Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon Nuclear

RS-10-080

10 CFR 50.90

April 19, 2010

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limit Curves

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," to incorporate revised P/T curves that are valid for up to 32 effective full power years of operation.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS pages.
- Attachment 3 provides a markup of the affected TS Bases page. The TS Bases page is provided for information only and does not require NRC approval.
- Attachments 4 and 5 provide General Electric Company (GE) proprietary reports that support the proposed change, for LSCS Units 1 and 2, respectively.
- Attachments 6 and 7 provide non-proprietary versions of the GE reports contained in Attachments 4 and 5, for LSCS Units 1 and 2, respectively.

Some of the information in Attachments 4 and 5 is proprietary to GE, and is supported by an affidavit signed by GE-Hitachi, the owner of the information. The affidavits, which are provided within the applicable documents, set forth the basis on which the information may be withheld from public disclosure by the NRC, and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

A001
NRR

Accordingly, it is respectfully requested that the information be withheld from public disclosure in accordance with 10 CFR 2.390. Non-proprietary versions of the information contained in Attachments 4 and 5 are provided in Attachments 6 and 7, respectively.

The proposed change has been reviewed by the LSCS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

The existing P/T curves contained in TS 3.4.11 are valid for up to 20 effective full power years of operation. Current projections indicate that LSCS Units 1 and 2 will reach 20 effective full power years of operation in April 2011 and December 2011, respectively. Therefore, EGC requests approval of the proposed change by April 19, 2011. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 19th day of April 2010.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Pages
3. Markup of Proposed Technical Specifications Bases Page
4. GE-NE-0000-0003-5526-02R1, "Pressure-Temperature Curves For Exelon LaSalle Unit 1," dated May 2004 (Proprietary)
5. GE-NE-0000-0003-5526-01R1, "Pressure-Temperature Curves For Exelon LaSalle Unit 2," dated May 2004 (Proprietary)
6. GE-NE-0000-0003-5526-02R1a, "Pressure-Temperature Curves For Exelon LaSalle Unit 1," dated May 2004 (Non-Proprietary)
7. GE-NE-0000-0003-5526-01R1a, "Pressure-Temperature Curves For Exelon LaSalle Unit 2," dated May 2004 (Non-Proprietary)

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

- 4.1 Applicable Regulatory Requirements/Criteria
- 4.2 No Significant Hazards Consideration
- 4.3 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

ATTACHMENT 1
Evaluation of Proposed Change

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," to incorporate revised P/T curves that are valid for up to 32 effective full power years (EFPY) of operation.

2.0 DETAILED DESCRIPTION

TS 3.4.11 contains Reactor Coolant System (RCS) P/T curves for heatup, cooldown, inservice leak and hydrostatic testing, and criticality, and also limits the maximum rate of change of RCS temperature. The existing P/T curves contained in TS 3.4.11 are valid for up to 20 EFPY of operation. Current projections indicate that LSCS Units 1 and 2 will reach 20 EFPY of operation in April 2011 and December 2011, respectively.

The proposed change modifies Surveillance Requirements (SRs) 3.4.11.1 and 3.4.11.2 to replace the existing references to the 20 EFPY curves with references to the new 32 EFPY curves. Specifically, the SRs are revised to read as follows.

SR 3.4.11.1

----------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 within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3 for Unit 1 up to 32 EFPY, and Figures 3.4.11-4, 3.4.11-5, and 3.4.11-6 for Unit 2 up to 32 EFPY;

b. RCS heatup and cooldown rates are $\leq$ 100°F in any 1 hour period; and

c. RCS temperature change during system leakage and hydrostatic testing is < 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 up to 32 EFPY and Figure 3.4.11-5 for Unit 2 up to 32 EFPY.

SR 3.4.11.2

Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.11-3 for Unit 1 up to 32 EFPY and Figure 3.4.11-6 for Unit 2 up to 32 EFPY.

The proposed change also replaces Figures 3.4.11-1 through 3.4.11-6 with revised figures that are applicable to 32 EFPY.

A markup of the proposed TS changes is provided in Attachment 2.

3.0 TECHNICAL EVALUATION

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 (i.e., heatup) and shutdown (i.e., cooldown) operations, power transients, and reactor trips. TS 3.4.11 limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

TS 3.4.11 contains P/T curves for heatup, cooldown, inservice leak and hydrostatic testing, and criticality and also limits the maximum rate of change of RCS temperature. Each P/T curve defines an acceptable region for normal operation. The typical use of the curves is during RCS 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 P/T curves establish operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB).

The P/T curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The existing P/T curves contained within TS 3.4.11 (i.e., Figures 3.4.11-1 through 3.4.11-6) are applicable up to 20 EFPY. The NRC approved the existing 20 EFPY P/T curves for LSCS in Reference 1, in response to EGC's license amendment request dated January 31, 2003, (i.e., Reference 2), as supplemented by References 3 and 4.

Reference 4 provided information, in response to an NRC request, regarding the chemical composition of vertical and girth welds, and the impact of these chemical compositions on the adjusted reference temperature and P/T limits. In addition, Reference 4 provided information to demonstrate that the Low Pressure Coolant Injection (LPCI) nozzle forging adjusted reference temperature bounds that for the LPCI nozzle weld for both units. The information in Reference 4 described above is applicable to the new 32 EFPY curves.

Updated P/T curves have been created to replace the existing P/T curves with new curves that are applicable up to 32 EFPY, which represents the end of the current 40-year operating licenses (i.e., assuming an 80 percent capacity factor). Attachments 4 and 5 provide General Electric Company (GE) proprietary reports that describe the analyses scope, assumptions, methodology, and results for LSCS Units 1 and 2, respectively. The new P/T curves were developed to present steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline.

Non-proprietary versions of the information contained in Attachments 4 and 5 are provided as Attachments 6 and 7, respectively.

GE recently became aware of an administrative error in the Unit 2 report provided in Attachment 5. Specifically, on Table F-1, "Upper Shelf Energy Evaluation for LaSalle Unit 2 Beltline Materials," weld heat 3P4000 was listed as being associated with a "Lower-Intermediate" weld, and weld heat 3P4966 was listed as being associated with a "Lower" weld. As listed in Table 4-4, "LaSalle Unit 2 Beltline ART Values (32 EFPY)," weld heat 3P4000 is actually associated with a "Lower" weld, and weld heat 3P3966 is associated with a "Lower-Intermediate" weld. This administrative error has no impact on the fracture toughness evaluation.

In Reference 1, the NRC indicated that when the 32 EFPY curves are submitted for NRC review and approval, EGC must perform a quantitative evaluation to demonstrate that the feedwater nozzles are more limiting than the Unit 2 LPCI nozzles, or provide P/T limit curves based on the adjusted reference temperature for the LPCI nozzles. As discussed in Attachment 5, for Unit 2, the LPCI nozzle is the limiting material for the beltline region for 32 EFPY. The beltline pressure test P/T curves were calculated in the same manner as the feedwater nozzle P/T curves as described in Section 4.3.2.1.3 of Attachment 5. The initial RT_{NDT} for the LPCI nozzle materials is -6°F. The generic pressure test P/T curve is applied to the Unit 2 feedwater nozzle curve by shifting the P vs. ($T - RT_{NDT}$) values in Section 4.3.2.1.3 of Attachment 5 to reflect the LPCI nozzle adjusted reference temperature value of 52°F. Therefore, the proposed 32 EFPY curves for Unit 2 are based on the adjusted reference temperature for the LPCI nozzles.

The new P/T curves for LSCS were developed using the methodology of GE Topical Report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," (i.e., Reference 5). In Reference 6, the NRC approved the NEDC-32983P methodology for pressure and core shroud fast neutron flux evaluation. However, the approval was subject to the following limitations.

(1) Within three years from the day of the approval of this methodology, GE Nuclear Energy will perform predictive calculations of at least four additional boiling water reactor surveillance capsule dosimetry measurements which will be submitted to the NRC before initiation of the measurements.

(2) Comparisons of the measurements and calculations will also be submitted to the NRC.

(3) Shroud fluence estimates will be limited to the beltline region, without bias adjustment.

(4) GE Nuclear Energy will perform dosimetry analysis to confirm and remove the conservatism in the shroud fluence calculations.

(5) Revisions to the fluence methodology and supporting uncertainty analysis will be provided, if the calculated/measured comparisons (i.e., for the additional analysis for the vessel and the shroud) are not consistent with the NEDC-32983P fluence methodology.

Subsequently, GE Nuclear Energy submitted additional information to the NRC to justify removing methodology limitations (1) through (4), listed above, associated with NEDC-32983P. In Reference 7, the NRC concluded that the information submitted by GE Nuclear Energy provided sufficient justification to remove limitations (1) through (4). However, limitation (5) remains as a condition of applicability of the NEDC-32983P methodology.

EGC has evaluated compliance with limitation (5) and has concluded that there are no new data obtained from the material surveillance program that impacts the NEDC-32983P fluence methodology. Both the current 20 EFPY curves and the proposed 32 EFPY curves are based on the fluence methodology and uncertainty analyses in NEDC-32983P. These analytical methods can be compared to measured values when surveillance capsules are removed from the reactor and analyzed in accordance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements." LSCS demonstrates its compliance with the Appendix H requirements through participation in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) (i.e., Reference 8). Reference 9 provides industry results of surveillance capsule testing.

The ISP representative surveillance materials for LSCS Unit 1 plate and weld metals are contained in the LSCS Unit 1 capsule. EGC removed this surveillance capsule during the spring 2010 refueling outage, and plans are to evaluate the materials in 2010. Therefore, no new materials data applicable to LSCS Unit 1 have been obtained since the approval of the 20 EFPY curves, and no comparison between calculated and measured values can be made.

The ISP representative surveillance materials for LSCS Unit 2 plate metals are contained in the River Bend capsules. The River Bend capsule is not scheduled for testing until 2025. The ISP representative surveillance materials for LSCS Unit 2 weld metals are contained in the Susquehanna Unit 1 capsules. The Susquehanna Unit 1 capsule is not scheduled for testing until 2012. Therefore, no new materials data applicable to LSCS Unit 2 have been obtained since the approval of the 20 EFPY curves, and no comparison between calculated and measured values can be made.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," paragraph (a) states:

> Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

The NRC has established requirements in Appendix G to 10 CFR 50, "Fracture Toughness Requirements," to protect the integrity of the RCPB in nuclear power plants. Appendix G to 10 CFR 50 requires that the P/T limits be at least as conservative as those obtained by following the methods of analysis and the margins of safety of

Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Appendix G also requires that applicable surveillance data from material surveillance programs be incorporated into the calculations of plant-specific P/T limits, and that the P/T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the reactor vessel beltline materials.

Table 1, "Pressure and Temperature Requirements for the Reactor Pressure Vessel," of Appendix G to 10 CFR 50 provides the NRC's criteria for meeting the P/T limit requirements of ASME Code, Section XI, Appendix G, as well as the minimum temperature requirements of the rule for bolting up the reactor vessel during normal and pressure testing operations. In addition, NRC guidance related to P/T limit curves is found in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," (i.e., Reference 10) and Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," (i.e., Reference 11).

Appendix H to 10 CFR 50 establishes requirements related to facility material surveillance programs. LSCS demonstrates its compliance with the Appendix H requirements through participation in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) (i.e., Reference 8).

In March 2001, the NRC issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (i.e., Reference 12). RG 1.190 describes methods and assumptions acceptable to the NRC for determining neutron fluence. The new P/T curves for LSCS were developed using the methodology of GE Topical Report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," (i.e., Reference 5), which was approved by the NRC in Reference 6 and adheres to the guidance in RG 1.190.

Section 182a of the Atomic Energy Act of 1954 requires applicants for nuclear power plant operating licenses to include TS as part of the license. The NRC requirements related to the content of TS are set forth in 10 CFR 50.36, "Technical specifications," which requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

10 CFR 50.36(c)(2)(ii) requires that LCOs be established for the P/T limits because the parameters fall within the scope of Criterion 2 identified in the rule:

> A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The P/T limits fall within the scope of Criterion 2 of 10 CFR 50.36(c)(2)(ii) and are therefore required to be included within the TS LCOs for a plant-specific facility operating license.

The regulatory requirements described above will continue to be met with implementation of the proposed change. The analyses provided in Attachments 4 and 5 provide additional details to demonstrate compliance with these regulatory requirements.

4.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," to incorporate revised P/T curves that are valid for up to 32 effective full power years (EFPY) of operation.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license 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 any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The proposed change revises TS Section 3.4.11 to replace the existing P/T curves with revised curves that are valid up to 32 EFPY. The revised curves were developed using the methodology of General Electric (GE) Topical Report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations." The NEDC-32983P methodology has been approved by the NRC for use by licensees. The P/T limits are not derived from design basis accident analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary, a condition that is unanalyzed. Since the P/T limits are not derived from any design basis accident, 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.

Thus, the proposed changes do not have any affect on the probability of an accident previously evaluated.

The P/T curves are used as operational limits 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 P/T curves provide assurance that station operation is consistent with previously evaluated accidents. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change does not change the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation, or failure mechanisms.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The proposed change adopts P/T curves that have been developed using the methodology of GE Topical Report NEDC-32983P. The NEDC-32983P methodology adheres to the guidance in NRC Regulatory Guide 1.190, "Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. In a letter dated September 14, 2001, the NRC approved NEDC-32983P for use by licensees. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed change.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

4.3 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from S. P. Sands (U.S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "LaSalle County Station, Units 1 and 2, Issuance of Amendments Re: Pressure-Temperature Limits (TAC Nos. MB7795 and MB7796)," dated December 10, 2004

2. Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Request for Amendment to Technical Specifications Section 3.4.11, 'RCS Pressure and Temperature (P/T) Limits,'" dated January 31, 2003

3. Letter from K. A. Ainger (Exelon Generation Company, LLC) to U.S. NRC, "Supplement to Request for Amendment to Technical Specifications Section 3.4.11, 'RCS Pressure and Temperature (P/T) Limits,'" dated July 7, 2004

4. Letter from K. A. Ainger (Exelon Generation Company, LLC) to U.S. NRC, "Response to Request for Additional Information Regarding Request for Amendment to Technical Specifications Section 3.4.11, 'RCS Pressure and Temperature (P/T) Limits,'" dated November 15, 2004

5. General Electric Topical Report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations"

6. Letter from S. A. Richards (U.S. NRC) to J. F. Klapproth (GE Nuclear Energy), "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation' (TAC No. MA9891)," dated September 14, 2001

7. Letter from H. N. Berkow (U.S. NRC) to G. B. Stramback (GE Nuclear Energy), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation' (TAC No. MC3788)," dated November 17, 2005

8. Letter from W. A. Macon (U.S. NRC) to J. L. Skolds, "LaSalle County Station, Units 1 and 2 – Issuance of Amendment (TAC Nos. MB7001 and MB7002)," dated August 13, 2003

9. Boiling Water Reactor Vessels and Internals Project (BWRVIP) Report BWRVIP-135, Revision 1, "Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations," dated June 2007

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

11. NUREG-0800, Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," dated March 2007

12. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

LaSalle County Station, Units 1 and 2

Facility Operating License Nos. NPF-11 and NPF-18

REVISED TECHNICAL SPECIFICATIONS PAGES

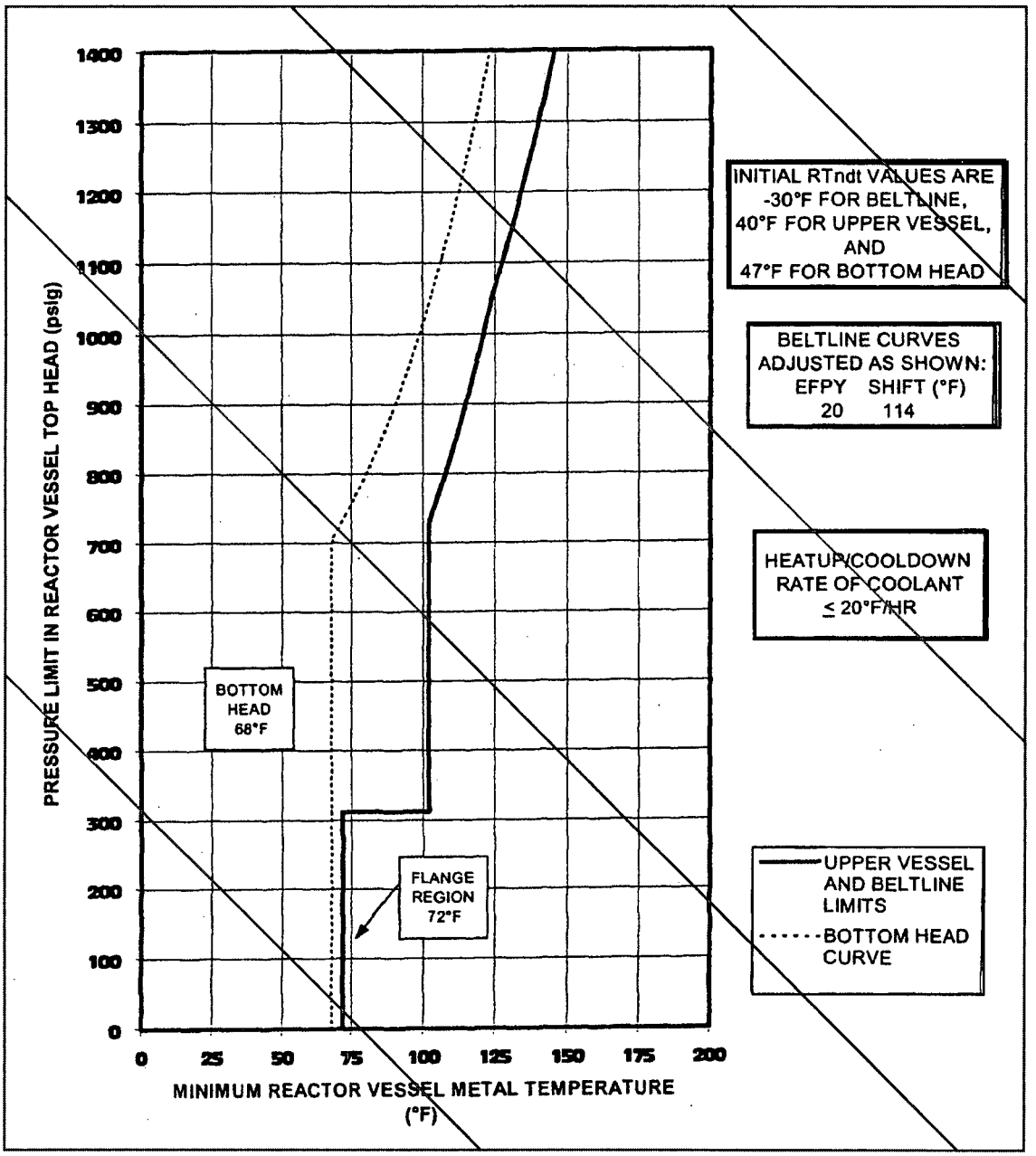
3.4.11-3
3.4.11-6
3.4.11-7
3.4.11-8
3.4.11-9
3.4.11-10
3.4.11-11

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------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 within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3 for Unit 1 up to ~~20~~ EFPY, and Figures 3.4.11-4, 3.4.11-5, and 3.4.11-6 for Unit 2 up to ~~20~~ EFPY; b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period; and c. RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 up to ~~20~~ EFPY and Figure 3.4.11-5 for Unit 2 up to ~~20~~ EFPY.	30 minutes 32
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.11-3 for Unit 1 up to ~~20~~ EFPY and Figure 3.4.11-6 for Unit 2 up to ~~20~~ EFPY. 32	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

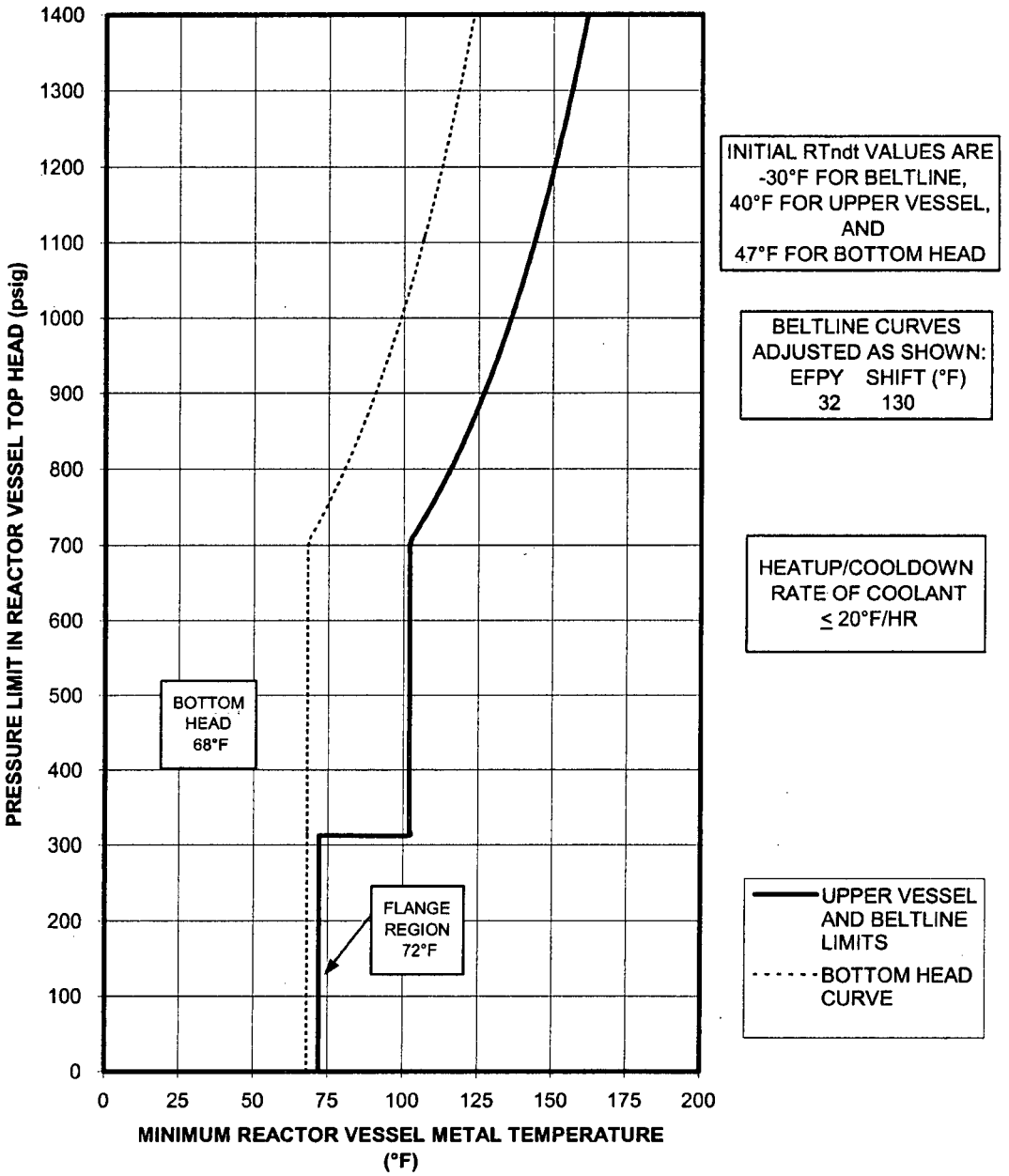
Replace with INSERT 3.4.11-1



32

Figure 3.4.11-1 (Page 1 of 1)
Unit 1
P-T Curves for Hydrostatic or Leak Testing up to ~~20~~ EFPY

INSERT 3.4.11-1



Replace with INSERT 3.4.11-2

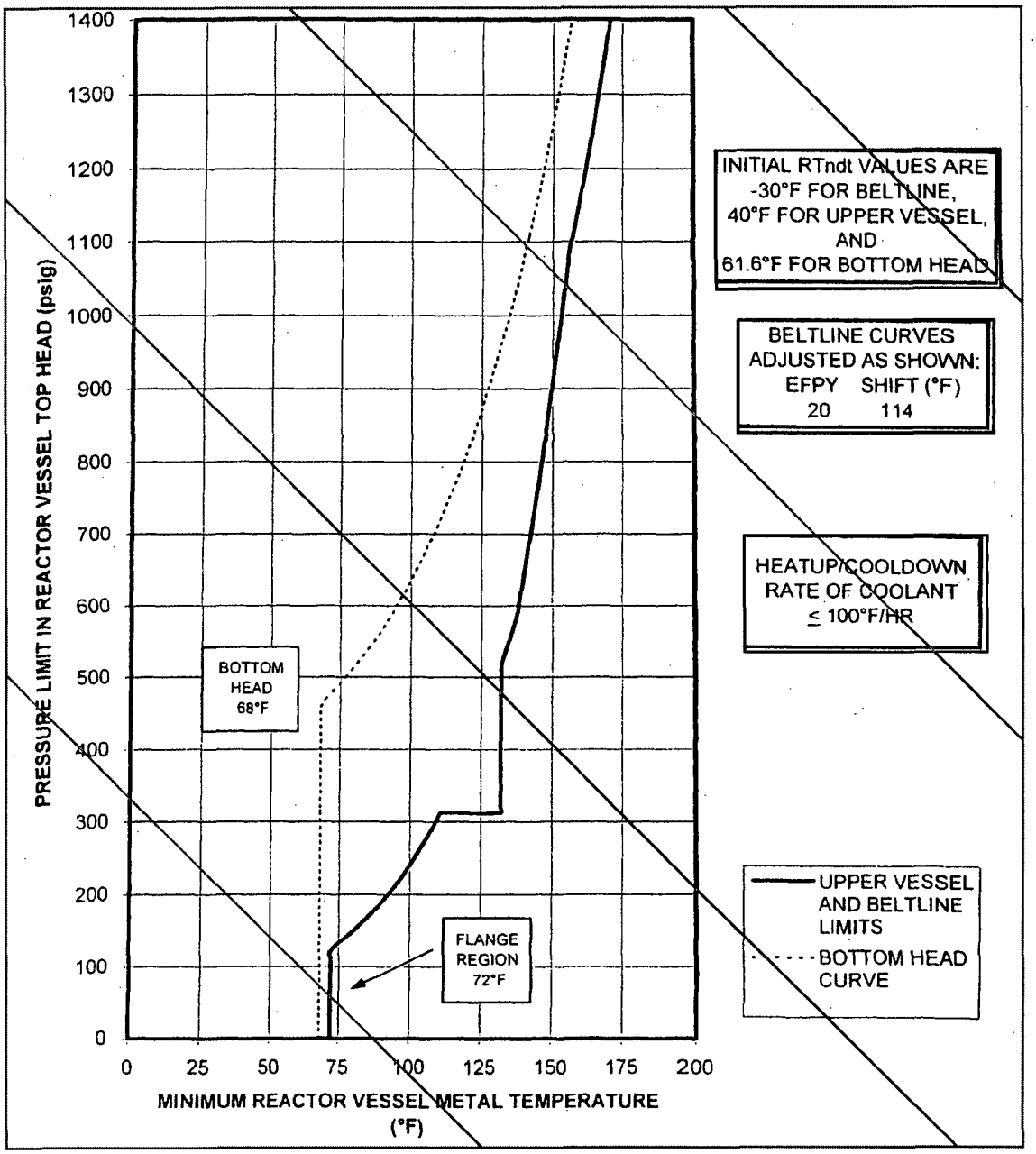
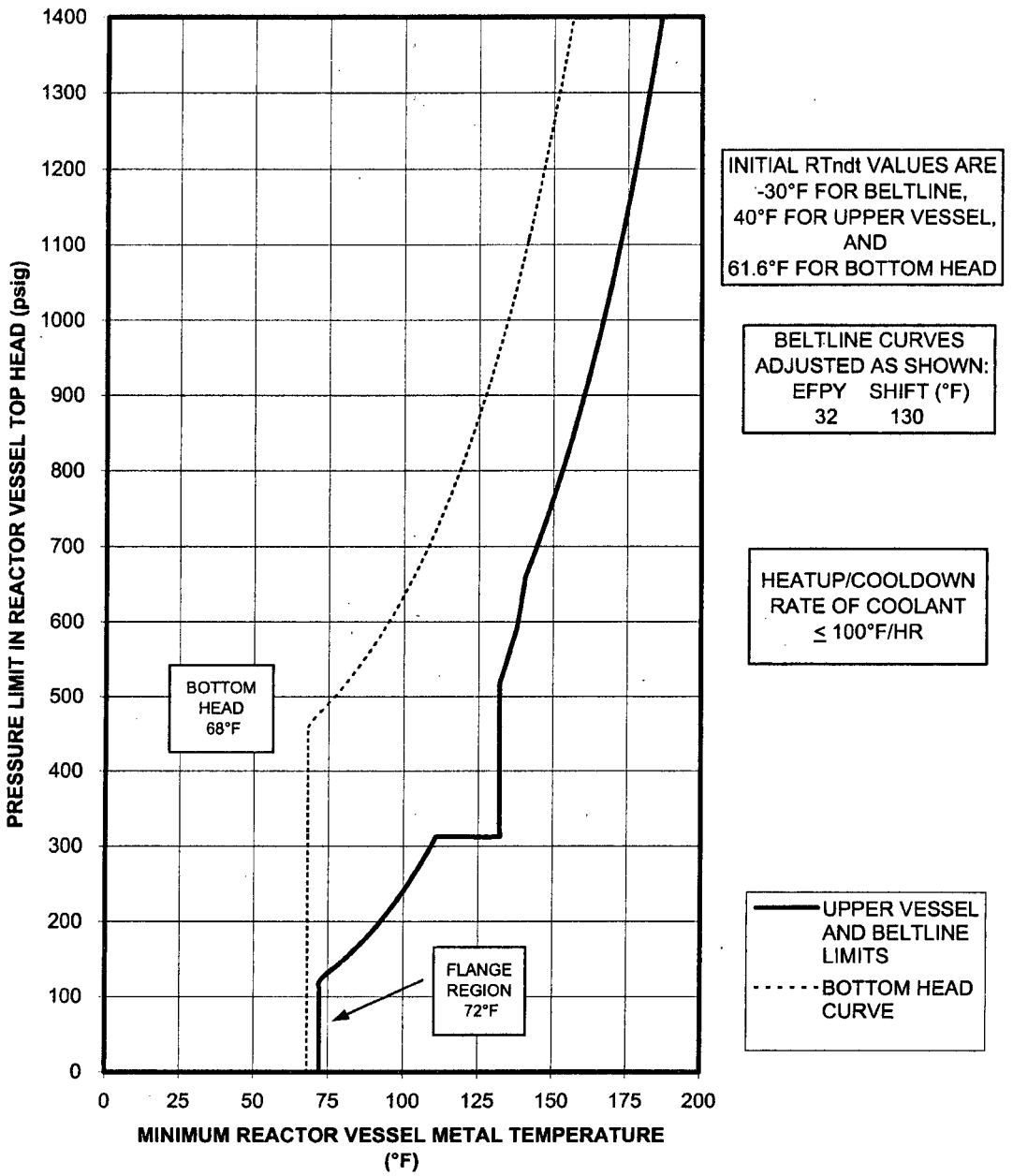


Figure 3.4.11-2 (Page 1 of 1)
Unit 1
P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following a Nuclear Shutdown and Low Power Physics Testing up to ~~20~~ EFPY

32

INSERT 3.4.11-2



Replace with INSERT 3.4.11-3

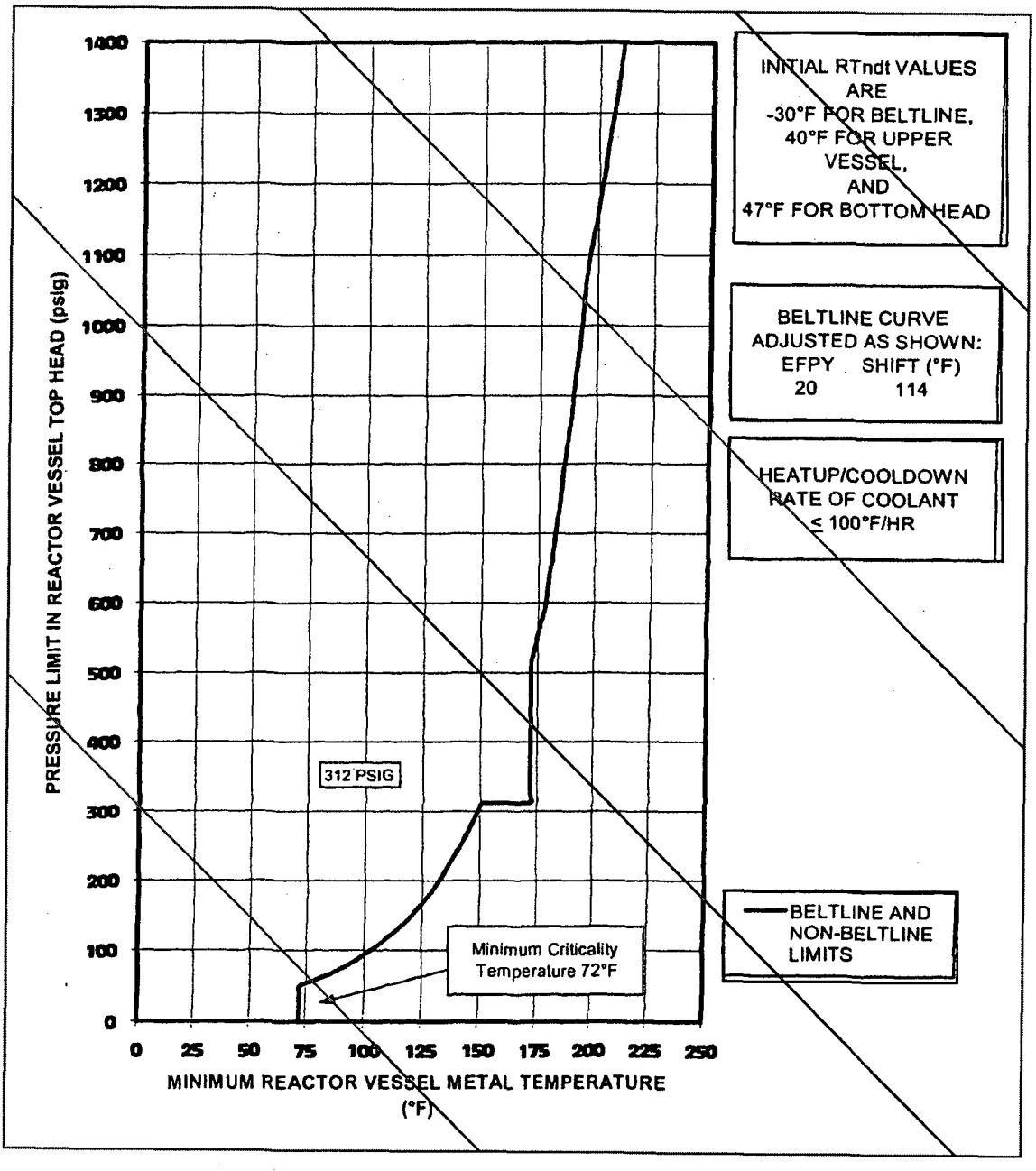
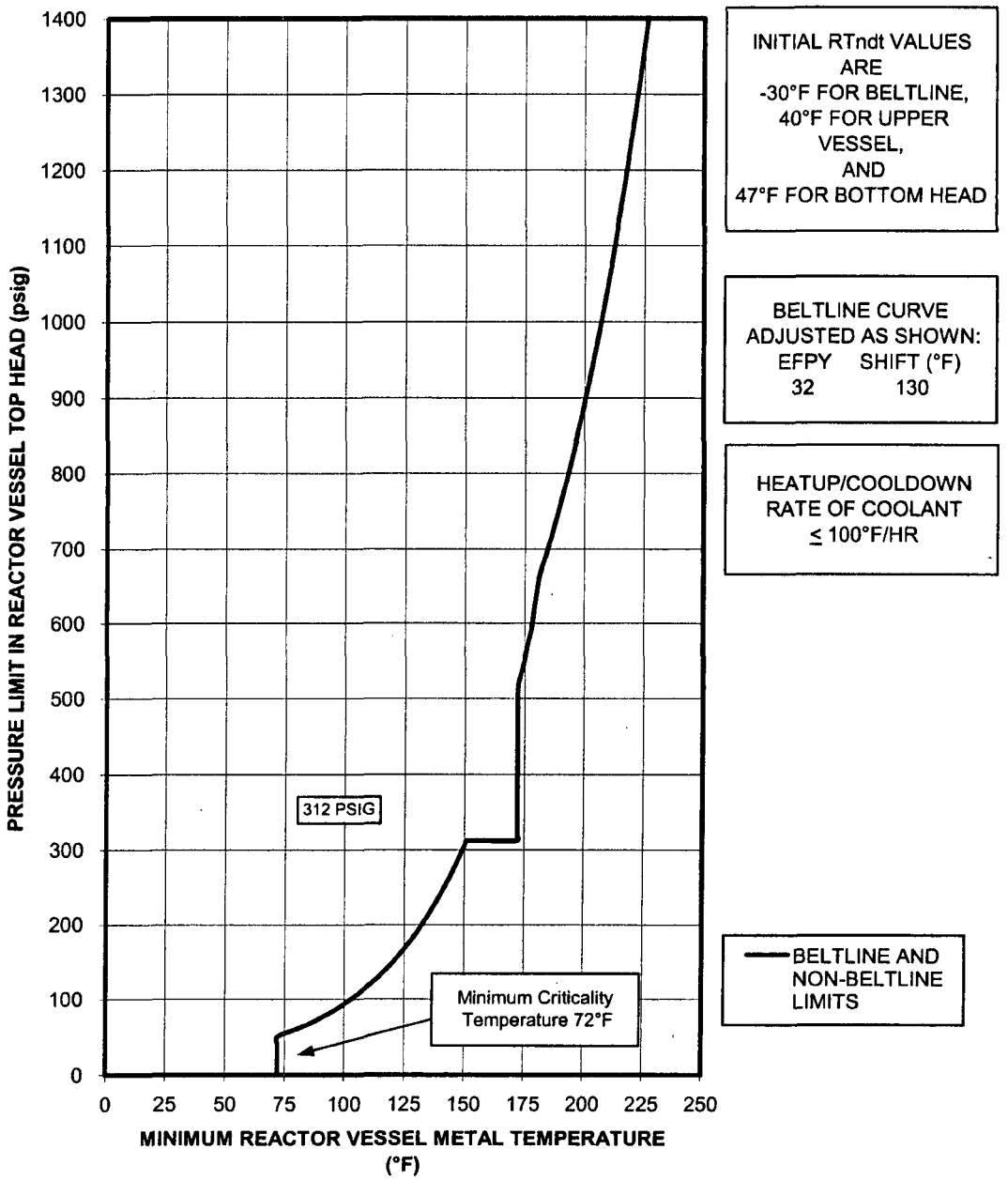
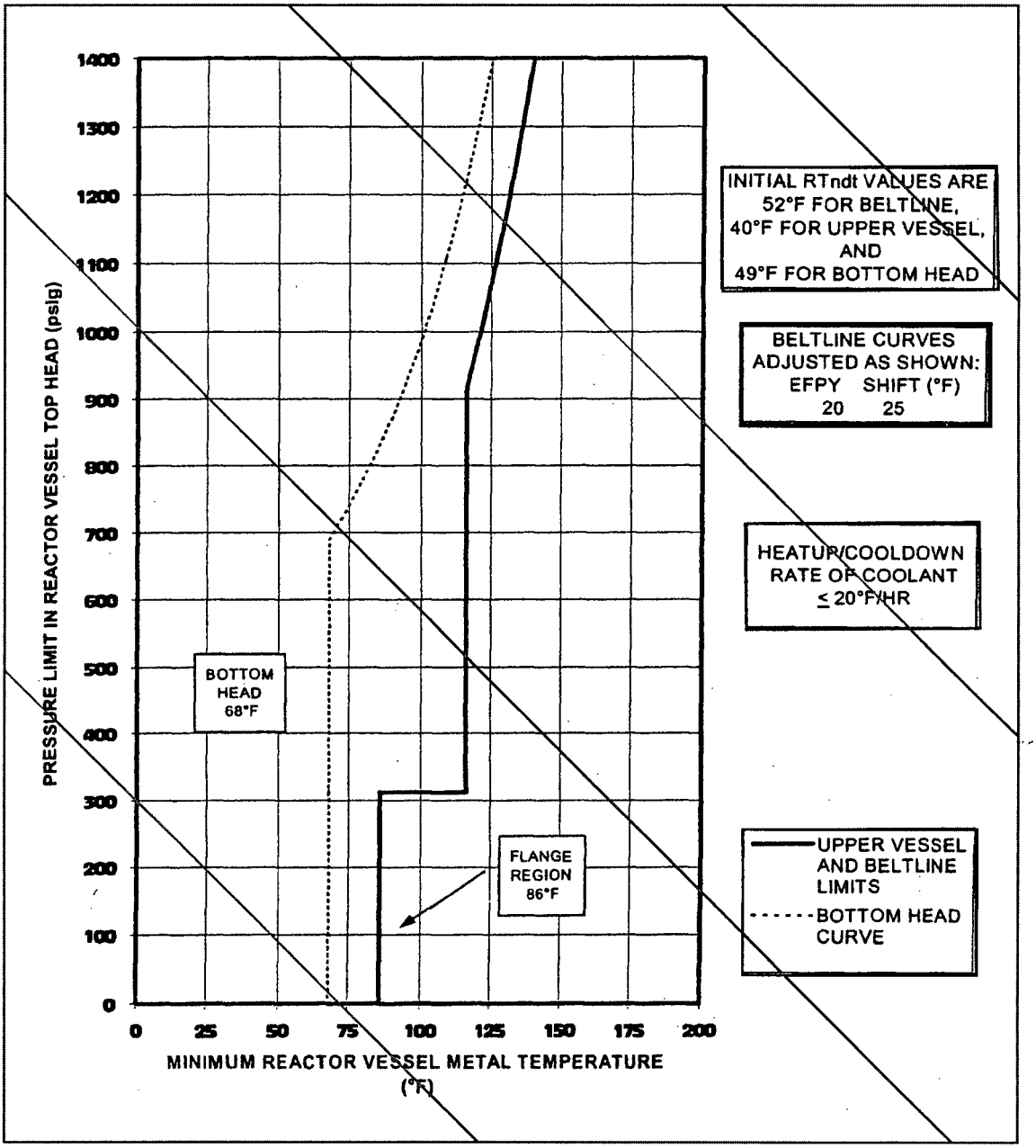


Figure 3.4.11-3 (Page 1 of 1)
Unit 1
P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing up to ~~20~~ EFPY 32

INSERT 3.4.11-3



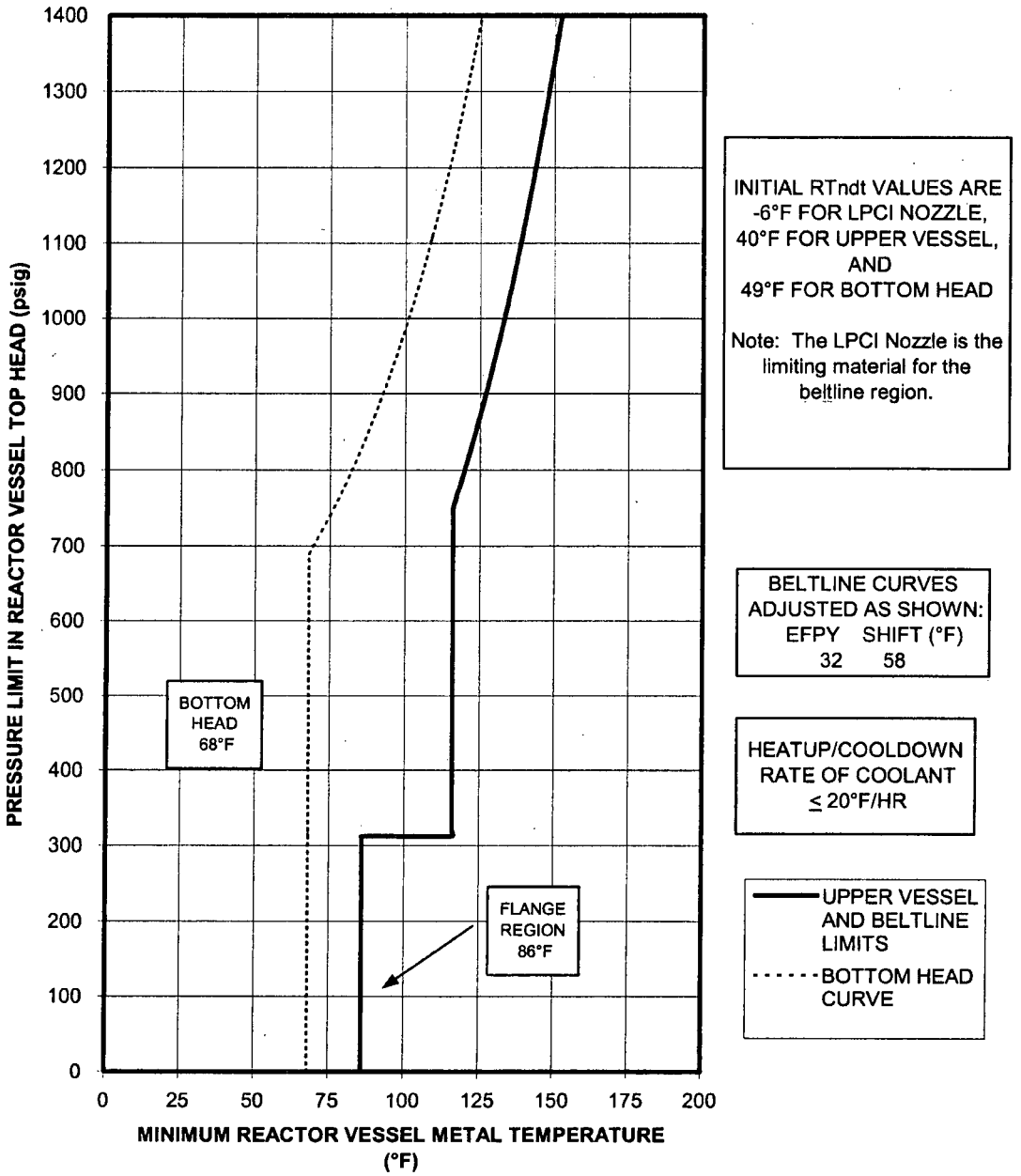
Replace with INSERT 3.4.11-4



32

Figure 3.4.11-4 (Page 1 of 1)
Unit 2
P-T Curves for Hydrostatic or Leak Testing up to ~~20~~ EFPY

INSERT 3.4.11-4



Replace with INSERT 3.4.11-5

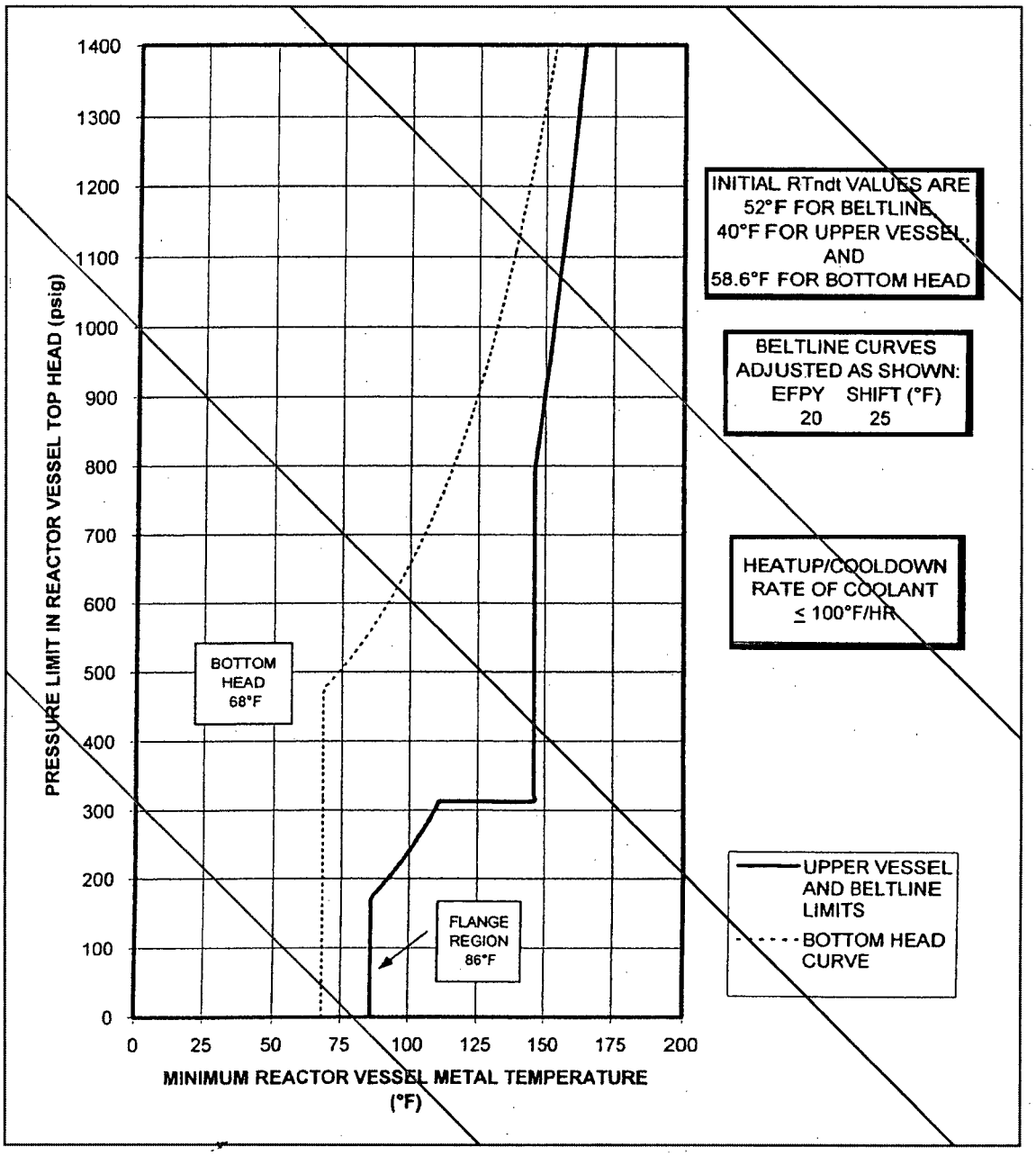
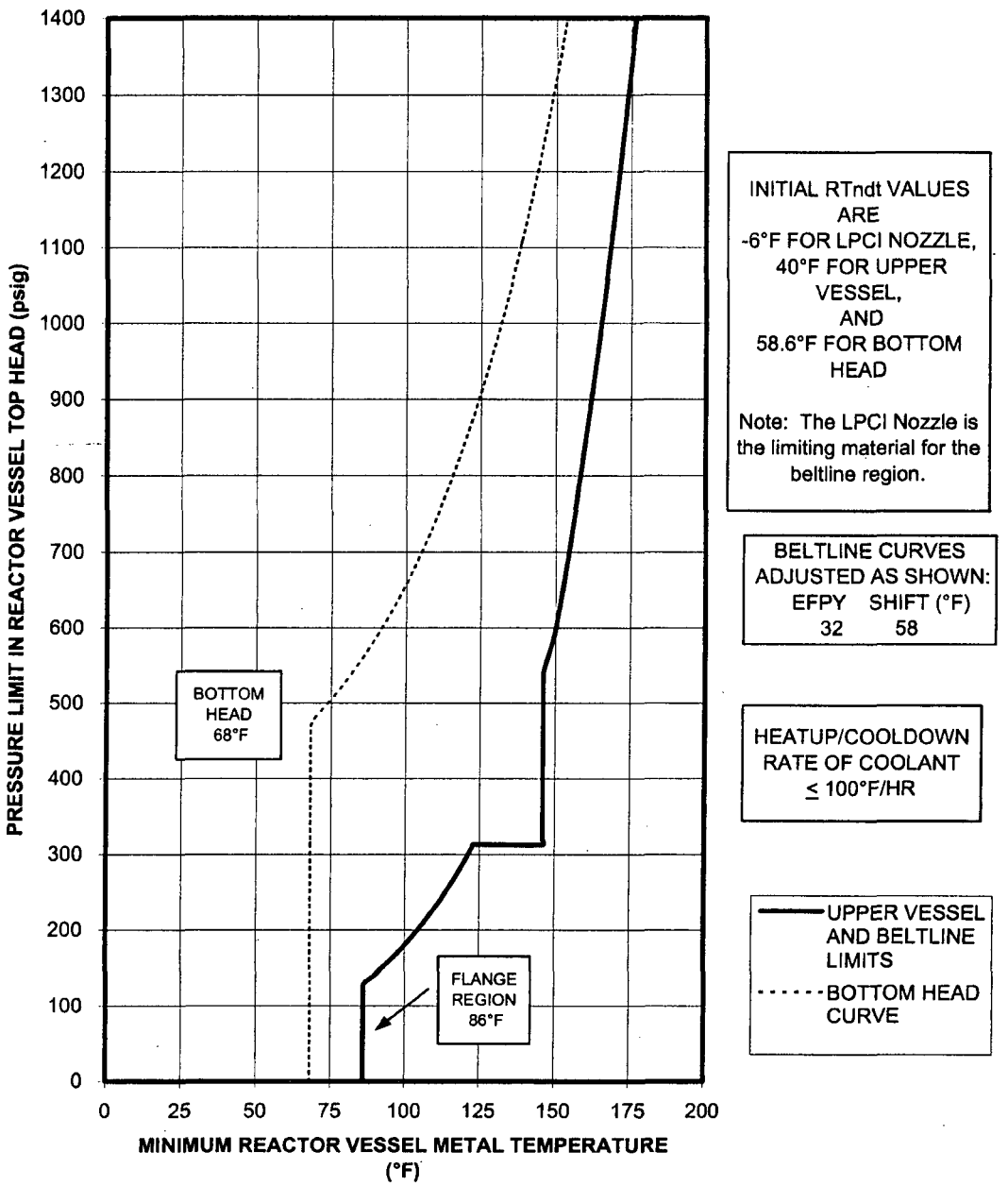


Figure 3.4.11-5 (Page 1 of 1)
Unit 2
P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following a Nuclear Shutdown and Low Power Physics Testing up to ~~20~~ EFPY 32

INSERT 3.4.11-5



Replace with INSERT 3.4.11-6

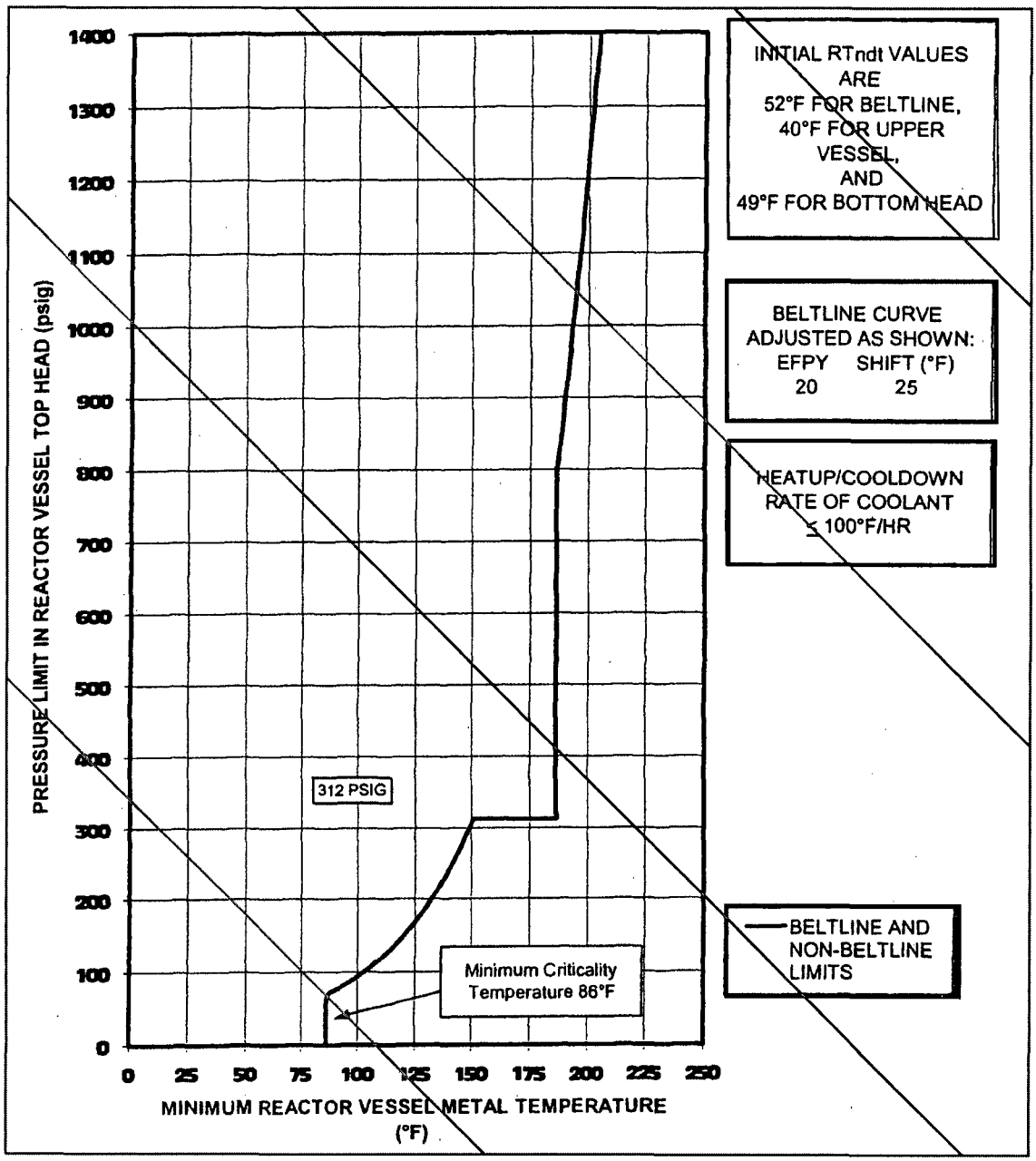
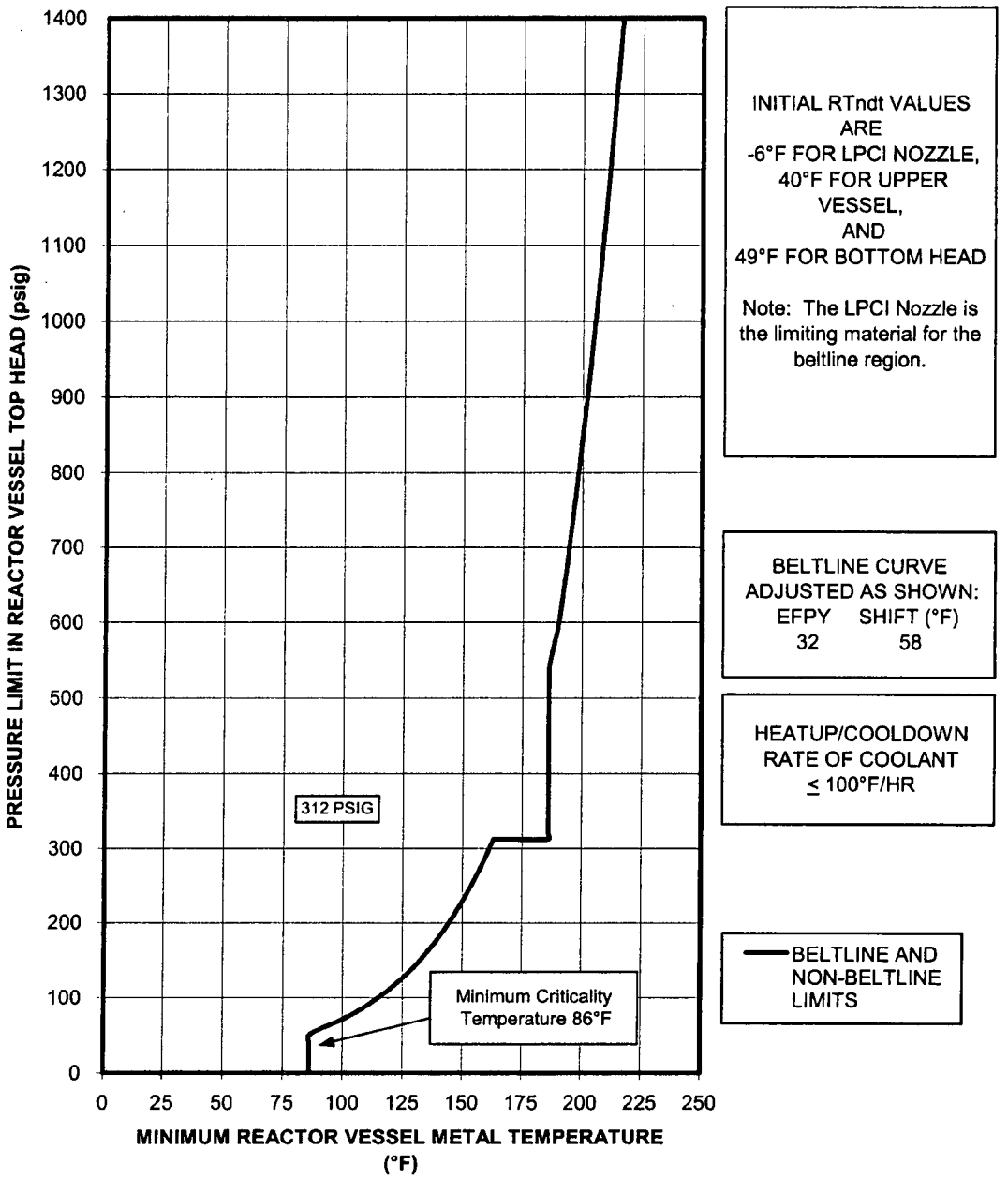


Figure 3.4.11-6 (Page 1 of 1)
Unit 2
P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing up to ~~20~~ EFPY 32

INSERT 3.4.11-6



ATTACHMENT 3
Markup of Proposed Technical Specifications Bases Page

LaSalle County Station, Units 1 and 2

Facility Operating License Nos. NPF-11 and NPF-18

REVISED TECHNICAL SPECIFICATIONS BASES PAGE

B 3.4.11-1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 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.

The Specification contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, and criticality and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable for ~~20~~ 32 effective full power years.

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 American Society of Mechanical Engineers (ASME) Code, Section III, 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 10 CFR 50, Appendix H

(continued)