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	57.83	24.37	517	70	487
75	75.0	60.42	25.67	544	75	514
80	80.0	63.28	27.10	575	80	545
85	85.0	66.44	28.68	608	85	578
90	90.0	69.94	30.43	645	90	615
95	95.0	73.80	32.36	686	95	656
100	100.0	78.07	34.50	731	100	701
105	105.0	82.79	36.86	782	105	752
110	110.0	88.00	39.46	837	110	807
115	115.0	93.77	42.35	898	115	868
120	120.0	100.14	45.53	965	120	935
125	125.0	107.18	49.05	1040	125	1010
130	130.0	114.96	52.94	1123	130	1093
135	135.0	123.56	57.24	1214	135	1184
140	140.0	133.06	61.99	1314	140	1284
145	145.0	143.56	67.24	1426	145	1396

Table 8
Tabulated Values for Unit 2 Beltline Core Not Critical Curve (Curve B)

Inputs:

Plant =	**Susquehanna**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1875**	inches, so √t	2.487	√inch
Vessel Radius, R =	**126.6875**	inches		
ART_{NDT} =	**46.7**	°F ======>	**32 EFPY**	
Cooldown Rate =	**100.0**	°F/hr		
K_{It} =	**9.08**	ksi*inch$^{1/2}$		
Safety Factor =	**2.0**			
M_m =	**2.303**			
Temperature Adjustment =	**0.0**	°F		
Pressure Adjustment =	**30.0**	psig (hydrostatic pressure for a full vessel)		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	KIc (ksi*inch$^{1/2}$)	KIp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	66.24	28.58	606	70	576
75	75.0	69.72	30.32	643	75	613
80	80.0	73.56	32.24	684	80	654
85	85.0	77.80	34.36	729	85	699
90	90.0	82.49	36.71	778	90	748
95	95.0	87.68	39.30	833	95	803
100	100.0	93.41	42.17	894	100	864
105	105.0	99.74	45.33	961	105	931
110	110.0	106.74	48.83	1035	110	1005
115	115.0	114.47	52.70	1117	115	1087
120	120.0	123.02	56.97	1208	120	1178
125	125.0	132.46	61.69	1308	125	1278
130	130.0	142.90	66.91	1419	130	1389

Table 9
Tabulated Values for Unit 1 Upper Vessel/Feedwater Nozzle Region Core Not Critical Curve (Curve B)

Inputs:					
Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	40.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp	
σ_{am} =	16.19	ksi @	546 °F		90 °F
σ_{ab} =	19.04	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.54				
Safety Factor =	2.0				
$F(a/r_a)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{am} (ksi)	σ_{ab} (ksi)	σ_{total} (ksi)	R	KIt (ksi*inch$^{1/2}$)	KIp (ksi*inch$^{1/2}$)	Total KIc (ksi*inch$^{1/2}$)	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	0.0	70.0	50
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	0.0	70.0	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	64.9	70.0	150
165.9	373.4	3.24	0.03	10.06	11.83	25.16	1.00	45.5	12.7	71.0	70.0	70.0	166
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	79.1	79.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	89.6	89.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	98.0	98.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	99.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	130.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	105.0	130.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	111.0	130.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	116.2	130.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	120.9	130.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	125.1	130.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	129.0	130.0	600
614	491.6	11.98	0.13	14.26	16.77	43.13	1.00	64.5	47.1	158.7	130.0	130.0	614
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	132.5	132.5	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	135.2	135.2	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	137.2	137.2	750
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	139.1	139.1	800
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	140.9	140.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	142.7	142.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	144.4	144.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	146.1	146.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	147.7	147.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	149.3	149.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	150.8	150.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	152.2	152.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	153.7	153.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	155.0	155.0	1300

Table 10
Tabulated Values for Unit 2 Upper Vessel/Feedwater Nozzle Region Core Not Critical Curve (Curve B)

Inputs:				
Plant =	Susquehanna			
Component =	Upper Vessel			
ART_{NDT} =	30.0	°F		
σ_{pm} =	20.49	ksi @	1050 psig	
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp
σ_{tm} =	16.19	ksi @	546 °F	90 °F
σ_{tb} =	19.04	ksi @	546 °F	90 °F
σ_{ys} =	45.0	ksi		
M_m =	2.54			
Safety Factor =	2.0			
$F(a/r_n)$ =	1.6			
Temperature Adjustment =	0.0	°F		
Pressure Adjustment =	0.0	psig		
Hydro Test Pressure =	1563	psig		
Flange RT_{NDT} =	10.0	°F		

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{tm} (ksi)	σ_{tb} (ksi)	σ_{total} (ksi)	R	KIt (ksi•inch$^{1/2}$)	KIp (ksi•inch$^{1/2}$)	Total KIc (ksi•inch$^{1/2}$)	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	0.0	70.0	50
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	0.0	70.0	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	54.9	70.0	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	69.1	70.0	200
203.7	389.4	3.98	0.04	10.63	12.50	27.15	1.00	48.1	15.6	79.3	70.0	70.0	204
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	79.6	79.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	88.0	88.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	130.0	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	130.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	95.0	130.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	101.0	130.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	106.2	130.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	110.9	130.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	115.1	130.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	119.0	130.0	600
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	122.5	130.0	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	125.2	130.0	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	127.2	130.0	750
757	514.3	14.77	0.16	15.06	17.71	47.71	0.92	62.7	58.0	178.7	127.4	130.0	757
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	129.1	130.0	800
825	523.9	16.10	0.17	15.41	18.12	49.79	0.86	60.1	63.2	186.5	130.0	130.0	825
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	130.9	130.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	132.7	132.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	134.4	134.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	136.1	136.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	137.7	137.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	139.3	139.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	140.8	140.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	142.2	142.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	143.7	143.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	145.0	145.0	1300

Table 11
Tabulated Values for Unit 1 Bottom Head Core Not Critical Curve (Curve B)

Inputs:

Plant =	**Susquehanna**			
Component =	**Bottom Head**	(Penetrations Portion)		
Vessel thickness, t =	**6.1875**	inches, so √t	2.487	√inch
Vessel Radius, R =	**126.6875**	inches		
Cooldown Rate =	**100.0**	°F/hr		
Safety Factor =	**2.0**			
Stress Concentration Factor =	**3.0**			
ART_{NDT} =	**34.0**	°F		
M_m =	**2.303**			
K_{It} =	**9.08**	$ksi^{\circ}inch^{1/2}$		
Temperature Adjustment =	**0.00**	°F		
Height of full vessel =	**882.0**	inches		
Pressure Adjustment =	**31.85**	psig		
Unit Pressure =	**1563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc ($ksi^{\circ}inch^{1/2}$)	Klp ($ksi^{\circ}inch^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	75.80	33.36	472	70	440
75	75.0	80.28	35.60	503	75	471
80	80.0	85.23	38.08	538	80	506
85	85.0	90.70	40.81	577	85	545
90	90.0	96.75	43.84	620	90	588
95	95.0	103.43	47.18	667	95	635
100	100.0	110.82	50.87	719	100	687
105	105.0	118.98	54.95	777	105	745
110	110.0	128.00	59.46	841	110	809
115	115.0	137.97	64.45	911	115	879
120	120.0	148.99	69.96	989	120	957
125	125.0	161.17	76.05	1075	125	1043
130	130.0	174.63	82.78	1170	130	1138
135	135.0	189.50	90.21	1275	135	1243
140	140.0	205.94	98.43	1391	140	1360

Table 12
Tabulated Values for Unit 2 Bottom Head Core Not Critical Curve (Curve B)

Inputs:

Plant =	**Susquehanna**			
Component =	**Bottom Head**	(Penetrations Portion)		
Vessel thickness, t =	**6.1875**	inches, so √t	2.487	√inch
Vessel Radius, R =	**126.6875**	inches		
Cooldown Rate =	**100.0**	°F/hr		
Safety Factor =	**2.0**			
Stress Concentration Factor =	**3.0**			
ART_{NDT} =	**24.0**	°F		
M_m =	**2.303**			
K_{It} =	**9.08**	ksi*inch$^{1/2}$		
Temperature Adjustment =	**0.00**	°F		
Height of full vessel =	**882.0**	inches		
Pressure Adjustment =	**31.85**	psig		
Unit Pressure =	**1563**	psig		
Flange RT_{NDT} =	**10.0**	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	85.23	38.08	538	70	506
75	75.0	90.70	40.81	577	75	545
80	80.0	96.75	43.84	620	80	588
85	85.0	103.43	47.18	667	85	635
90	90.0	110.82	50.87	719	90	687
95	95.0	118.98	54.95	777	95	745
100	100.0	128.00	59.46	841	100	809
105	105.0	137.97	64.45	911	105	879
110	110.0	148.99	69.96	989	110	957
115	115.0	161.17	76.05	1075	115	1043
120	120.0	174.63	82.78	1170	120	1138
125	125.0	189.50	90.21	1275	125	1243
130	130.0	205.94	98.43	1391	130	1360

Table 13
Tabulated Values for Unit 1 Core Critical Curve (Curve C)

Inputs:					
Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	40.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp	
σ_{sm} =	16.19	ksi @	546 °F		90 °F
σ_{sb} =	19.04	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.54				
Safety Factor =	2.0				
$F(a/r_n)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	KIt ksi*inch$^{1/2}$	KIp ksi*inch$^{1/2}$	Total KIc ksi*inch$^{1/2}$	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	-	70.0	50
81.4	324.7	1.59	0.02	8.33	9.80	19.73	1.00	37.7	6.2	50.2	30.0	70.0	81
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	42.8	82.8	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	64.9	104.9	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	79.1	119.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	89.6	129.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	98.0	138.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	139.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	170.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	105.0	170.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	111.0	170.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	116.2	170.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	120.9	170.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	125.1	170.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	129.0	170.0	600
614	491.6	11.98	0.13	14.26	16.77	43.13	1.00	64.5	47.1	158.7	130.0	170.0	614
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	132.5	172.5	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	135.2	175.2	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	137.2	177.2	750
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	139.1	179.1	800
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	140.9	180.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	142.7	182.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	144.4	184.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	146.1	186.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	147.7	187.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	149.3	189.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	150.8	190.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	152.2	192.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	153.7	193.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	155.0	195.0	1300

Table 14
Tabulated Values for Unit 2 Core Critical Curve (Curve C)

Inputs:				
Plant =	Susquehanna			
Component =	Upper Vessel			
ART_{NDT} =	30.0	°F		
σ_{pm} =	20.49	ksi @	1050 psig	
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp
σ_{sm} =	16.19	ksi @	546 °F	90 °F
σ_{sb} =	19.04	ksi @	546 °F	90 °F
σ_{ys} =	45.0	ksi		
M_m =	2.54			
Safety Factor =	2.0			
$F(a/r_n)$ =	1.6			
Temperature Adjustment =	0.0	°F		
Pressure Adjustment =	0.0	psig		
Hydro Test Pressure =	1563	psig		
Flange RT_{NDT} =	10.0	°F		

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	KIt ksi•inch$^{1/2}$	KIp ksi•inch$^{1/2}$	Total KIc ksi•inch$^{1/2}$	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	-	70.0	50
95.5	334.7	1.86	0.02	8.69	10.22	20.79	1.00	39.3	7.3	54.0	30.0	70.0	96
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	32.8	72.8	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	54.9	94.9	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	69.1	109.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	79.6	119.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	88.0	128.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	129.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	170.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	95.0	170.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	101.0	170.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	106.2	170.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	110.9	170.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	115.1	170.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	119.0	170.0	600
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	122.5	170.0	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	125.2	170.0	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	127.2	170.0	750
757	514.3	14.77	0.16	15.06	17.71	47.71	0.92	62.7	58.0	178.7	127.4	170.0	757
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	129.1	170.0	800
825	523.9	16.10	0.17	15.41	18.12	49.79	0.86	60.1	63.2	186.5	130.0	170.0	825
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	130.9	170.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	132.7	172.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	134.4	174.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	136.1	176.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	137.7	177.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	139.3	179.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	140.8	180.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	142.2	182.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	143.7	183.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	145.0	185.0	1300

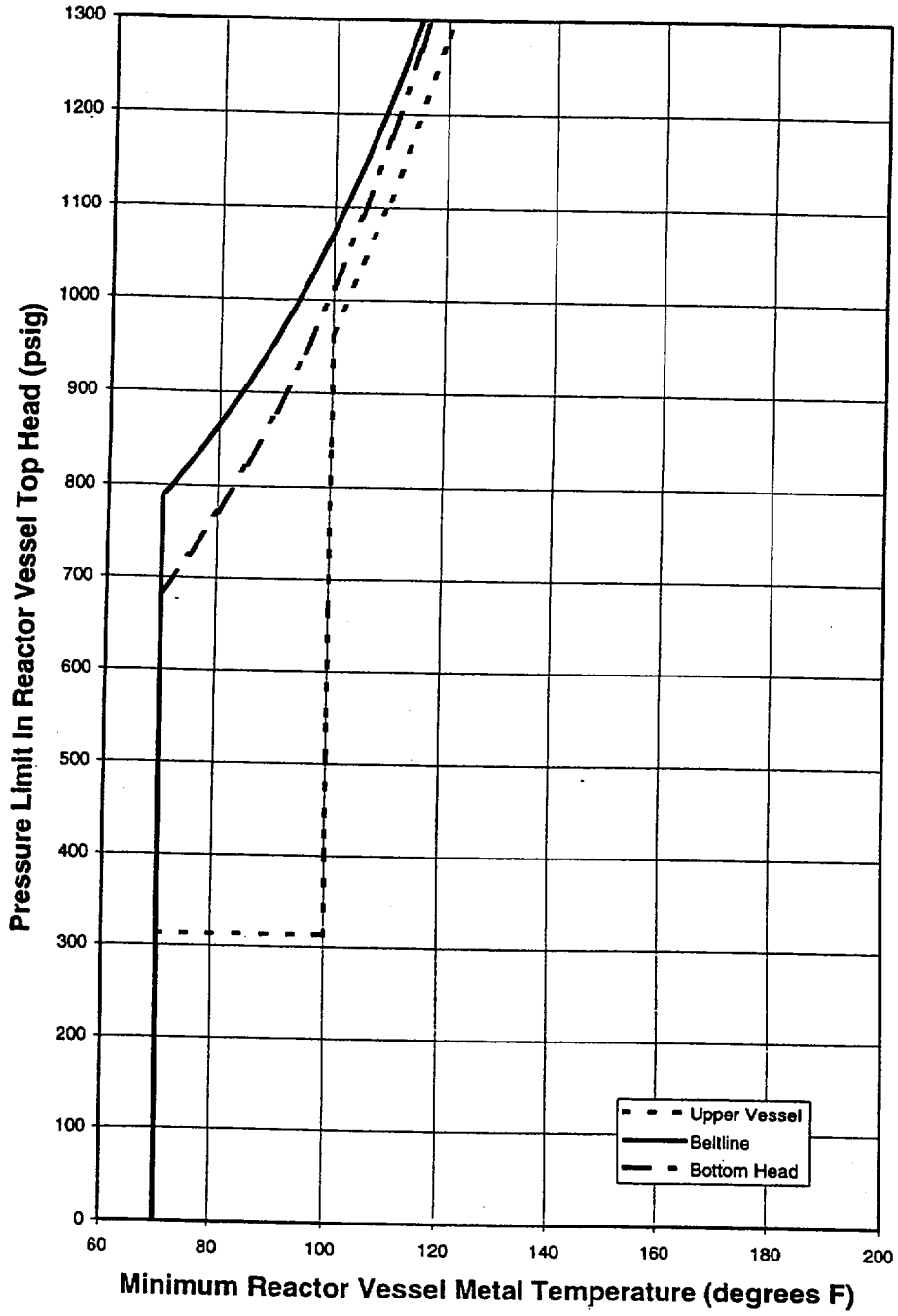

Figure 1
Pressure Test P-T Curve (Curve A) for Unit 1

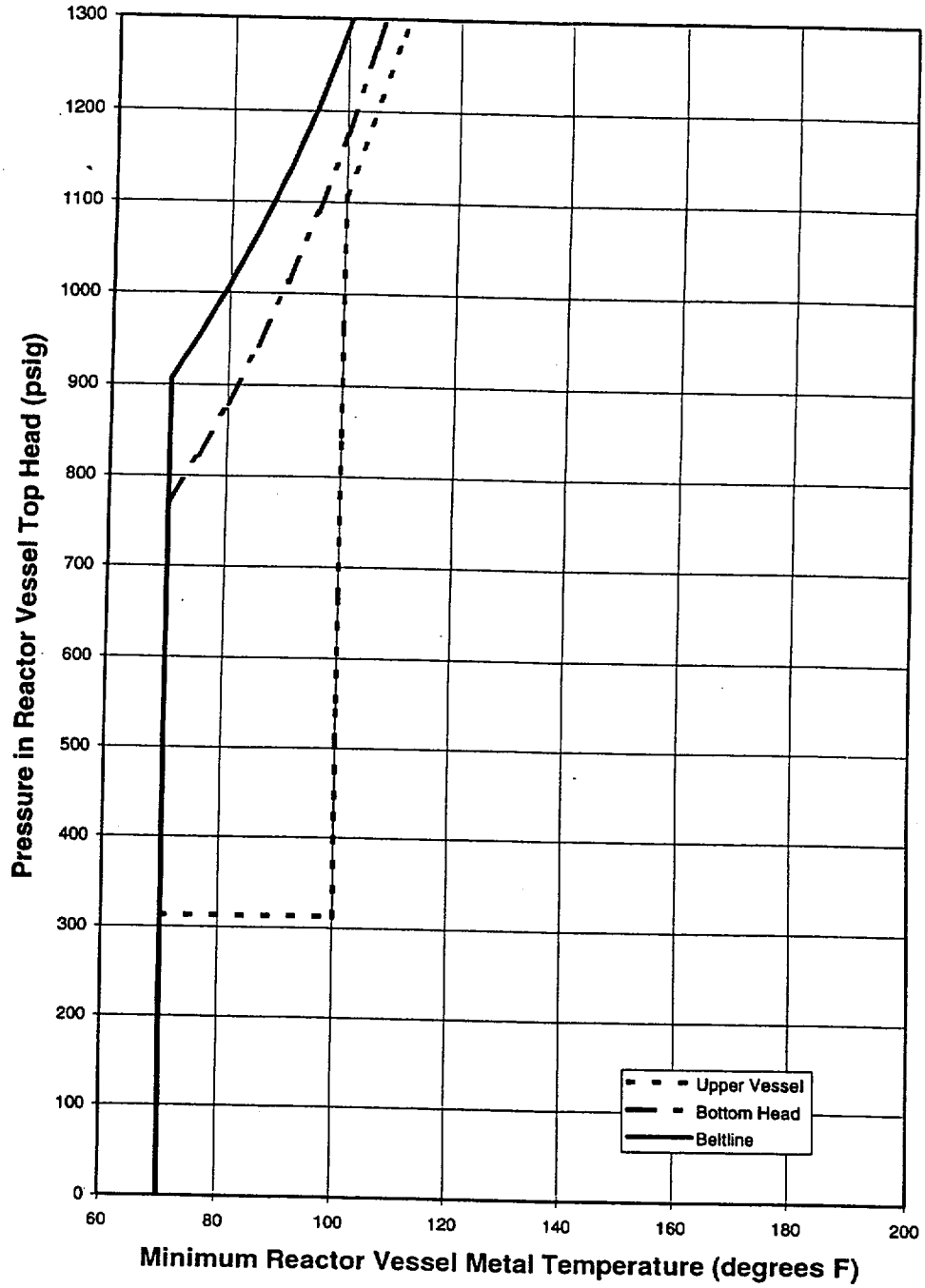

Figure 2
Pressure Test P-T Curve (Curve A) for Unit 2

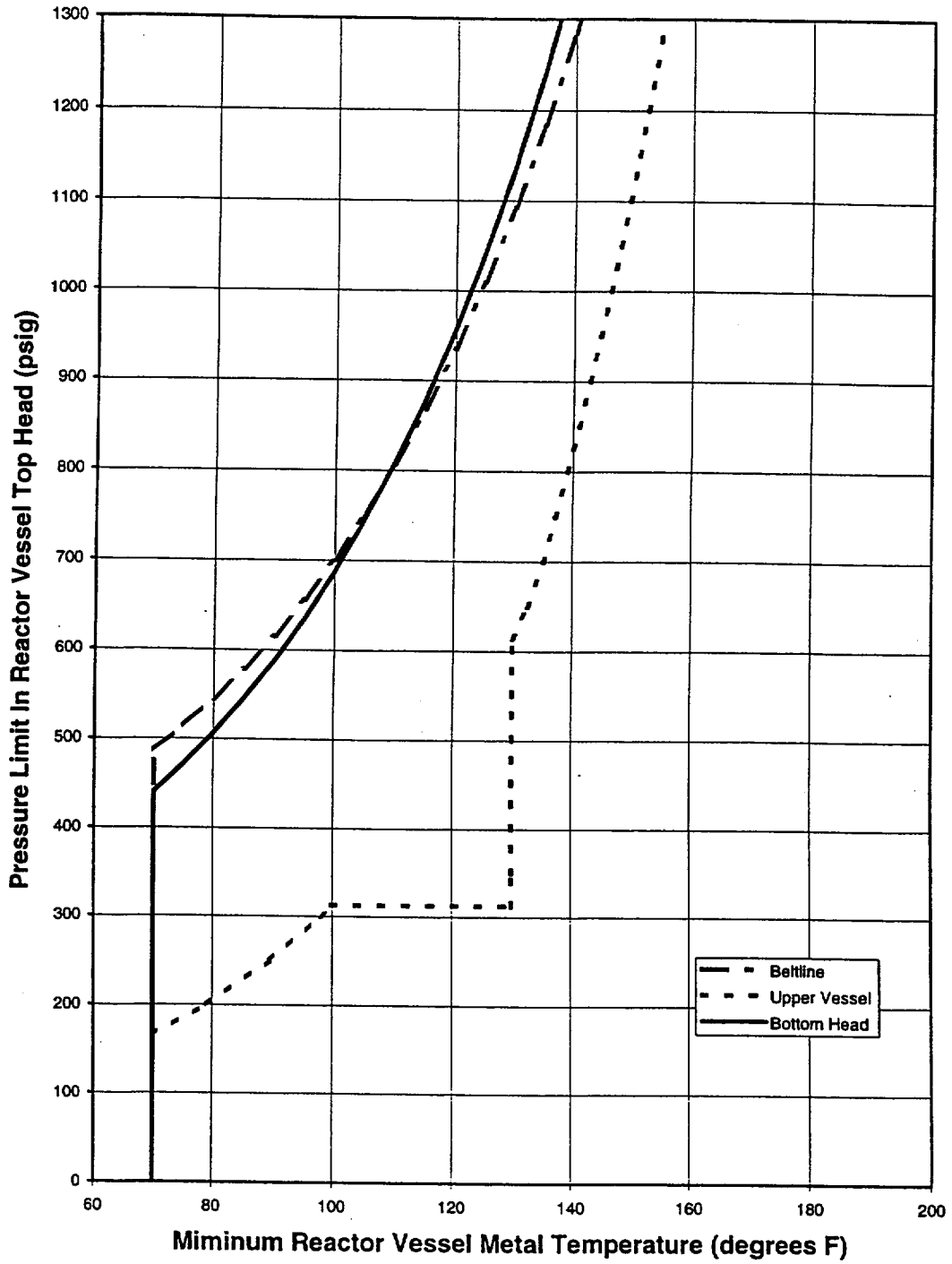

Figure 3
Core Not Critical Curve (Curve B) for Unit 1

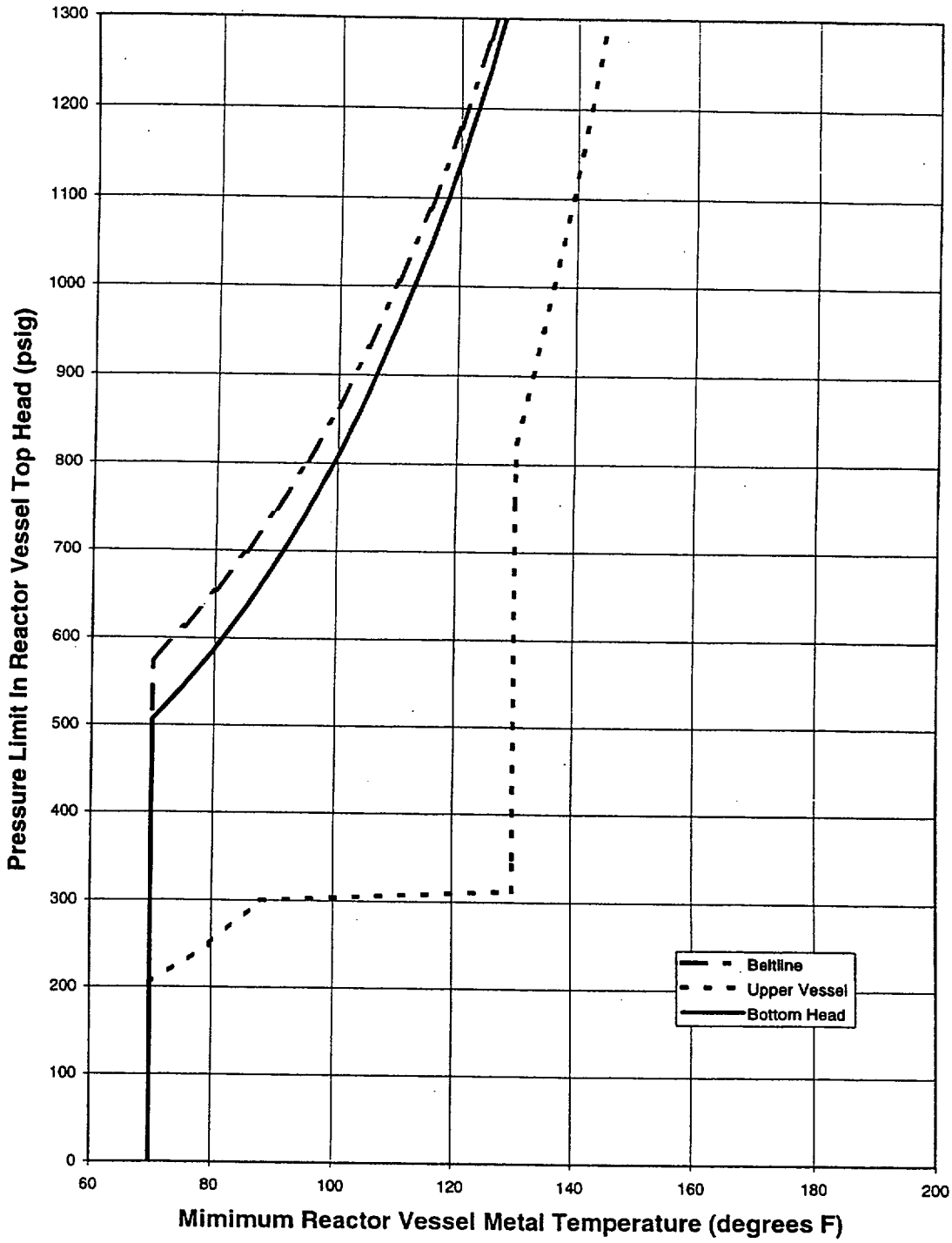

Figure 4
Core Not Critical Curve (Curve B) for Unit 2

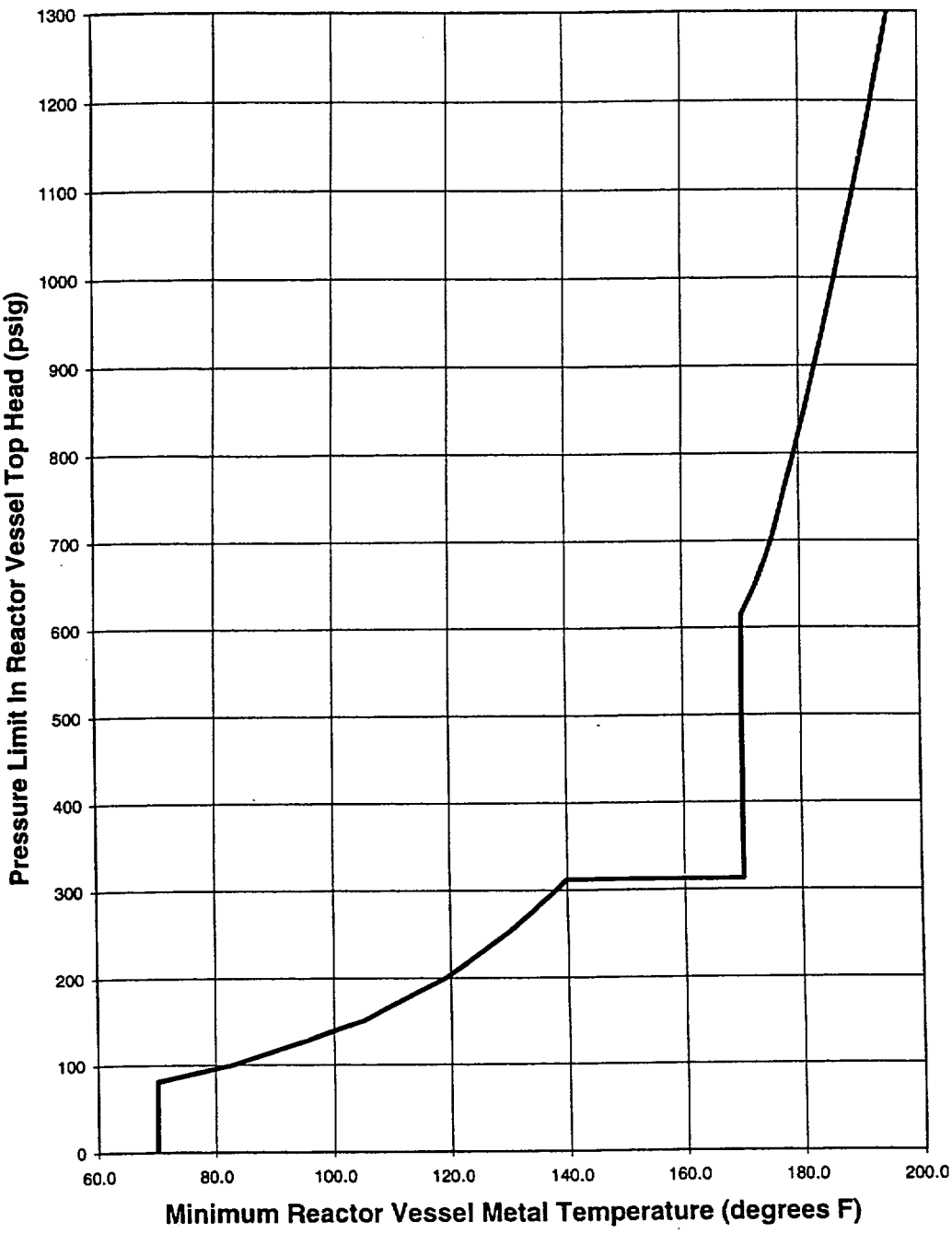

Figure 5
Core Critical Curve (Curve C) for Unit 1

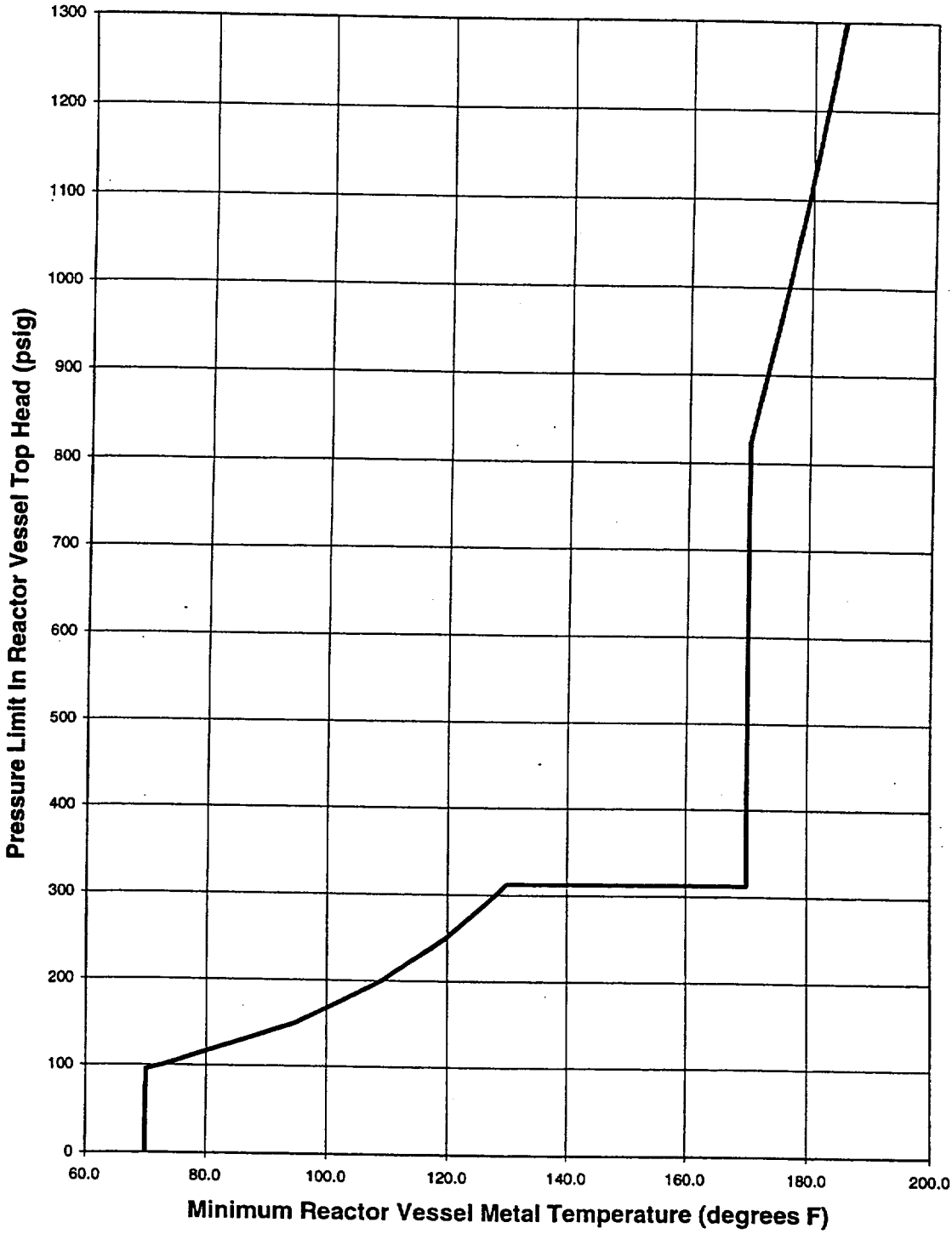

Figure 6
Core Critical Curve (Curve C) for Unit 2