Exelon Generation.

4300 Winfield Road
Warrenville, IL 60555
630 657 2000 Office

Proprietary Information – Withhold from Public Disclosure Under 10 CFR 2.390

10 CFR 50.90

RS-13-266
December 20, 2013

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

LaSalle County Station, Unit 1
Facility Operating License No. NPF-11
NRC Docket No. 50-373

Subject: License Amendment Request to Revise Reactor Coolant System (RCS) Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1

Reference: Letter from P. J. Karaba (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Evaluation of LaSalle County Station Unit 1 120° Capsule Surveillance Data," dated January 10, 2013

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 No. NPF-11 for LaSalle County Station (LSCS), Unit 1. The proposed change would revise Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits", Figures 3.4.11-1 through 3.4.11-3. The changes to TS 3.4.11 are necessary to address the discovery of a non-conservative TS.

LSCS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), currently administrated by the Electric Power Research Institute (EPRI). The 120° capsule was removed from Unit 1 in February 2010, in accordance with the BWRVIP protocol of the ISP. The referenced letter provided the results of the testing performed on the specimens. Specifically, the limiting beltline material shift value for Unit 1 is increased, and consequently, results in an increase in the Adjusted Reference Temperature (ART). As a result, the Unit 1 P/T curves are non-conservative for 32 Effective Full Power Years (EFPY).

This issue is not applicable to LSCS Unit 2 because the Unit 2 reactor vessel does not contain the specific materials evaluated in the surveillance test report. In addition, Unit 2 is not included in the scope of this LAR because the P/T curves for Unit 2 will not expire until 32 EFPY, and as of April 30, 2012, Unit 2 had operated for 20.37 EFPY.

Attachments 4 and 6 contains Proprietary Information. Withhold from public disclosure under 10 CFR 2.390. When separated from Attachments 4 and 6, this document is decontrolled.

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NRR

Proprietary Information – Withhold from Public Disclosure Under 10 CFR 2.390

Currently plant operations in TS 3.4.11 are administratively controlled under the provisions of NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," to assure that plant safety is maintained. This license amendment request is submitted in accordance with the guidance in AL 98-10. In accordance with the guidance of AL 98-10, EGC submits the proposed change as a required license amendment request to resolve a non-conservative TS. As such, this is not a "voluntary request from a licensee to change its licensing basis" and should not be subject to "forward fit" considerations.

The attached request is subdivided as follows:

- Attachment 1 provides a description and evaluation of the proposed changes.
- Attachment 2 provides the markup of the affected TS pages.
- Attachment 3 provides a revised copy of the TS pages with the proposed changes incorporated.
- Attachment 4 provides the request for withholding EPRI proprietary information, EPRI Affidavit and the EPRI proprietary Pressure and Temperature Limits Report up to 32 EFPY for LSCS, Unit 1.
- Attachment 5 provides the non-proprietary Pressure and Temperature Limits Report up to 32 EFPY for LSCS, Unit 1.
- Attachment 6 provides the request for withholding EPRI proprietary information, request for withholding General Electric Hitachi (GEH) proprietary information, EPRI and GEH Affidavits, and the GEH and EPRI proprietary LSCS, Unit 1, specific responses to the Grand Gulf request for additional information (RAI).
- Attachment 7 provides the GEH and EPRI non-proprietary LSCS, Unit 1, specific responses to the Grand Gulf RAI.

Attachments 4 and 6 contain proprietary information as defined by 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." EPRI, as the owner of the proprietary information, has executed the affidavit contained in Attachment 4, enclosure 2, and Attachment 6, enclosure 2, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to EGC in an EPRI transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in the attached information such that the affidavit remains applicable. EPRI hereby requests that the attached proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17. Information that is not considered proprietary in Attachments 4 and 6 is provided separately in Attachments 5 and 7, respectively.

Proprietary Information – Withhold from Public Disclosure Under 10 CFR 2.390

Attachment 6 contains proprietary information as defined by 10 CFR 2.390. GEH, as the owner of the proprietary information, has executed the affidavit contained in Attachment 6, enclosure 4, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to EGC in a GEH transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in the attached information such that the affidavit remains applicable. GEH hereby requests that the attached proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17. Information that is not considered proprietary is provided separately in Attachment 7.

Exelon requests approval of the proposed license amendment request by December 20, 2014. Once approved, this amendment shall be implemented within 60 days of issuance.

The proposed changes have 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.

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 within this letter. Should you have any questions concerning this letter, please contact Ms. Lisa A. Simpson at (630) 657-2815.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of December 2013.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Proprietary Information – Withhold from Public Disclosure Under 10 CFR 2.390

Attachments:

1) Evaluation of Proposed Changes
2) Mark-up of Proposed Technical Specifications Pages
3) Revised Technical Specifications Pages
4) LaSalle County Station, Unit 1, Pressure and Temperature Limits Report up to 32 EFPY
   Enclosure 1 – EPRI Request for Withholding
   Enclosure 2 – EPRI Affidavit
   Enclosure 3 – LaSalle County Station Unit 1 Pressure / Temperature Limits Report (Proprietary)
5) Non-Proprietary LaSalle County Station Unit 1 Pressure / Temperature Limits Report
6) LaSalle County Station, Unit 1, Responses to Grand Gulf RAI
   Enclosure 1 – EPRI Request for Withholding
   Enclosure 2 – EPRI Affidavit
   Enclosure 3 – GEH Request for Withholding
   Enclosure 4 – GEH Affidavit
   Enclosure 5 – LaSalle County Station Unit 1 Responses to Grand Gulf RAI (Proprietary)
7) Non-Proprietary LaSalle County Station, Unit 1, Responses to Grand Gulf RAI

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 Changes

Subject: License Amendment Request to Revise Reactor Coolant System (RCS) Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Facility Operating License No. NPF-11 for LaSalle County Station (LSCS), Unit 1.

Exelon Generation Company, LLC (EGC) proposes to revise Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits", Figures 3.4.11-1 through 3.4.11-3. The changes to TS 3.4.11 are necessary to address the discovery of a non-conservative TS.

Currently plant operations in TS 3.4.11 are administratively controlled under the provisions of NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," (Reference 1) to assure that plant safety is maintained. This license amendment request is submitted in accordance with the guidance in AL 98-10. In accordance with the guidance of AL 98-10, EGC submits the proposed change as a required license amendment request to resolve a non-conservative TS. As such, this is not a "voluntary request from a licensee to change its licensing basis" and should not be subject to "forward fit" considerations.

This issue is not applicable to LSCS Unit 2 because the Unit 2 reactor vessel does not contain the materials evaluated in the surveillance test report. Specifically, the LSCS Unit 1 surveillance capsule report provided data on weld 1P3571 and plate C6345-1. These heats were used by Combustion Engineering in the fabrication of the Unit 1 reactor vessel; however, these heats were not used by Chicago Bridge & Iron in the fabrication of the Unit 2 reactor vessel. Therefore, the capsule data is not applicable to Unit 2. In addition, Unit 2 is not included in the scope of this LAR because the P/T curves for Unit 2 will not expire until 32 EFPY, and as of April 30, 2012, Unit 2 had operated for 20.37 EFPY.

Approval of this amendment application is requested by December 20, 2014. Once approved, this amendment will be implemented within 60 days.

2.0 DETAILED DESCRIPTION

The proposed change revises TS Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits", Figures 3.4.11-1 through 3.4.11-3 based on the results of testing of the Integrated Surveillance Capsule.

Attachment 2 provides the existing TS pages marked-up to show the proposed changes.

Attachment 3 provides the P/T curves developed to represent steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline region. LSCS Unit 1 is currently licensed to P/T curves for up to 32 EFPY; the analysis performed in this report provides curves for up to 32 EFPY. The 1998 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code including 2000 Addenda was used in this evaluation.

3.0 BACKGROUND

The revised P/T curves were developed in accordance with the General Electric Hitachi Nuclear Energy Americas LLC (GEH) Licensing Topical Report NEDC-33178P-A, Revision 1 (Reference 2).

As documented in Section 4.0 of the NRC Safety Evaluation for NEDC-33178P-A (Reference 3), licensees who choose to implement NEDC-33178P-A, Revision 1, as their facility's methodology must address the following plant-specific action item:

> The licensee must identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.

Accordingly, the LSCS Unit 1 P/T curves incorporate a neutron fluence that was calculated using the following NRC approved methodologies:

- The first thirteen cycles of fluence were calculated in accordance with EPRI Report BWRVIP-126, "BWR Vessel and Internals Project, RAMA Fluence Methodology Software," Version 1.0, EPRI, Palo Alto, CA: December 2003, Technical Report 1007823 (Reference 4), which was approved by the NRC on May 13, 2005 (Reference 5).

- The fluence subsequent to cycle 13 was calculated in accordance with General Electric Licensing Topical Report NEDC-32983P-A, "GE Hitachi Nuclear Energy Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," Revision 2, January 2006 (Reference 6), which was approved by the NRC on November 17, 2005 (Reference 7).

Each NRC approved methodology meets the RG 1.190 requirements and the plant-specific condition of the NRC Safety Evaluation for NEDC-33178P-A. A comparison of the fluence values for 32 EFPY between the dual calculation (RAMA followed by GEH) and draft calculations of RAMA alone indicates that the dual calculation bounds the single calculation (i.e., results for RAMA alone are less than the dual methodology used in the development of the P/T curves). Therefore, EGC has determined the dual methodology approach utilized to support this LAR results in more conservative fluence input to the P/T curves.

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. TS 3.4.11 Limiting Condition for Operation (LCO) limits the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

TS Section 3.4.11 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. This specification establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary.

10 CFR 50, Appendix G requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary materials, an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests and mandates the use of the ASME Code, Section III, Appendix G.

The actual shift in the reference temperature (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 and 10 CFR 50, Appendix H. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2.

LSCS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), currently administrated by Electric Power Research Institute (EPRI). The 120° capsule was removed from Unit 1 in February 2010 and, in accordance with the BWRVIP protocol of the ISP was tested. Based on testing performed on the specimens, the limiting beltline material shift value for Unit 1 is increased, and consequently, results in an increase in the Adjusted Reference Temperature (ART), which is the initial RT_{NDT} plus the change in RT_{NDT} (ΔRT_{NDT}) plus margin. As a result, the Unit 1 P/T curves are non-conservative for 32 Effective Full Power Years (EFPY).

The ISP test results are provided in Attachments 4 and 6.

In Reference 8, the U. S. Nuclear Regulatory Commission requested additional information concerning the Grand Gulf Nuclear Station, Unit 1, license amendment request pertaining to the implementation of a Pressure and Temperature Limits Report (PTLR). Attachment 6 provides the LSCS, Unit 1, specific responses to the Grand Gulf RAI. The NRC requested that future PTLR or P/T curve submittals include responses to the Grand Gulf questions.

4.0 TECHNICAL EVALUATION

10 CFR 50, Appendix G, requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. 10 CFR 50, Appendix G requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G.

The purpose of GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, is to provide the methodology developed by GEH for the determination of reactor pressure vessel P/T curves. The adequacy of the GEH methodology is demonstrated through a detailed description of the calculation procedures and examples showing agreement between GEH practices and the standards and Code requirements set forth in 10 CFR 50, Appendix G. NEDC-33178P-A, Revision 1, does not include development or licensing of vessel fluence methods. The fluence methods are provided in EPRI Report BWRVIP-126, Version 1.0, and GEH Licensing Topical Report NEDC-32983P-A, Revision 2.

GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, provides the current methodology for developing reactor coolant system P/T limit curves and other

associated numerical limits for BWRs. The LSCS Unit 1 P/T curves have been developed in accordance with the NEDC-33178P-A, Revision 1 methodology.

The P/T curves included in Attachment 3 have been developed to present steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline region. Complete P/T curves were developed for 32 EFPY. These P/T curves and a tabulation of the curves are provided in the Attachment 4. This report incorporates a fluence (E > 1 MeV) calculated in accordance with EPRI Report BWRVIP-126, the RAMA fluence methodology (Reference 4), and with GE Licensing Topical Report NEDC-32983P-A, the RPV fast neutron flux methodology (Reference 6). Both of these methodologies have been approved by the NRC (References 5 and 7, respectively) and are in compliance with Regulatory Guide 1.190. The latest information from the BWRVIP ISP that is applicable to LSCS Unit 1 has been utilized.

The methodology used to generate the P/T curves in this report is presented in Section 3.0 of Attachment 4. The 1998 Edition of the ASME Boiler and Pressure Vessel Code including 2000 Addenda was used in this evaluation.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A, (b) non-nuclear heatup/cooldown and low-level physics tests, referred to as core not critical operation or Curve B, and (c) core critical operation, referred to as Curve C. There are four vessel regions that should be monitored against the P/T curve operating limits; these regions are defined on the thermal cycle diagram:

- Closure flange region (Region A)
- Core beltline region (Region B)
- Upper vessel (Regions A & B)
- Lower vessel (Regions B & C)

For the core not critical and the core critical curves, the P/T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. The bounding transients used to develop the curves are described in NEDC-33178P-A, Revision 1. For the hydrostatic pressure and leak test curve, a coolant heatup and cooldown temperature rate of 20°F/hr or less must be maintained at all times.

The P/T curves apply for both heatup and cooldown and for both the 1/4T and 3/4T locations because the maximum tensile stress for either heatup or cooldown is applied at the 1/4T location. For beltline curves this approach has added conservatism because irradiation effects cause the allowable toughness at 1/4T to be less than that at 3/4T for a given metal temperature.

Curves A and B provide separate bottom head as well as composite upper vessel and beltline requirements.
Separate P/T curves were developed for the upper vessel, beltline (at end of license EFPY), and bottom head for the Pressure Test and Core Not Critical conditions. Composite P/T curves

were generated for each of the Pressure Test, Core Not Critical and Core Critical conditions at intermediate and end of license EFPY. The composite curves were generated by enveloping the most restrictive P/T limits from the separate bottom head, beltline, upper vessel and closure assembly P/T limits.

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements/Criteria

As discussed in the Safety Evaluation for GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, the NRC has established requirements in 10 CFR 50, Appendix G in order to protect the integrity of the Reactor Coolant Pressure Boundary in nuclear power plants. Appendix G requires that the P/T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code were used to generate the P/T limits. 10 CFR Part 50, Appendix G also requires that applicable surveillance data from RPV 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 RPV beltline materials. NRC regulatory guidance related to P/T limit curves is found in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, and Standard Review Plan (NUREG-0800) Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock."

Adoption of the NRC-approved methodology described in the GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, for the preparation of the P/T limit curves ensures that the requirements of 10 CFR 50, Appendix G will be satisfied. 10 CFR Part 50, Appendix H, provides criteria for the design and implementation of reactor pressure vessel material surveillance programs for operating light water reactors. LSCS, Unit 1 demonstrates its compliance with the requirements of 10 CFR Part 50, Appendix H, through participation in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) (Reference 9).

The NRC-approved methodology of GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, has been adopted for preparation of the LSCS, Unit 1 P/T limit curves.

As previously discussed, the Unit 1 P/T limits curves incorporate a fluence that was calculated using a combination of NRC approved methods. The first thirteen cycles of fluence were calculated in accordance with EPRI Report BWRVIP-126, the RAMA fluence methodology (Reference 4), approved by the NRC in Reference 5. The fluence subsequent to cycle 13 was calculated in accordance with GE Licensing Topical Report NEDC-32983P-A, the RPV fast neutron flux methodology (Reference 6), approved by the NRC in Reference 7.

Proposed revisions to TS Section 3.4.11, Figures 3.4.11-1 through 3.4.11-3 have been prepared and are provided in Attachment 2 of this submittal.

5.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction, or early site permit," Exelon Generation Company, LLC (EGC) is requesting a change to the Technical Specifications (TS) of Facility Operating License No. NPF-11 for LaSalle County Station (LSCS), Unit 1. LSCS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), currently administrated by Electric Power Research Institute (EPRI). The 120° capsule was removed from Unit 1 in February 2010, in accordance with the BWRVIP protocol of the ISP. Based on testing performed on the specimens, the limiting beltline material shift value for Unit 1 is increased, and consequently, results in an increase in the Adjusted Reference Temperature (ART), which is the initial RT_{NDT} plus the change in RT_{NDT} (ΔRT_{NDT}) plus margin. As a result, the currently licensed Unit 1 P/T curves are non-conservative and need to be revised.

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 an 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 for LSCS, 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.

Criteria

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

   Response: No.

   The proposed change makes no physical changes to the plant. The proposed amendment incorporates the recent ISP results into the NRC-approved methodology of the GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, for the preparation of the LSCS, Unit 1 P/T limit curves. In 10 CFR 50, Appendix G, requirements are established to protect the integrity of the Reactor Coolant Pressure Boundary in nuclear power plants. Implementing the NRC-approved methodology for calculating P/T limit curves

provide an equivalent level of assurance that Reactor Coolant Pressure Boundary integrity will be maintained, as specified in 10 CFR 50, Appendix G.

The proposed changes do not adversely affect accident initiators or precursors, and do not negatively alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

Therefore, the proposed activity 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 revised P/T limits do not alter or involve any design basis accident initiators. Reactor Coolant Pressure Boundary integrity will continue to be maintained in accordance with 10 CFR 50, Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. These changes do not involve any physical alteration of the plant (i.e., no new or different type of equipment will be installed), and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

   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 changes do not affect the function of the Reactor Coolant Pressure Boundary or its response during plant transients. By calculating the P/T limits using NRC-approved methodology, adequate margins of safety relating to Reactor Coolant Pressure Boundary integrity are maintained. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There are no changes to setpoints at which protective actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

   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 amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 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 to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has evaluated the proposed amendment for environmental considerations. The review has resulted in the determination 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, and would change an inspection or surveillance requirement. 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(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1) NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," December 29, 1998

2) GE Licensing Topical Report NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, June 2009

3) Letter from Thomas B. Blount (NRC) to Doug Coleman (Chair, BWROG), "Final Safety Evaluation for Boiling Water Reactors Owners' Group Licensing Topical Report NEDC-33178P, General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves (TAC No. MD2693)," April 27, 2009

4) EPRI Report BWRVIP-126, "BWR Vessel and Internals Project, RAMA Fluence Methodology Software," Version 1.0, EPRI, Palo Alto, CA: December 2003, 1007823

5) Letter from William H. Bateman (NRC) to Bill Eaton (BWRVIP), "Safety Evaluation of Proprietary EPRI Reports BWRVIP-114, 115, 117, and 121 and TWE-PSE-001-R-001," May 13, 2005

6) GE Licensing Topical Report NEDC-32983P-A, "GE Hitachi Nuclear Energy Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," Revision 2, January 2006

7) Letter from Herbert N. Berkow (NRC) to George 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)," November 17, 2005

8) Email from Alan Wang (U. S. Nuclear Regulatory Commission) to Francis Burford and Dana Millar (Grand Gulf Nuclear Station), "GG EPU Request for Additional Information Related to Vessel and Internals Integrity (ME4679)," dated January 31, 2011

9) Letter from William A. Macon, Jr. (USNRC) to John L. Skolds (EGC), "LaSalle County Station, Units 1 and 2 – Issuance of Amendment (TAC Nos. MB7001 and MB7002)," August 13, 2003

ATTACHMENT 2

Mark-up of Proposed Technical Specifications Pages

LASALLE COUNTY STATION
UNIT 1

Docket No. 50-373

Facility Operating License No. NPF-11

<u>REVISED TS PAGES</u>

3.4.11-6
3.4.11-7
3.4.11-8

DELETE and
INSERT new
Figure 3.4.11-1

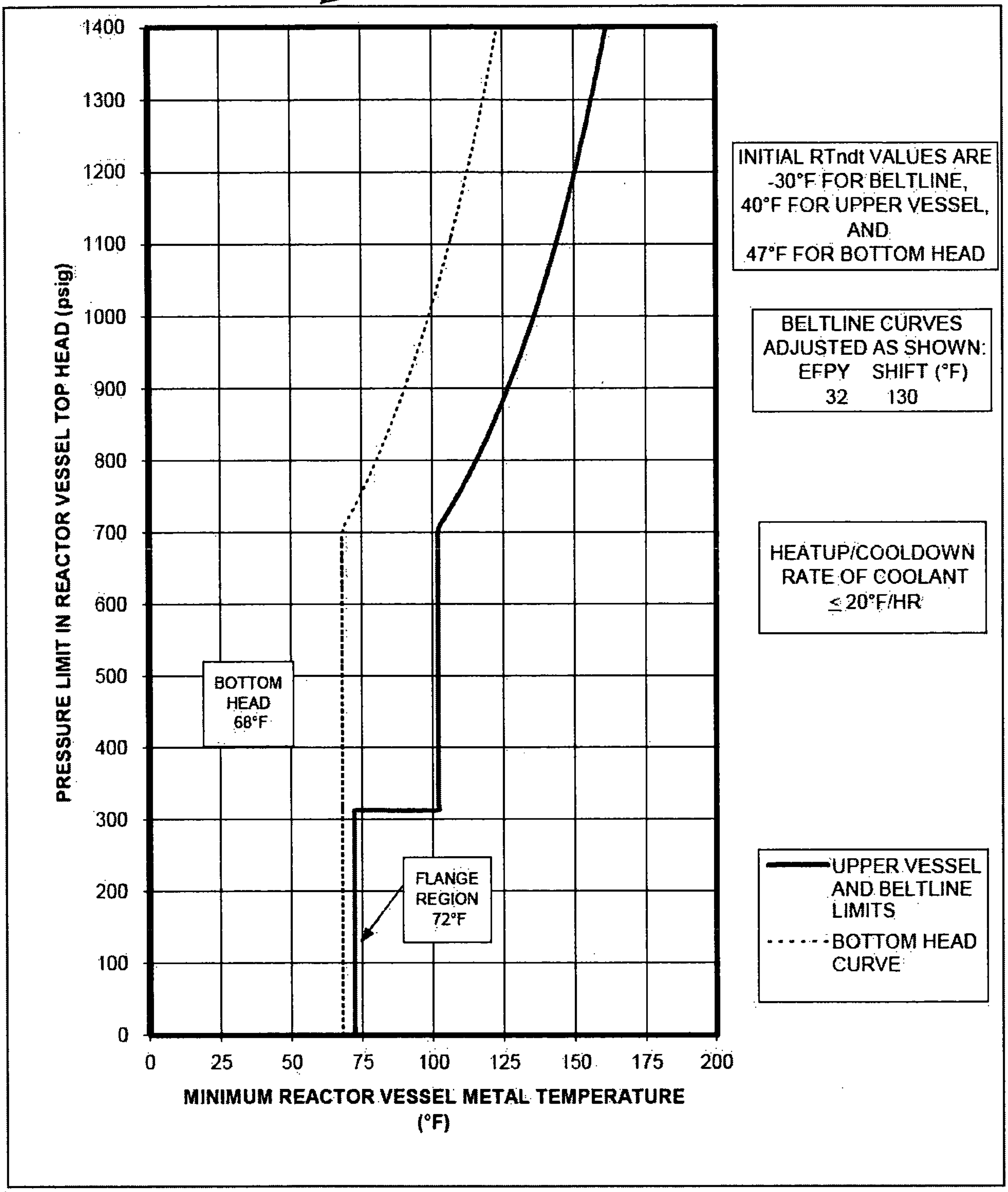
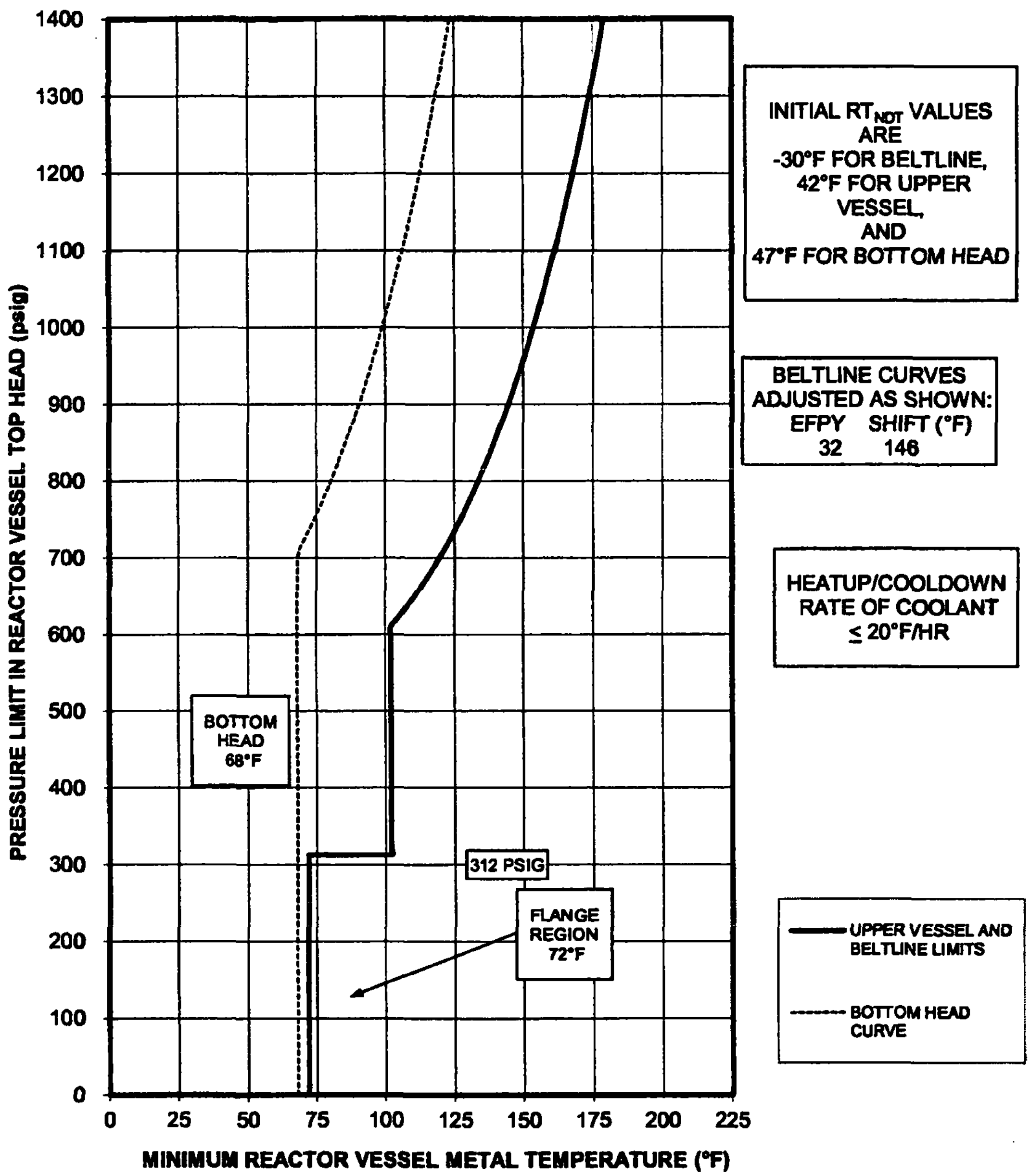


Figure 3.4.11-1 (Page 1 of 1)
Unit 1
P-T Curves for Hydrostatic or Leak Testing up to 32 EFPY

New Figure 3.4.11-1



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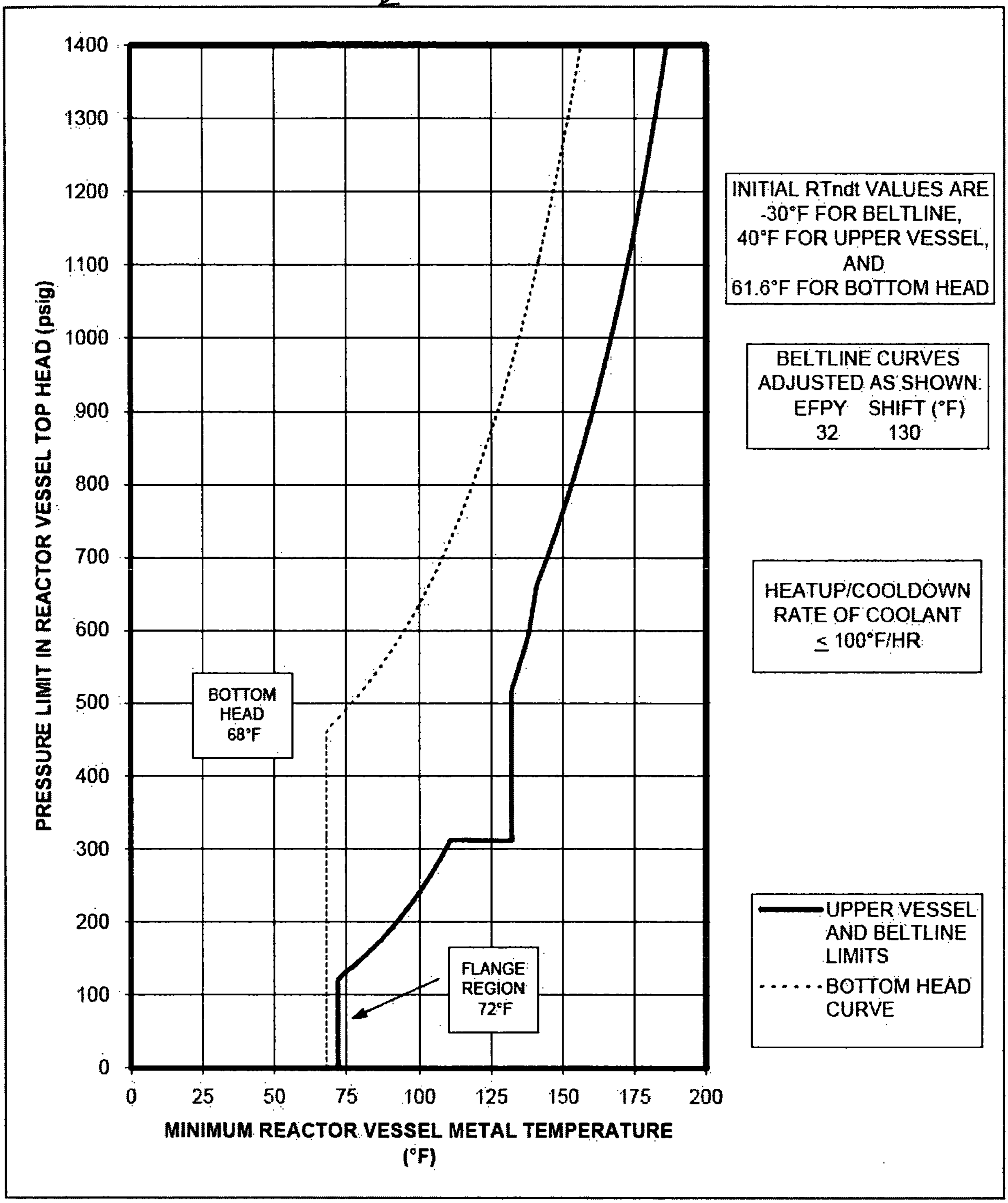
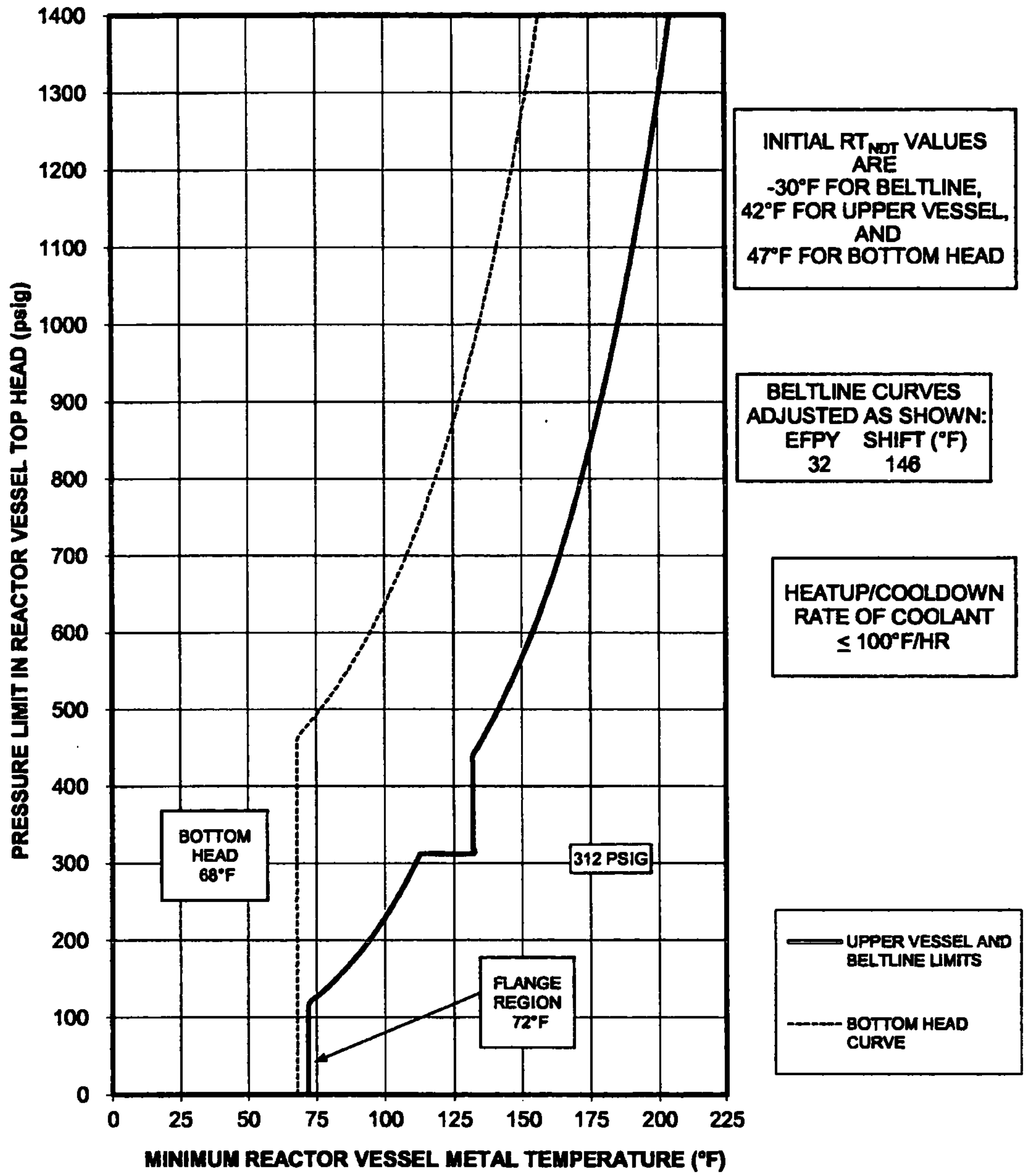


Figure 3.4.11-2 (Page 1 of 1)
Unit 1
P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following a Nuclear Shutdown and Low Power Physics Testing up to 32 EFPY

(Core Not Critical)

New Figure 3.4.11-2



DELETE and INSERT
new Figure 3.4.11-3

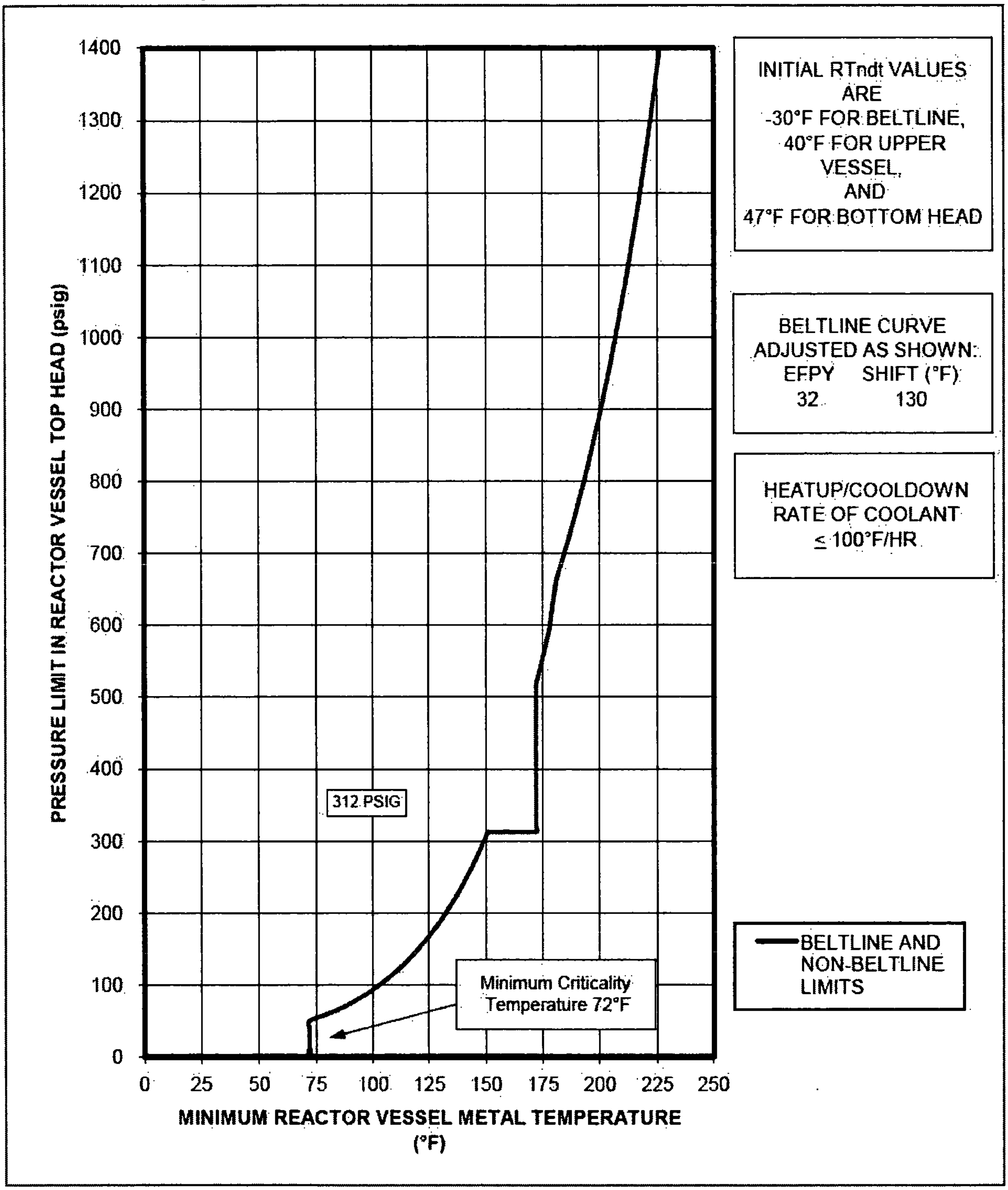
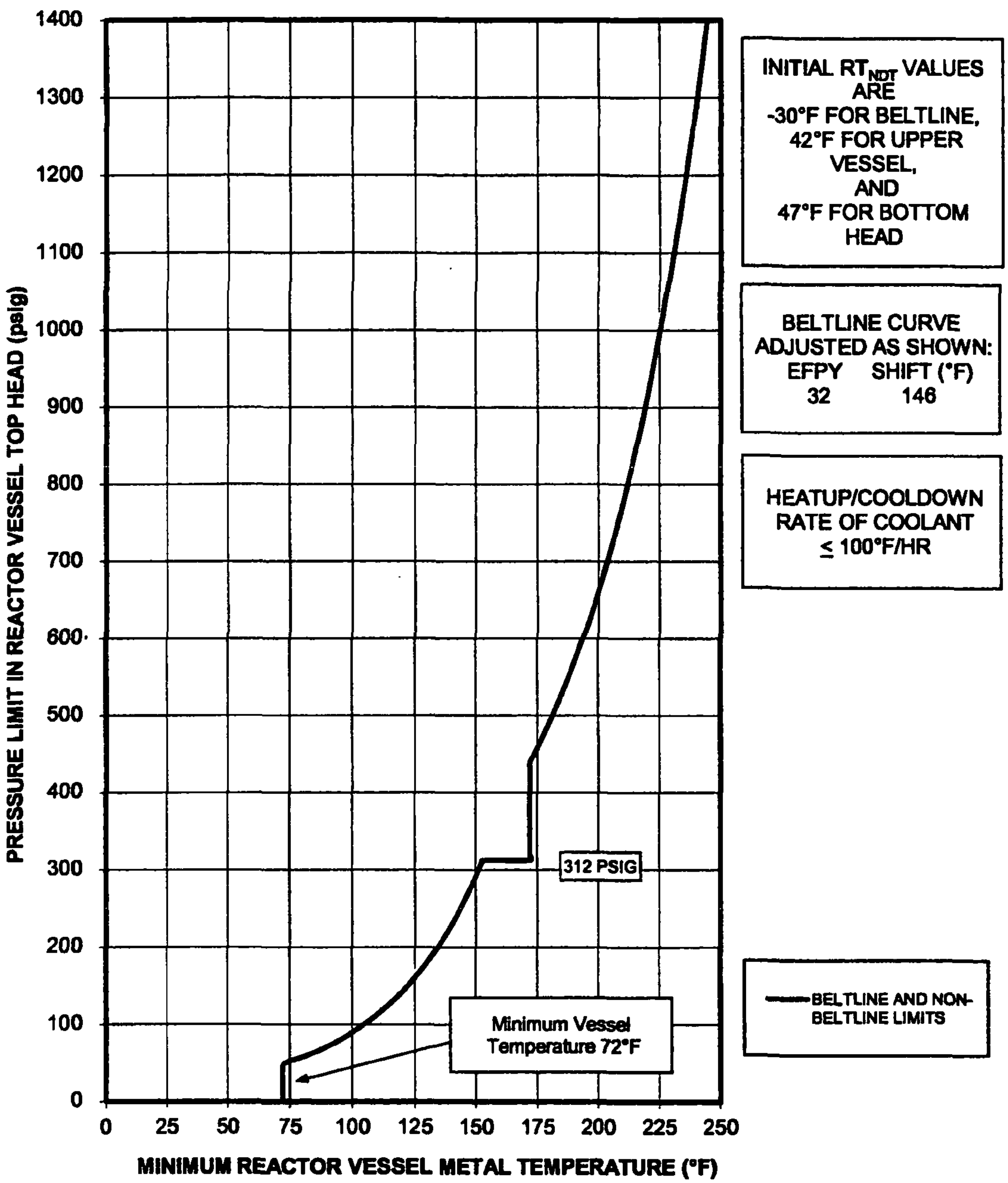


Figure 3.4.11-3 (Page 1 of 1)
Unit 1
P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing up to 32 EFPY

New Figure 3.4.11-3



ATTACHMENT 3

Revised Technical Specifications Pages

LASALLE COUNTY STATION
UNIT 1

Docket No. 50-373

Facility Operating License No. NPF-11

REVISED TS PAGES

3.4.11-6
3.4.11-7
3.4.11-8

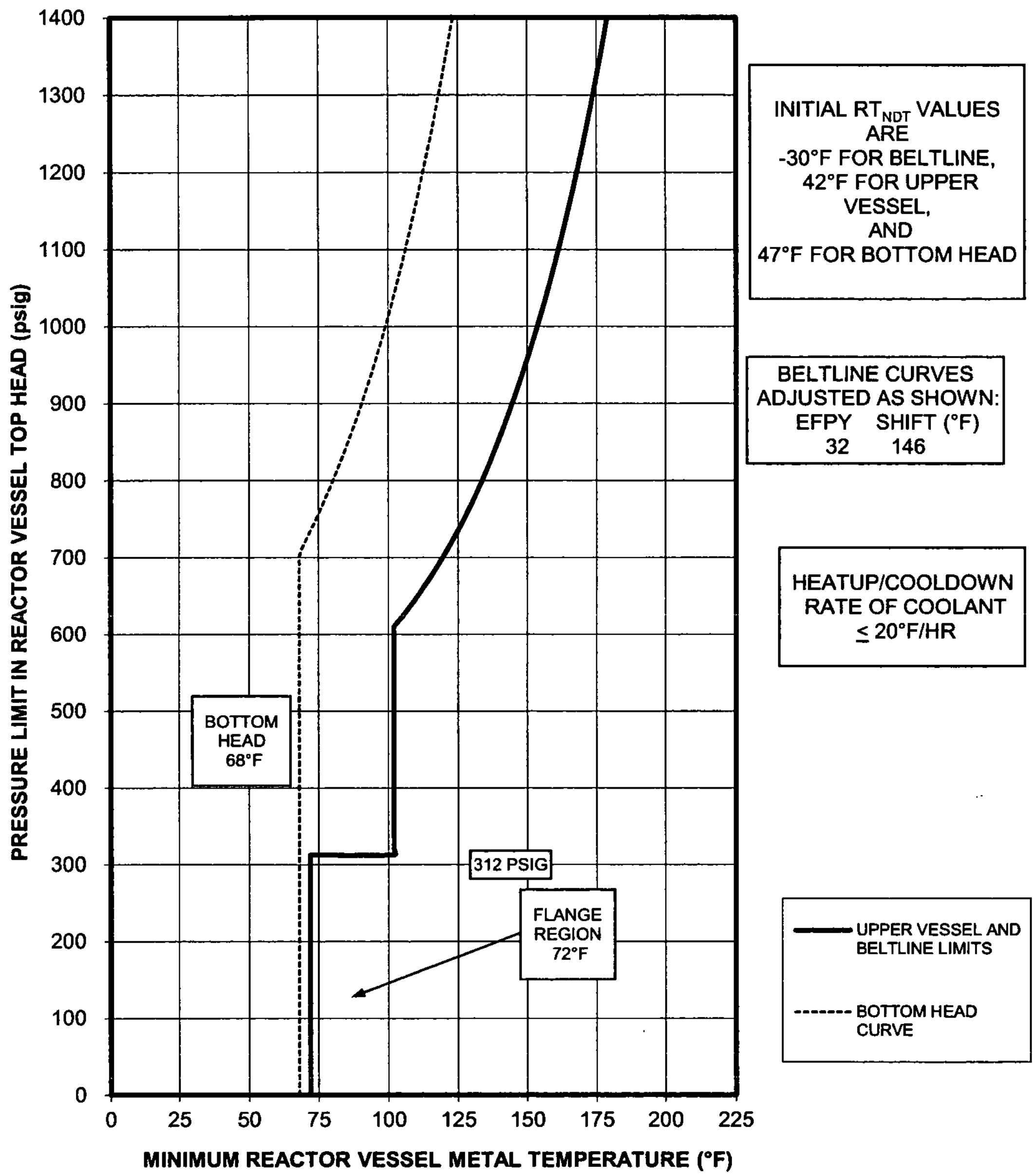


Figure 3.4.11-1 (Page 1 of 1)
Unit 1
P-T Curves for Hydrostatic or Leak Testing up to 32 EFPY

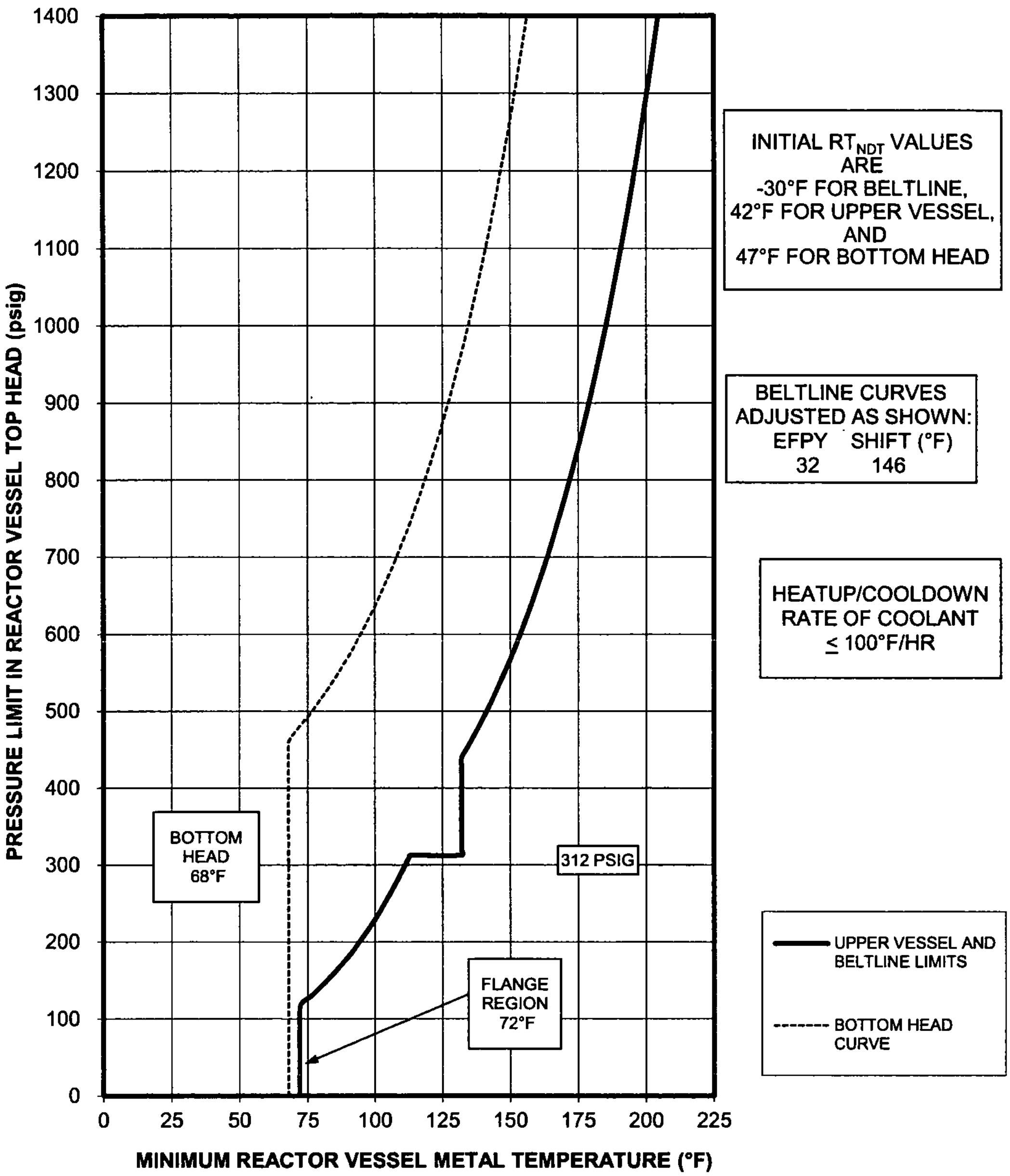


Figure 3.4.11-2 (Page 1 of 1)
Unit 1
P-T Curves for Heatup by Non-Nuclear Means, (Core Not Critical) Cooldown Following a Nuclear Shutdown and Low Power Physics Testing up to 32 EFPY

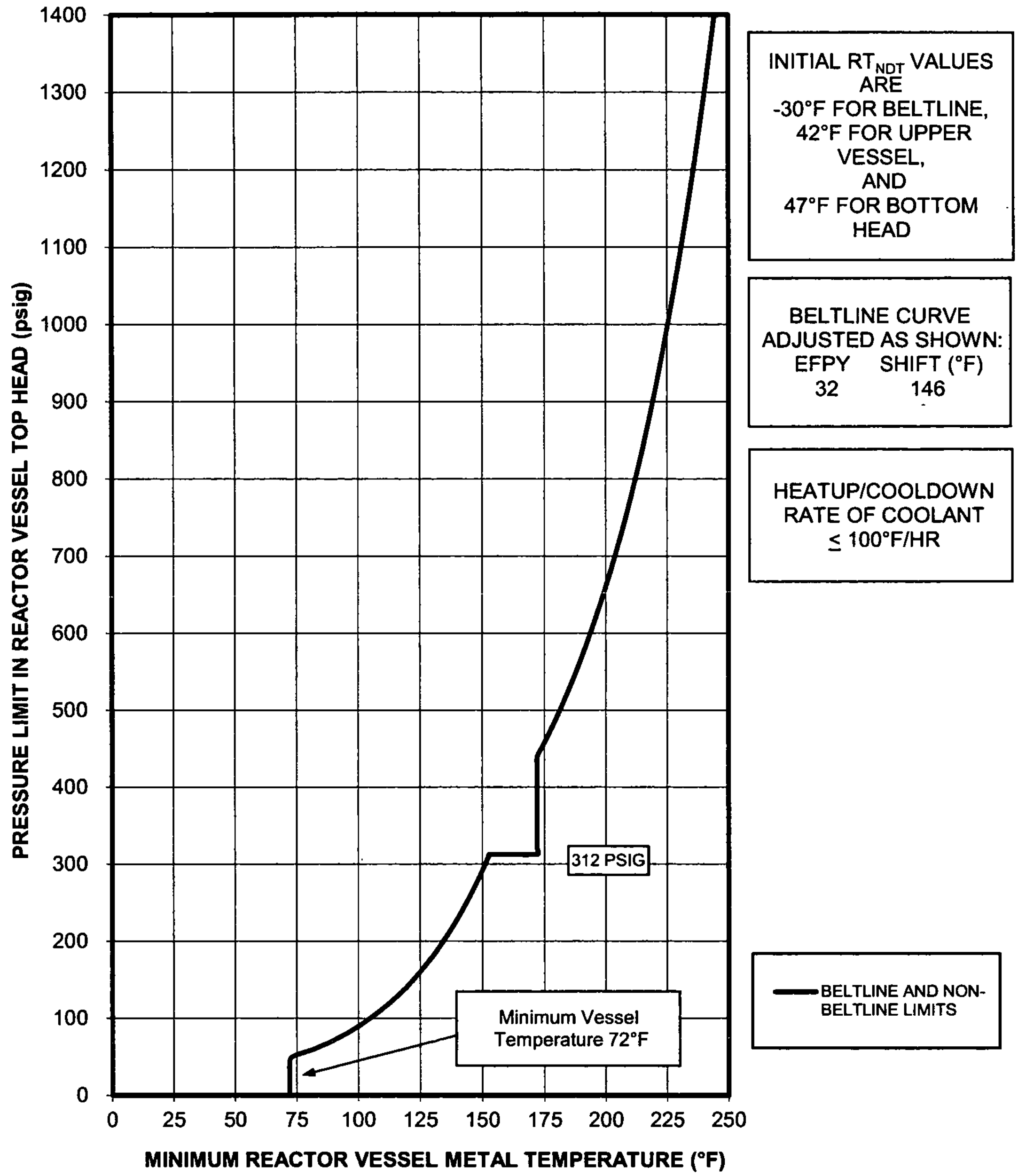


Figure 3.4.11-3 (Page 1 of 1)
Unit 1
P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing up to 32 EFPY