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PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

10 CFR 50.90
10 CFR Part 54

April 27, 2012

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: License Amendment Request – Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) requests proposed changes that would modify Technical Specification (TS) Section 1.1 ("Definitions"), Section 3.4.9 ("RCS Pressure and Temperature (P/T) Limits"), and Section 5.6 ("Reporting Requirements") by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR) at Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

The proposed changes have been reviewed by the PBAPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by April 27, 2013. Once approved, this amendment shall be implemented within 60 days of issuance.

Attachment 1 contains the evaluation of the proposed changes. Attachment 2 provides the marked up TS and Bases pages. The Bases pages are being provided for information only. Attachment 3 contains the proprietary PBAPS, Units 2 and 3 PTLR. Attachment 4 contains proprietary responses to requests for additional information related to this issue involving a separate nuclear power facility which have been customized for PBAPS, Units 2 and 3. Attachments 5 and 6 are non-proprietary versions of Attachments 3 and 4, respectively.

In accordance with 10 CFR 50.91, Exelon is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Attachments 3 and 4 transmitted herewith contain Proprietary Information.
When separated from attachments, this document is decontrolled.

There are no commitments contained in this submittal; however, commitment 23 associated with license renewal for PBAPS, as detailed in NUREG-1769, Appendix D, is completed with this submittal. Commitment 23 required submittal of the reactor pressure vessel P-T curves for 54 EFPY. These curves are included as part of Attachment 3 of this license amendment request.

Attachment 3 contains information proprietary to the Electric Power Research Institute (EPRI). EPRI requests that Attachment 3 be withheld from public disclosure in accordance with 10 CFR 2.390. Attachment 5 contains a non-proprietary version of the document. An affidavit supporting this request is also contained in Attachment 5.

Attachment 4 contains information proprietary to General Electric Hitachi and EPRI. General Electric Hitachi and EPRI request that Attachment 4 be withheld from public disclosure in accordance with 10 CFR 2.390. Attachment 6 contains a non-proprietary version of the document. Two (2) affidavits supporting this request are also contained in Attachment 6.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th of April 2012.

Respectfully,

Michael D. Jesse
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1:	Evaluation of Proposed Changes
Attachment 2:	Markup of Technical Specifications and Bases Pages
Attachment 3:	Proprietary Version - Peach Bottom Atomic Power Station Unit 2 and Unit 3 Pressure and Temperature Limits Report
Attachment 4:	Proprietary Version - Responses to Requests for Additional Information
Attachment 5:	Non-Proprietary Version - Peach Bottom Atomic Power Station Unit 2 and Unit 3 Pressure and Temperature Limits Report
Attachment 6:	Non-Proprietary Version - Responses to Requests for Additional Information

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, PBAPS
USNRC Senior Project Manager, PBAPS
R. R. Janati, Bureau of Radiation Protection
S. T. Gray, State of Maryland

Attachment 1
Evaluation of Proposed Changes

SUBJECT: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report

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Attachment 1
Evaluation of Proposed Changes

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 (Exelon) requests an amendment to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendment modifies the Technical Specifications (TS) by replacing the pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). Relocation of the P-T limit curves to the PTLR is consistent with the guidance provided in U.S. Nuclear Regulatory Commission (USNRC) approved GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1 ("GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves") (Reference 1). This topical report uses the guidelines provided in USNRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 2). Additionally, the TS changes in this license amendment request are consistent with the guidance provided in GL 96-03 as supplemented by Technical Specification Task Force (TSTF) traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 3), and the guidance contained in the August 4, 2011 USNRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6 ("Reporting Requirements"), as discussed below.

2.0 DETAILED DESCRIPTION

The proposed change includes the following TS revisions:

a) TS Section 1.1, "Definitions" – A new definition, "Pressure and Temperature Limits Report," is added. The wording for this definition is consistent with that in TSTF-419-A.

b) TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits" - The P-T curves and the associated TS wording have been deleted and replaced with references to the PTLR.

c) TS Section 5.6, "Reporting Requirements" – A new Section 5.6.7 ("Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)") has been added. The format and content of new Section 5.6.7 are consistent with that in TSTF-419-A, and the guidance contained in the August 4, 2011 USNRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6 ("Reporting Requirements"). This new section: (1) identifies the individual TS that address reactor coolant system P-T limits; (2) references the USNRC-approved topical report that documents PTLR methodologies in a complete citation; and (3) requires that the PTLR and any revision or supplement thereto be submitted to the USNRC.

Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Marked-up pages showing corresponding changes to the TS Bases are provided in Attachment 2 for information only. The TS Bases changes will be processed in accordance with the PBAPS, Units 2 and 3 Bases Control Program (TS 5.5.10).

The Attachment 3 PTLR 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. PBAPS, Units 2 and 3 are currently licensed to P-T curves for up to 32 effective full power years (EFPY); the analysis performed in this report establishes curves for up to 32 and 54 EFPY. The 1998 Edition of the ASME Boiler and Pressure Vessel Code including 2000 Addenda was used in this evaluation.

As documented in Section 4.0 of the Safety Evaluation Report for GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, licensees who choose to implement NEDC-33178P-A, Revision 1 as their facility's PTLR methodology must address one 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 PTLR incorporates a fluence calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the USNRC (Reference 5), and is in compliance with Regulatory Guide 1.190. The latest information from the BWRVIP Integrated Surveillance Program that is applicable to PBAPS, Units 2 and 3 has been utilized.

This license amendment request is being submitted to satisfy commitment 23 associated with license renewal for PBAPS, as detailed in NUREG-1769, Appendix D. In order to bound potential future operations at uprated conditions, the revised fluence represents a power level of 120% of original licensed thermal power, equal to 3951 MWt. Current Licensed Power level fluence (3514 MWt) was considered through 31.06 Effective Full Power Years (EFPY) for Unit 2 and 31.96 EFPY for Unit 3, which represents the end of Cycle 20. For the remainder of the plant's operating license, fluence at a power level of 120% of original licensed thermal power was assumed. Use of this fluence for the P-T curves for current operating conditions is conservative and is not dependent on the approval of an Extended Power Uprate application.

As stated previously, relocation of the P-T limit curves to the PTLR is consistent with the guidance provided in USNRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 2), TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 3), and the guidance contained in the August 4, 2011 USNRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6 ("Reporting Requirements").

3.0 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.

Historically, the P-T limit curves for BWRs have been contained in the TS, necessitating the submittal of license amendment requests to update the curves. This caused both the USNRC and licensees to expend resources that could otherwise be devoted to other activities.

Generic Letter 96-03 allows plants to relocate their P-T curves and associated numerical limits (such as heatup and cooldown rates) from the plant TS to a PTLR, which is a licensee-controlled document. As stated in Generic Letter 96-03, during the development of the improved Standard Technical Specifications (STS), a change was proposed to relocate the P-T limits currently contained in the plant TS to a PTLR. As one of the improvements to the STS, the USNRC staff agreed with the industry that the curves may be relocated outside the plant TS to a PTLR so that the licensee could maintain these limits efficiently. One of the prerequisites for having the PTLR option is that the P-T curves and limits be derived using methodologies approved by the USNRC, and that the associated licensing topical reports describing the approved methodologies be referenced in the plant TS.

The purpose of GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, is to provide the methodology developed by GE Hitachi Nuclear Energy (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 GE Licensing Topical Report NEDC-32983P-A, Revision 2.

In order to implement the PTLR, the analytical methods used to develop the P-T limits must be consistent with those previously reviewed and approved by the USNRC and must be referenced in the Administrative Controls section of the plant TS. GE Hitachi 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 PBAPS, Units 2 and 3 P-T curves have been developed in accordance with the NEDC-33178P-A, Revision 1, methodology as documented in the PTLR provided in Attachment 3.

The P-T curves included in the PTLR 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 and 54 effective full power years (EFPY). These P-T curves and a tabulation of the curves are provided in the PTLR. This report incorporates a fluence (E > 1 MeV) calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the USNRC, and is in compliance with Regulatory Guide 1.190. The latest information from the BWRVIP Integrated Surveillance Program that is applicable to PBAPS, Units 2 and 3 has been utilized.

The methodology used to generate the P-T curves in this report is presented in Section 3.0 of the PTLR. 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, K_{Ir}, 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 various intermediate and end of license EFPYs), 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 EFPYs. 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.

The proposed TS revisions associated with relocation of the P-T limits to a PTLR are consistent with the guidance provided in GL 96-03 as supplemented by TSTF-419-A, and the Reference 4 letter.

In the Reference 7 letter, 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 PTLR. Attachment 4 provides a response to these questions for PBAPS, Units 2 and 3.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

As discussed in the Safety Evaluation Report for GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, the USNRC 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. USNRC regulatory guidance related to P-T limit curves is found in Regulatory Guide 1.99, Revision 2 ("Radiation Embrittlement of Reactor Materials") and Standard Review Plan (NUREG-0800) Section 5.3.2 ("Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock").

Adoption of the USNRC-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. PBAPS, Units 2 and 3 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 6).

Generic Letter 96-03 provides regulatory guidance regarding relocation of P-T curves and associated numerical limits (such as heatup and cooldown rates) from plant TS to a PTLR (a licensee-controlled document). As stated in GL 96-03, a licensee requesting such a change must satisfy the following three criteria:

(1) Have USNRC-approved methodologies to reference in the TS,
(2) Develop a PTLR to contain the P-T limit curves, associated numerical limits, and any necessary explanation, and
(3) Modify applicable sections of the TS accordingly.

The USNRC-approved methodology of GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, has been adopted for preparation of the PBAPS, Units 2 and 3 P-T limit curves. As discussed in Section 5.0 ("Conclusion") of the Reference 1 USNRC Safety Evaluation Report:

> "The NRC staff concludes that BWROG LTR NEDC-33178P, Revision 1, satisfies the criteria in Attachment 1 in GL 96-03 and provides adequate methodology for BWR licensees to calculate P-T limit curves, given that licensees referencing this LTR comply with the conditions listed in Section 4.0 of this SE. Using this methodology and following the PTLR guidance in GL 96-03, as amended by NRC TSTF-419, BWR licensees will be able to relocate the P-T limit curves from TS to a PTLR, a licensee-controlled document."

As discussed previously, the PTLR incorporates a fluence calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the USNRC (Reference 5), and is in compliance with Regulatory Guide 1.190, thus satisfying the requirement contained in Section 5.0 of the USNRC Safety Evaluation Report.

Proposed revisions to applicable sections of the TS have been prepared and are provided in Attachment 2 to this submittal. These proposed TS changes are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A, and the guidance contained in the August 4, 2011 USNRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6 ("Reporting Requirements").

4.2 Precedent

The USNRC has approved similar license amendments to relocate P-T limit curves to a PTLR. Recent examples for boiling water reactor plants include:

1) Oyster Creek Nuclear Generating Station (License Amendment No. 269 issued by USNRC letter dated September 30, 2008 - ADAMS Accession No. ML082390685).

2) James A. Fitzpatrick Nuclear Power Plant (License Amendment No. 292 issued by USNRC letter dated October 3, 2008 - ADAMS Accession No. ML082630385).

4.3 No Significant Hazards Consideration

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

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

   Response: No

   The proposed changes modify the TS by replacing references to existing reactor vessel heatup and cooldown rate limits and P-T limit curves with references to the PTLR. The proposed amendment also adopts the USNRC-approved methodology of the GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, for the preparation of the PBAPS, Units 2 and 3 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 USNRC-approved methodology for calculating P-T limit curves and relocating those curves to the PTLR 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 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 changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

The change in methodology for calculating P-T limits and the relocation of those limits to the PTLR 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 changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes 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 USNRC-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 changes do not involve a significant reduction in a margin of safety.

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

4.4 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

Exelon 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 D. Coleman (BWR Owners' Group) to U.S. Nuclear Regulatory Commission, "Submittal of GE BWROG Topical Report NEDC-33178P-A, 'General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves'," ML092370486, dated July 29, 2009

2) Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996

3) Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," dated August 4, 2003

4) Letter from J. Jolicoeur (U.S. Nuclear Regulatory Commission) to Technical Specification Task Force, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR (Core Operating Limits Report),' TSTF-408, Revision 1, 'Relocation of LTOP (Low-Temperature Overpressure Protection) Enable Temperature and PORV (Power-Operated Relief Valve) Lift Setting to the PTLR (Pressure-Temperature Limits Report),' and TSTF-419, Revision 0, 'Revise PTLR Definition and References in ISTS (Improved Standard Technical Specification) 5.6.6, RCS (Reactor Coolant System) PTLR'," ML110660285, dated August 4, 2011

5) Letter from G. Stramback (GE) to U.S. Nuclear Regulatory Commission, "Accepted Version of GE Licensing Topical Report NEDC-32983P-A, Revision 2 (TAC No. MC3788)," ML072480116, dated February 1, 2006

6) Letter from G. Wunder (U.S. Nuclear Regulatory Commission) to J. Skolds (Exelon Generation Company, LLC), "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendment RE: Revision to the Reactor Pressure Vessel Material Surveillance Program (TAC Nos. MB7006 and MB7007)," dated November 4, 2003

7) Letter from M. A. Krupa (Entergy Operations, Inc.) to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding Extended Power Uprate," ML110540540, dated February 23, 2011

ATTACHMENT 2

Markup of Technical Specifications and Bases Pages

Revised Pages (Units 2 and 3)

TS Page 1.1-5
3.4-21
3.4-22
3.4-23
3.4-24
3.4-25
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3.4-27
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B 3.4-43
B 3.4-44
B 3.4-45
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B 3.4-47
B 3.4-48
B 3.4-49
B 3.4-50
B 3.4-51

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3514 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.
RECENTLY IRRADIATED FUEL	RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours. When using this definition to suspend the Applicability of LCOs, secondary containment ground-level hatches H15, H16, H17, H18, H19, and H33 shall be closed during the movement of any irradiated fuel in Secondary Containment.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within ~~limits.~~ the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	------------------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 3.4.9-1 and 3.4.9-2~~ [within the limits specified in the PTLR]; and b. RCS heatup and cooldown rates are ~~≤ 100°F in any 1 hour period.~~ [within the limits specified in the PTLR.]	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~Figure 3.4.9-3.~~ [the PTLR.]	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	--------------------NOTE-------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~≤ 145°F.~~ [within the limits specified in the PTLR.]	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	--------------------NOTE-------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. --- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~≤ 50°F.~~ [within the limits specified in the PTLR.]	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	-------------------NOTE------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. --- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.6	-------------------NOTE------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.7	-------------------NOTE------------------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F~~. within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

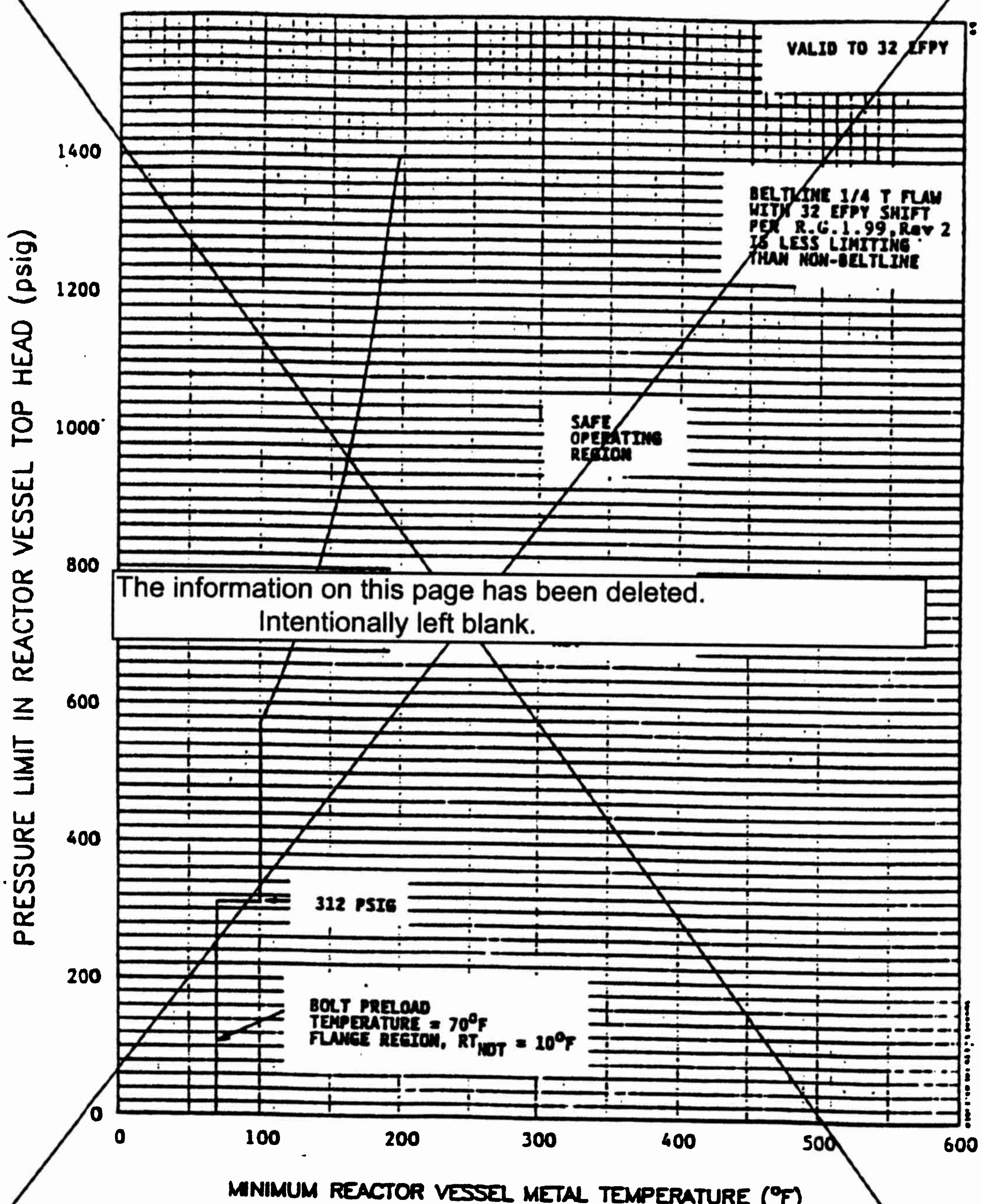

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Figure 3.4.9-1 (page 1 of 1)

Temperature/Pressure Limits for
Inservice Hydrostatic and Inservice Leakage Tests

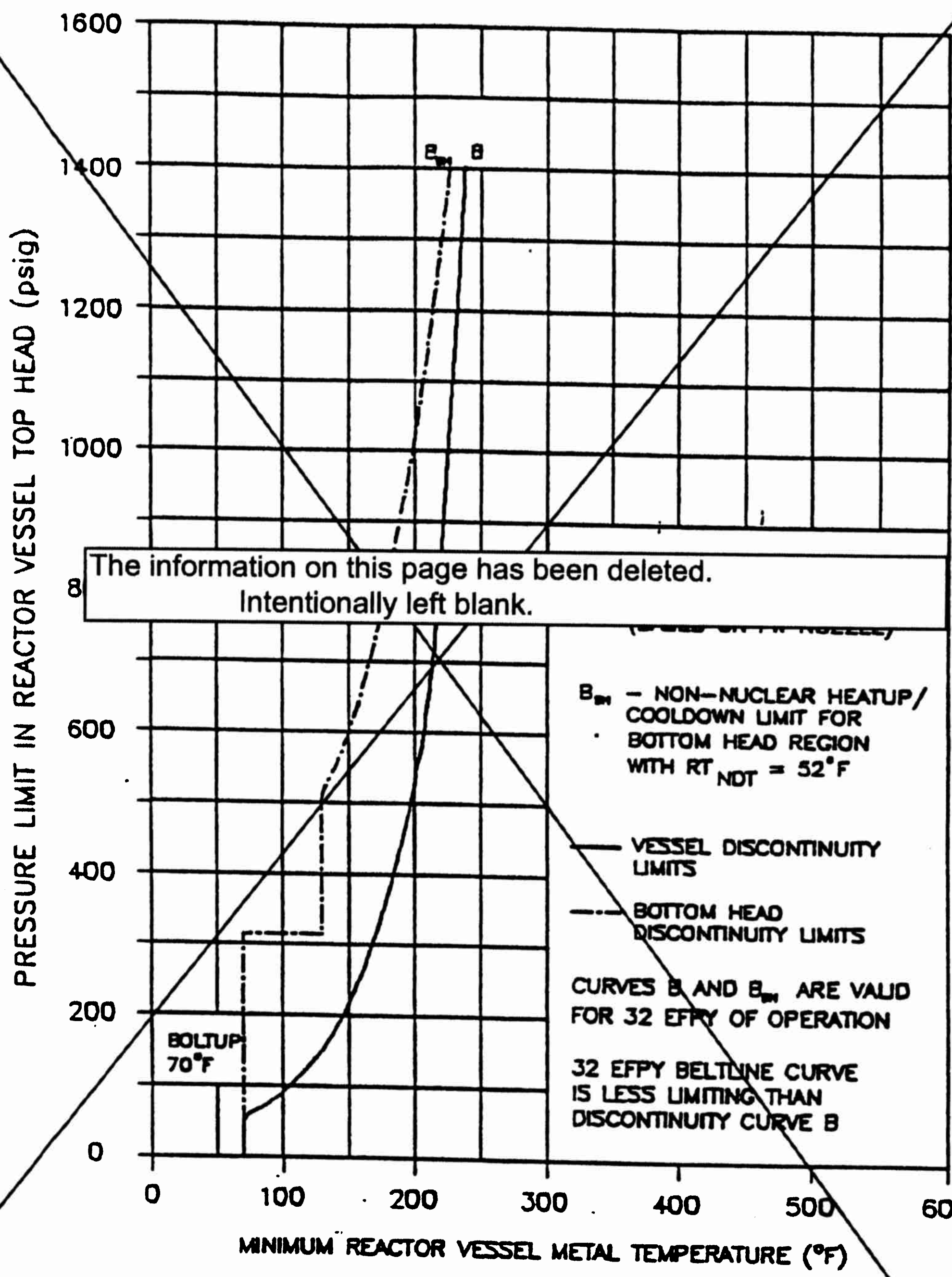

Figure 3.4.9-2 (page 1 of 1)

Temperature/Pressure Limits for
Non-Nuclear Heatup and Cooldown Following a Shutdown

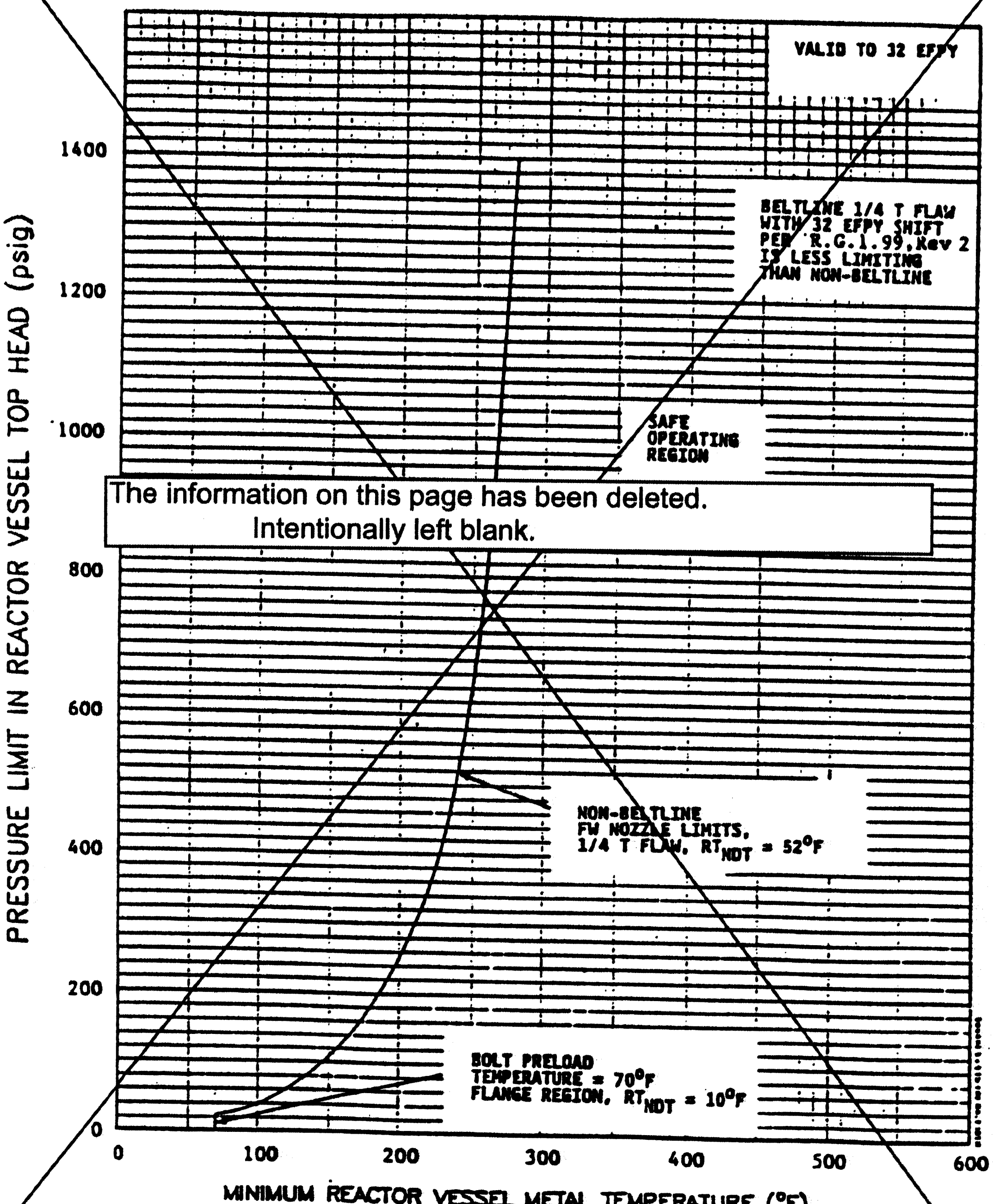

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Figure 3.4.9-3 (page 1 of 1)

Temperature/Pressure Limits for Criticality

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis";

8. PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"; and

9. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology And Reload Applications," August 1996.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

i) Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

ii) Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

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

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 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 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Ref. 10)

~~The Specification~~ contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and also limits the maximum rate of change of reactor coolant temperature. The criticality 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 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, abnormal operational transients, and system hydrostatic tests. It mandates the use of the 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 the UFSAR (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

(continued)

BASES

BACKGROUND (continued)

The P/T limit 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 limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than 60°F above the adjusted reference temperature of the reactor vessel material in the region that is controlling (reactor vessel flange region).

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the reactor pressure vessel, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) 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 nonductile failure of the reactor pressure vessel, a condition that is unanalyzed. ~~References 7 and 8 approved the curves and limits specified in this section.~~ 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in ~~Figures 3.4.9-1 and 3.4.9-2,~~ [the PTLR] and heatup or cooldown rates are ~~≤ 100°F during RCS heatup, cooldown, and inservice leak and hydrostatic testing~~ [within the limits specified in the PTLR];

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~≤ 145°F~~ [within the limits specified in the PTLR] during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ~~≤ 50°F~~ [within the limits specified in the PTLR] during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in ~~Figure 3.4.9-3,~~ [the PTLR,] prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ~~≥ 70°F~~ [within the limits specified in the PTLR] 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 controls the thermal gradient through the vessel wall and is used as input for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

(continued)

BASES

LCO (continued)

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

<u>A.1 and A.2</u>

in the PTLR

Operation outside the P/T limits while in MODES 1, 2, and 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, 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 vessel beltline.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more 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

in the PTLR

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.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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 > 212°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 3.4.9.1

the PTLR

Verification that operation is within limits is required when RCS pressure and temperature conditions are undergoing planned changes. Plant procedures specify the pressure and temperature monitoring points to be used during the performance of this Surveillance. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

in the PTLR

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.2 (continued)

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 the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

in the PTLR

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 ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 9) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Note also states the SR is only required to be met during a recirculation pump startup, since this is when the stresses occur.

in the PTLR

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

in the PTLR

~~The~~ flange temperatures must be verified to be above the limits 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, 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 to ensure the temperature is within the limits specified.

in the PTLR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated after RCS temperature ≤ 80°F in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated after RCS temperature ≤ 100°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the limits specified.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. UFSAR, Section 4.2.6 and Appendix K.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.

(continued)

BASES

REFERENCES (continued)

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

7. R.E. Martin (NRC) letter to G.A. Hunger (PECo), Amendment No. 153 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station Unit No. 2, dated October 25, 1989.

8. R.J. Clark (NRC) letter to G.J. Beck (PECo), Amendment Nos. 162 and 164 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3, dated June 27, 1991.

DELETED

9. UFSAR, Section 14.5.6.2.

10. PRESSURE AND TEMPERATURE LIMITS REPORT.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3514 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.
RECENTLY IRRADIATED FUEL	RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours. When using this definition to suspend the Applicability of LCOs, secondary containment ground-level hatches H20, H21, H22, H23, H24, and H34 shall be closed during the movement of any irradiated fuel in Secondary Containment.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within ~~limits.~~ the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. ---------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed if this Condition is entered. ---------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------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 3.4.9-1 and 3.4.9-2~~ [within the limits specified in the PTLR]; and b. RCS heatup and cooldown rates are ~~≤ 100°F in any 1 hour period.~~ [within the limits specified in the PTLR.]	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~Figure 3.4.9-3.~~ the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~≤ 145°F.~~ within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~≤ 50°F.~~ within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	------------------NOTE------------------ Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.6	------------------NOTE------------------ Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.7	------------------NOTE------------------ Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

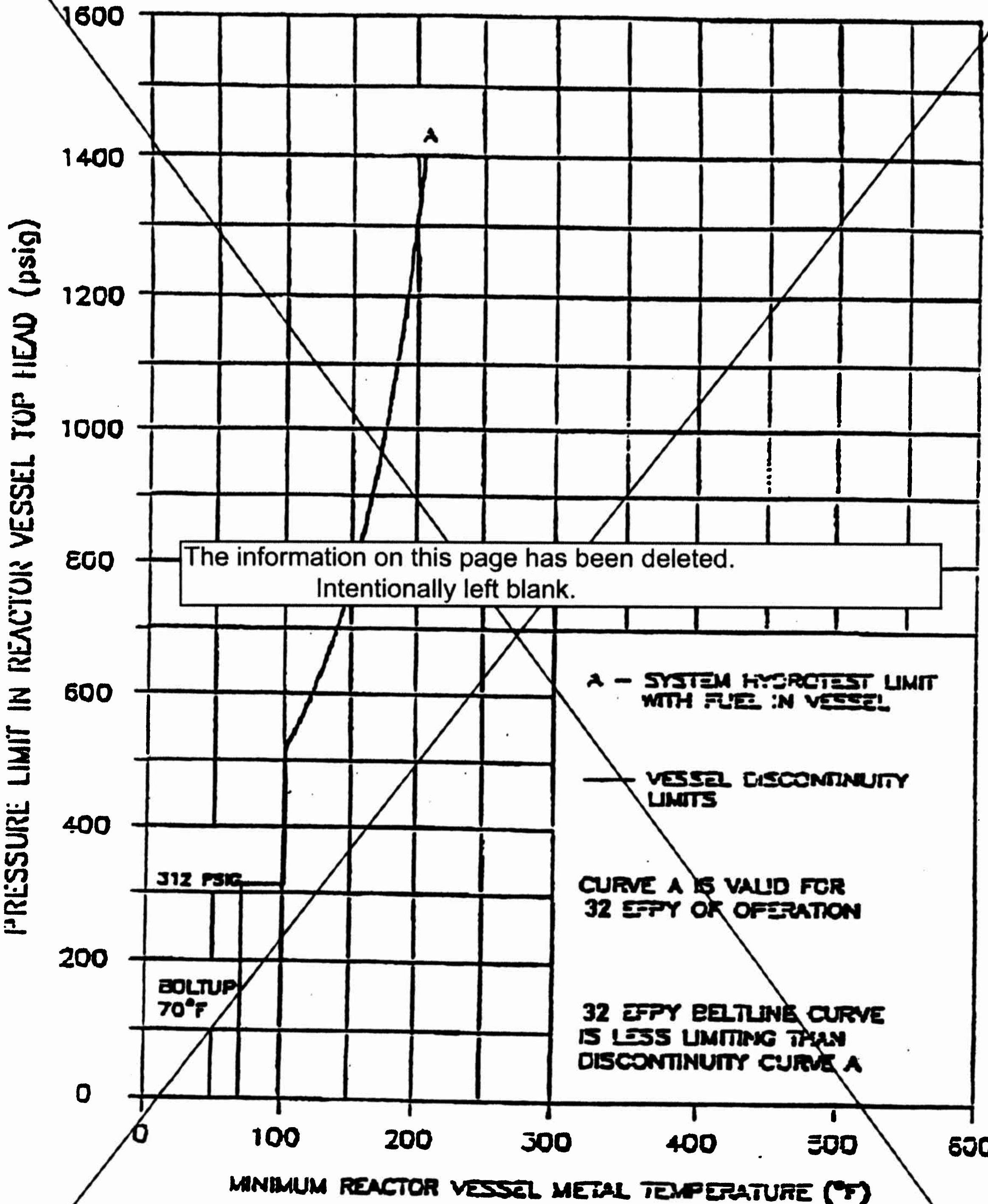

Figure 3.4.9-1 (page 1 of 1)

Temperature/Pressure Limits for
Inservice Hydrostatic and Inservice Leakage Tests

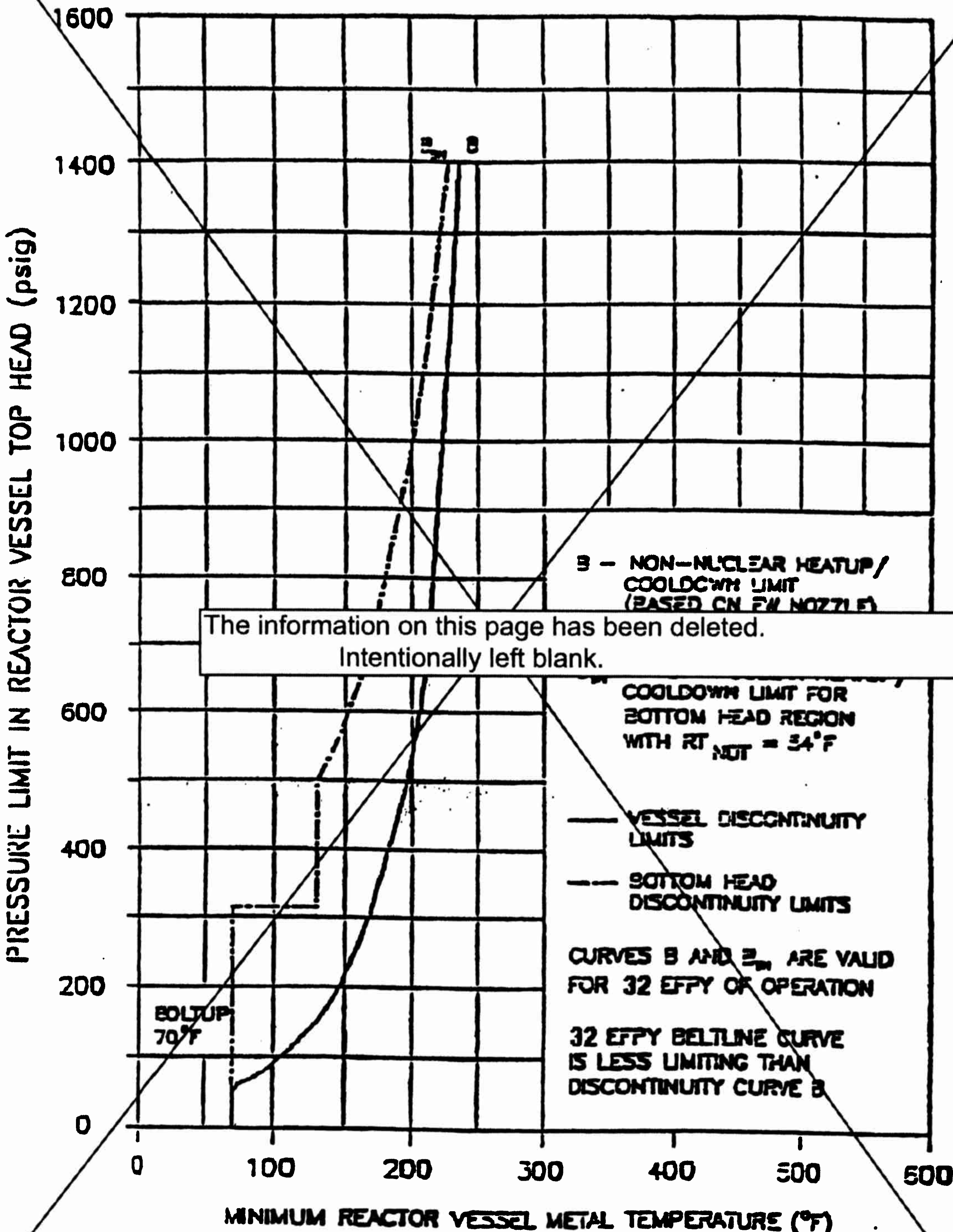

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Figure 3.4.9-2 (page 1 of 1)

Temperature/Pressure Limits for
Non-Nuclear Heatup and Cooldown Following a Shutdown

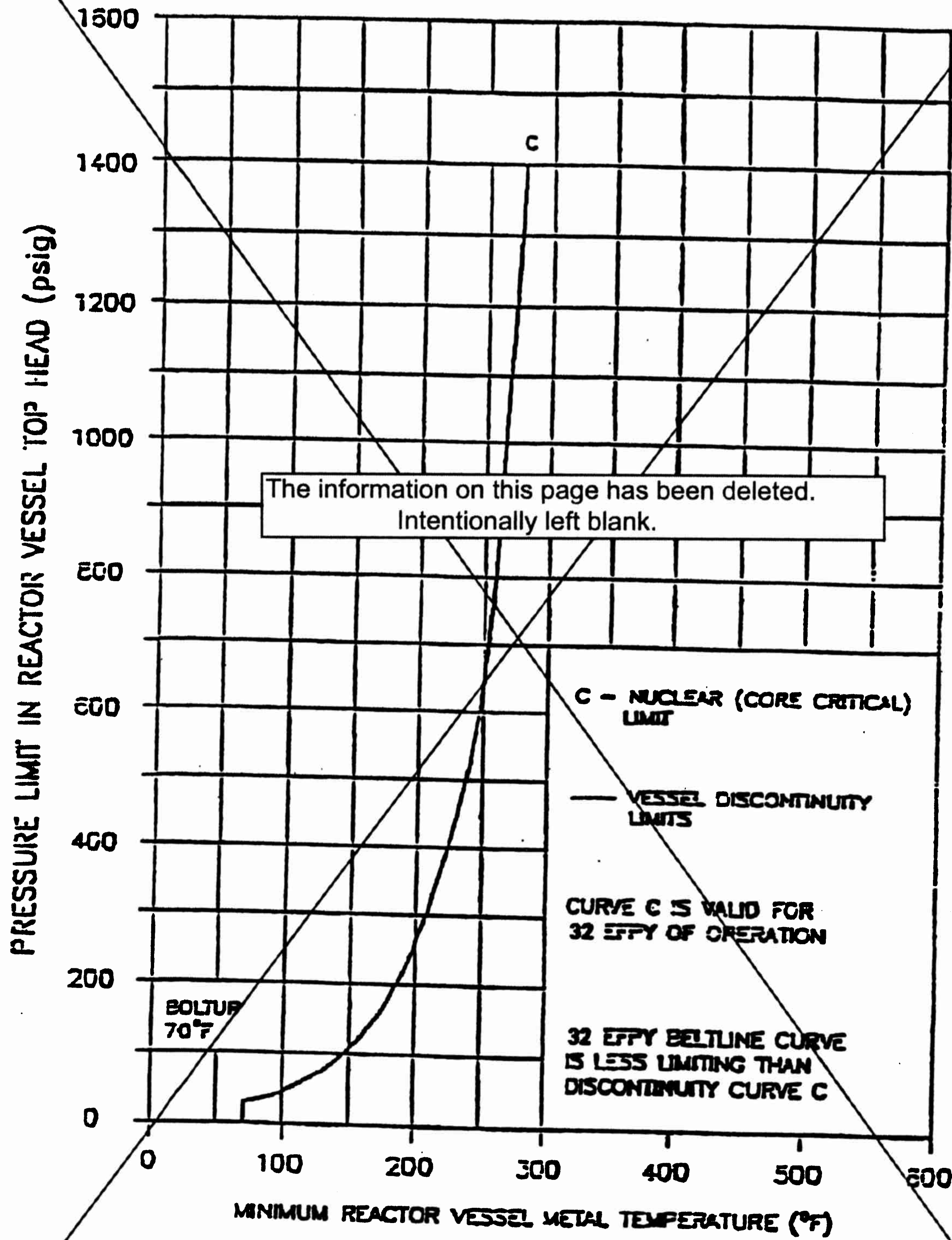

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Figure 3.4.9-3 (page 1 of 1)

Temperature/Pressure Limits for Criticality

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis";

8. PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"; and

9. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology And Reload Applications," August 1996.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
  i) Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"
  ii) Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:
  i) NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, June 2009

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 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 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Ref. 9)] ~~The Specification~~ contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and also limits the maximum rate of change of reactor coolant temperature. The criticality 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 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, abnormal operational transients, and system hydrostatic tests. It mandates the use of the 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 the UFSAR (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

(continued)

BASES

BACKGROUND (continued)

The P/T limit 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 limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than 60°F above the adjusted reference temperature of the reactor vessel material in the region that is controlling (reactor vessel flange region).

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the reactor pressure vessel, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) 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 nonductile failure of the reactor pressure vessel, a condition that is unanalyzed. ~~Reference 7 approved the curves and limits specified in this section.~~ 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in ~~Figures 3.4.9-1 and 3.4.9-2,~~ [the PTLR] and heatup and cooldown rates are ~~≤ 100°F during RCS heatup, cooldown, and inservice leak and hydrostatic testing~~ [within the limits specified in the PTLR];

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~≤ 145°F~~ [within the limits specified in the PTLR] during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ~~≤ 50°F~~ [within the limits specified in the PTLR] during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in ~~Figure 3.4.9-3~~ [the PTLR,] prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ~~> 70°F~~ [within the limits specified in the PTLR] 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 controls the thermal gradient through the vessel wall and is used as input for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

(continued)

BASES

LCO (continued)

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

in the PTLR

Operation outside the P/T limits while in MODES 1, 2, and 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, 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 vessel beltline.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more 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

in the PTLR

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.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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 > 212°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 3.4.9.1

the PTLR

Verification that operation is within limits is required when RCS pressure and temperature conditions are undergoing planned changes. Plant procedures specify the pressure and temperature monitoring points to be used during the performance of this Surveillance. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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.

SR 3.4.9.2

in the PTLR

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.2 (continued)

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 the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits [in the PTLR] ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Note also states the SR is only required to be met during a recirculation pump startup, since this is when the stresses occur.

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

Limits [in the PTLR] 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

in the PTLR

The flange temperatures must be verified to be above the limits 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, 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 to ensure the temperature is within the limits specified.

in the PTLR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated after RCS temperature ≤ 80°F in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated after RCS temperature ≤ 100°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the limits specified.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. UFSAR, Section 4.2.6 and Appendix K.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.

(continued)

BASES

REFERENCES (continued)

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

7. R.J. Clark (NRC) letter to G.J. Beck (PECo), Amendment Nos. 162 and 164 to Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station Unit Nos. 2 and 3, dated June 27, 1991.

DELETED

8. UFSAR, Section 14.5.6.2.

9. PRESSURE AND TEMPERATURE LIMITS REPORT.

ATTACHMENT 5

Non-Proprietary Version - Peach Bottom Atomic Power Station Unit 2 and Unit 3
Pressure and Temperature Limits Report

Attachment 1

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

NEIL WILMSHURST
Vice President and Chief Nuclear Officer

Ref. EPRI Project Number 669

April 17, 2012

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Andrew Hon

Subject: Request for Withholding of the following Proprietary Information Included in:

Exelon/Peach Bottom Atomic Power Station (PBAPS) Unit 2 & Unit 3
Pressure and Temperature Limits Report (PTLR)
up to 32 and 54 Effective Full-Power Years
Revision 0

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Report and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence for informational purposes regarding a submittal to the NRC by Exelon Corporation. The Proprietary Information is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Proprietary Information provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 704-595-2732. Questions on the content of the Proprietary Information should be directed to Randy Stark of EPRI at (650) 855-2122.

Sincerely,

c: Sheldon Stuchell, NRC (Sheldon.stuchell@nrc.gov)

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ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In::

Exelon/Peach Bottom Atomic Power Station (PBAPS) Unit 2 & Unit 3
Pressure and Temperature Limits Report (PTLR)
up to 32 and 54 Effective Full-Power Years
Revision 0

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information. EPRI made a substantial economic investment to develop the Proprietary Information and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

c. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

> "'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

e. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 4-17-2012.

[signature]

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 17th day of April, 2012, by Neil Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016.

EXELON/PEACH BOTTOM ATOMIC POWER STATION (PBAPS) UNIT 2 & UNIT 3

Pressure and Temperature Limits Report (PTLR)

up to 32 and 54 Effective Full-Power Years

Revision 0

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1.0 Purpose

The purpose of the PBAPS Unit 2 and Unit 3 Pressure and Temperature Limits Reports (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing;
2. RCS Heatup and Cooldown rates;
3. Reactor Pressure Vessel (RPV) to RCS coolant ΔT requirements during Recirculation Pump startups;
4. RPV bottom head coolant temperature to RPV coolant temperature ΔT requirements during Recirculation Pump startups;
5. RPV head flange bolt-up temperature limits.

This report has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.7, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)".

2.0 Applicability

This report is applicable to the PBAPS Unit 2 and Unit 3 RPVs for up to 32 and 54 Effective Full-Power Years (EFPY).

The following TS is affected by the information contained in this report:

TS 3.4.9 RCS Pressure and Temperature (P/T) Limits

3.0 Methodology

The limits in this report were derived from the NRC-approved methods listed in TS 5.6.7, using the specific revisions listed below:

1. The neutron fluence was calculated per *Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation*, NEDC-32983P-A, Revision 2, January 2006, approved in Reference 6.1.

2. The pressure and temperature limits were calculated per *GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves*, NEDC-33178P-A, Revision 1, June 2009, approved in Reference 6.2.

3. This revision of the pressure and temperature limits is to incorporate the following changes:

   - Initial issuance of the PTLR
   - Application of GEH Topical Report for P-T Curves
   - Fluence application for operation at 3951 Mwt

As discussed in Appendix A, PBAPS Unit 2 and Unit 3 participate in the BWRVIP Integrated Surveillance Program (ISP). Unit 2 is a host plant and is scheduled to remove its second capsule at 33.7 EFPY. The third Unit 2 capsule is classified as a standby capsule. As Unit 3 is not a host plant, the surveillance capsules have an ISP status designation of deferred (standby) per Reference 6.4.

The adjusted reference temperature (ART) values for 32 and 54 EFPY included in Appendix B (Unit 2) and Appendix C (Unit 3) are developed considering the latest ISP published surveillance data available that is representative of the applicable materials in the PBAPS Unit 2 and Unit 3 RPV beltline (Reference 6.3). The surveillance data used in the Unit 2 ART calculations is obtained from actual Unit 2 RPV test specimens; the surveillance data used in the Unit 3 ART calculations is not obtained from actual PBAPS Unit 3 RPV test specimens. For Unit 2, the ISP materials did not have the limiting ART. For Unit 3, the ISP plate material has the limiting ART; however, this value is not considered in the development of the P-T curves because the ISP material is not the identical heat to the material in the Unit 3 RPV.

Should actual surveillance capsules be withdrawn and tested from the PBAPS Unit 3 RPV (e.g., status change to be an ISP host plant under the BWRVIP ISP), compliance with 10 CFR 50 Appendix H requirements on reporting test results and evaluations on the effects to plant operations parameters (e.g., P-T limits, hydrostatic and leak test conditions) will be in accordance with Section 3 of Reference 6.3.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59, provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance.

4.0 Operating Limits

The pressure-temperature (P-T) curves provided in this report represent steam dome pressure versus minimum vessel metal temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

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 (core not critical), referred to as Curve B; and (c) core critical operation, referred to as Curve C.

Complete P-T curves were developed for 32 and 54 EFPY. The P-T curves are provided in Figures 1 through 12, and a tabulation of the curves is included in Table 1 (32 EFPY) and Table 2 (54 EFPY) for Unit 2 and in Table 3 (32 EFPY) and Table 4 (54 EFPY) for Unit 3.

Other temperature limits applicable to the RPV are:

- Heatup and Cooldown rate limit during Hydrostatic and Class 1 Leak Testing: ≤ 20 °F/hour.
- Normal Operating Heatup and Cooldown rate limit: ≤ 100 °F/hour.
- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: ≤ 145 °F.
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: ≤ 50 °F.
- RPV flange and adjacent shell temperature limit: ≥ 70 °F.

5.0 Discussion

The computer codes described in References 6.1 and 6.2 were used in the development of the P-T curves for PBAPS Units 2 and 3.

The method for determining the Initial Reference Temperature of Nil-Ductility Transition (RT_{NDT}) for all vessel materials is defined in Section 4.1.2 of Reference 6.2. Initial RT_{NDT} values for all vessel materials considered are presented in tables in Appendix B (for Unit 2) and Appendix C (for Unit 3) of this report.

For PBAPS Unit 2, the limiting surveillance material, plate heat C2761-2, was considered using Procedure 2 as defined in Appendix I of Reference 6.2. This procedure was used because the target vessel material and the surveillance material are not identical heats. However, C2761-2 is a beltline plate in the PBAPS Unit 2 vessel, with only one set of test results. The surveillance data should be considered after a second surveillance capsule is tested and credible surveillance data is available.

For PBAPS Unit 3, the limiting surveillance material, plate heat B0673-1, was considered using Procedure 2 as defined in Appendix I of Reference 6.2. This procedure was used because the target vessel material and the surveillance material are not identical heats.

For PBAPS Unit 2, there are four (4) thickness discontinuities in the vessel: (1) between the bottom head dollar plate and the bottom head torus, (2 and 3) between the bottom head torus and the support skirt attachment at two locations, and (4) between the bottom head torus and Shell Ring #1. The P-T curves defined in Section 4.3 of Reference 6.2 are based upon an RT_{NDT} of 50°F for the bottom head Curves A and C, 46°F for the bottom head Curve B, and 48°F for the upper vessel. The 32 EFPY beltline curves are based on an ART of 50°F, and the 54 EFPY beltline curves are based on an ART of 64.2°F. Curves based on these temperatures bound the requirements due to the thickness discontinuities.

For PBAPS Unit 3, there are four (4) thickness discontinuities in the vessel: (1) between the bottom head dollar plate and the bottom head torus, (2 and 3) between the bottom head torus and the support skirt attachment at two locations, and (4) between the bottom head torus and Shell Ring #1. The P-T curves defined in Section 4.3 of Reference 6.2 are based upon an RT_{NDT} of 54°F for the bottom head Curves A and C, 42°F for the bottom head Curve B, and 44°F for the upper vessel. The 32 EFPY beltline curves are based on an ART of 78.3°F, and the 54 EFPY beltline curves are based on an ART of 88.6°F. Curves based on these temperatures bound the requirements due to the thickness discontinuities.

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2 (RG 1.99) provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper and nickel values, except for the N16 nozzle, were obtained from plant-specific vessel purchase order records, Certified Material Test Reports (CMTRs). The N16 nozzle is fabricated from Alloy 600 material that does not require evaluation for fracture toughness, and was evaluated using the limiting material properties (chemistry and initial RT_{NDT}) of the adjoining Shell Ring #2. The copper (Cu) and nickel (Ni) values were used with Tables 1 and 2 of RG 1.99, to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds and plates, respectively. ART values for 32 and 54 EFPY are presented in Appendix B for Unit 2 and Appendix C for Unit 3. The ART tables also include plant-specific materials considering best estimate chemistry values obtained from Reference 6.3.

The P-T curves for the non-beltline region were conservatively developed for a Boiling Water Reactor Product Line 6 (BWR/6) with nominal inside diameter of 251 inches. The analysis is considered appropriate for PBAPS Unit 2 and Unit 3 because the plant-specific geometric values are bounded by the generic analysis for the large BWR/6. The generic value was adapted to the conditions at PBAPS using plant-specific RT_{NDT} values for the reactor pressure vessel.

The peak RPV ID fluence used in the P-T curve evaluation for Unit 2 at 32 EFPY is 9.54e17 n/cm^2 and at 54 EFPY is 1.61e18 n/cm^2. These values were calculated using methods that comply with the guidelines of RG 1.190, (as discussed in Reference 6.1). This fluence applies to the lower-intermediate shell plates and longitudinal welds. The fluence is adjusted for the lower plates, associated longitudinal welds, and the girth weld based upon an attenuation factor of 0.764; hence, the peak ID surface fluence for these components is 7.29e17 n/cm^2 for 32 EFPY and 1.23e18 n/cm^2 for 54 EFPY. Similarly, the fluence is adjusted for the N16 nozzle based upon an attenuation factor of

0.353; hence the peak ID surface fluence used for this component is 3.37e17 n/cm^2 for 32 EFPY and 5.69e17 n/cm^2 for 54 EFPY.

The peak RPV ID fluence used in the P-T curve evaluation for Unit 3 at 32 EFPY is 9.07e17 n/cm^2 and at 54 EFPY is 1.53e18 n/cm^2. These values were calculated using methods that comply with the guidelines of RG 1.190, (as discussed in Reference 6.1). This fluence applies to the lower-intermediate shell plates and longitudinal welds. The fluence is adjusted for the lower plates, associated longitudinal welds, and the girth weld based upon an attenuation factor of 0.62; hence, the peak ID surface fluence for these components is 5.62e17 n/cm^2 for 32 EFPY and 9.48e17 n/cm^2 for 54 EFPY. PBAPS Unit 3 has an additional intermediate shell in the beltline region. The fluence for the peak location in this shell is based upon an attenuation factor of 0.624; hence, the peak ID surface fluence for these components is 5.65e17 n/cm^2 for 32 EFPY and 9.54e17 n/cm^2 for 54 EFPY. Similarly, the fluence is adjusted for the N16 nozzle based upon an attenuation factor of 0.372; hence the peak ID surface fluence used for this component is 3.37e17 n/cm^2 for 32 EFPY and 5.69e17 n/cm^2 for 54 EFPY.

The P-T curves for the heatup and cooldown operating conditions at a given EFPY apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4T location is assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stress at the 1/4T location. This approach is conservative because irradiation effects cause the allowable toughness, K_{Ir}, at 1/4T to be less than that at 3/4T for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, well above the heatup/cooldown curve limits.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves specify a coolant heatup and cooldown temperature rate of ≤ 100°F/hr 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. For the hydrostatic pressure and leak test curve

(Curve A), a coolant heatup and cooldown temperature rate of ≤ 20°F/hr must be maintained. The P-T limits and corresponding heatup/cooldown rates of either Curve A or B may be applied while achieving or recovering from test conditions. Curve A applies during pressure testing and when the limits of Curve B cannot be maintained.

For PBAPS Unit 2, plate heat C2873-1 is the limiting material for the beltline region for 32 and 54 EFPY. The initial RT_{NDT} for plate heat C2873-1 material is -6°F. The generic pressure test P-T curve is applied to the PBAPS Unit 2 beltline curve by shifting the P vs. (T - RT_{NDT}) values to reflect the ART value of 50°F for 32 EFPY and 64.2°F for 54 EFPY. Using the fluence discussed above, the P-T curves are not beltline limited for Curves A, B, or C, for 32 or 54 EFPY.

For PBAPS Unit 3, plate heat C2773-2 is the limiting material for the beltline region for 32 and 54 EFPY. The initial RT_{NDT} for plate heat C2773-2 material is 10°F. The generic pressure test P-T curve is applied to the PBAPS Unit 3 beltline curve by shifting the P vs. (T - RT_{NDT}) values to reflect the ART value of 78.3°F for 32 EFPY and 88.6°F for 54 EFPY. Using the fluence discussed above, the P-T curves are not beltline limited for Curves A, B, or C, for 32 EFPY. For 54 EFPY, Curve A is beltline limited at pressures above 1070 psig and Curves B and C are beltline limited at pressures above 1160 psig.

In order to ensure that the limiting vessel discontinuity has been considered in the development of the P-T curves, the methods in Sections 4.3.2.1 and 4.3.2.2 of Reference 6.2 for the non-beltline and beltline regions, respectively, are applied.

6.0 References

6.1 *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.

6.2 *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.

6.3 *BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*, BWRVIP-135, Revision 1, EPRI, Palo Alto, CA, June 2007 (EPRI Proprietary)

6.4 *BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,* BWRVIP-86, Revision 1, EPRI, Palo Alto, CA: September 2008. 1016575. (EPRI Proprietary)

6.5 *Pressure-Temperature Limits Report for Exelon Generation Company LLC, Peach Bottom Atomic Power Station Unit 2*, 0000-0107-8606-R1, Revision 1, Class III (GEH Proprietary Information), March 2012

6.6 *Pressure-Temperature Limits Report for Exelon Generation Company LLC, Peach Bottom Atomic Power Station Unit 3*, 0000-0133-0506-R1, Revision 1, Class III (GEH Proprietary Information), March 2012

6.7 *Peach Bottom Atomic Power Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis*, SASR 88-24, Revision 1, DRF B13-01445-1, Class I, December 1991

6.8 *Peach Bottom Atomic Power Station Unit 3 Vessel Surveillance Materials Testing and Fracture Toughness Analysis*, SASR 90-50, Revision 1, DRF B11-00494, Class I, July 1995

Figure 1 – Unit 2 Composite Curve A Pressure Test P-T Curves Effective for up to 32 EFPY

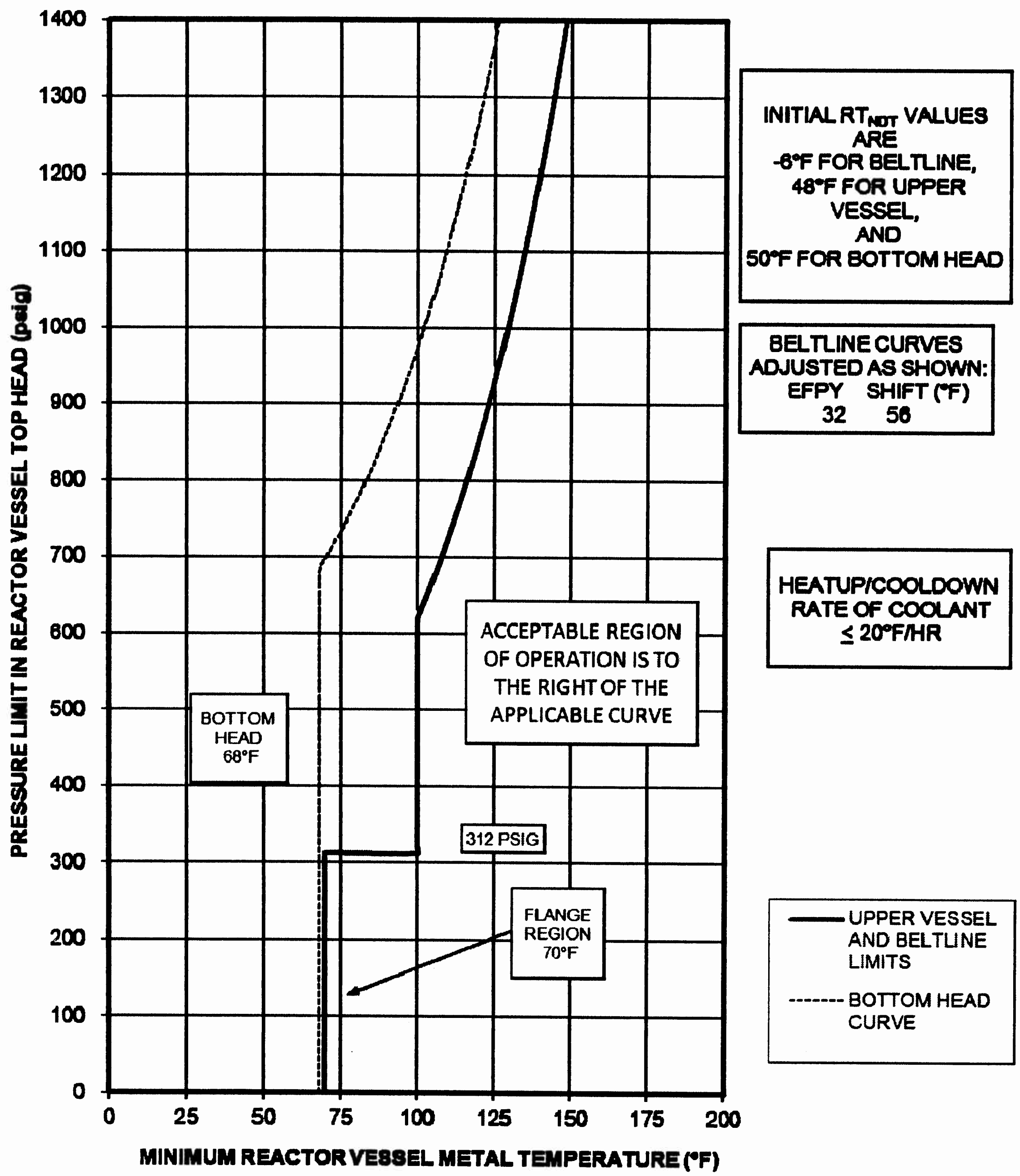

Figure 2 – Unit 2 Composite Curve B Core Not Critical P-T Curves Effective for up to 32 EFPY

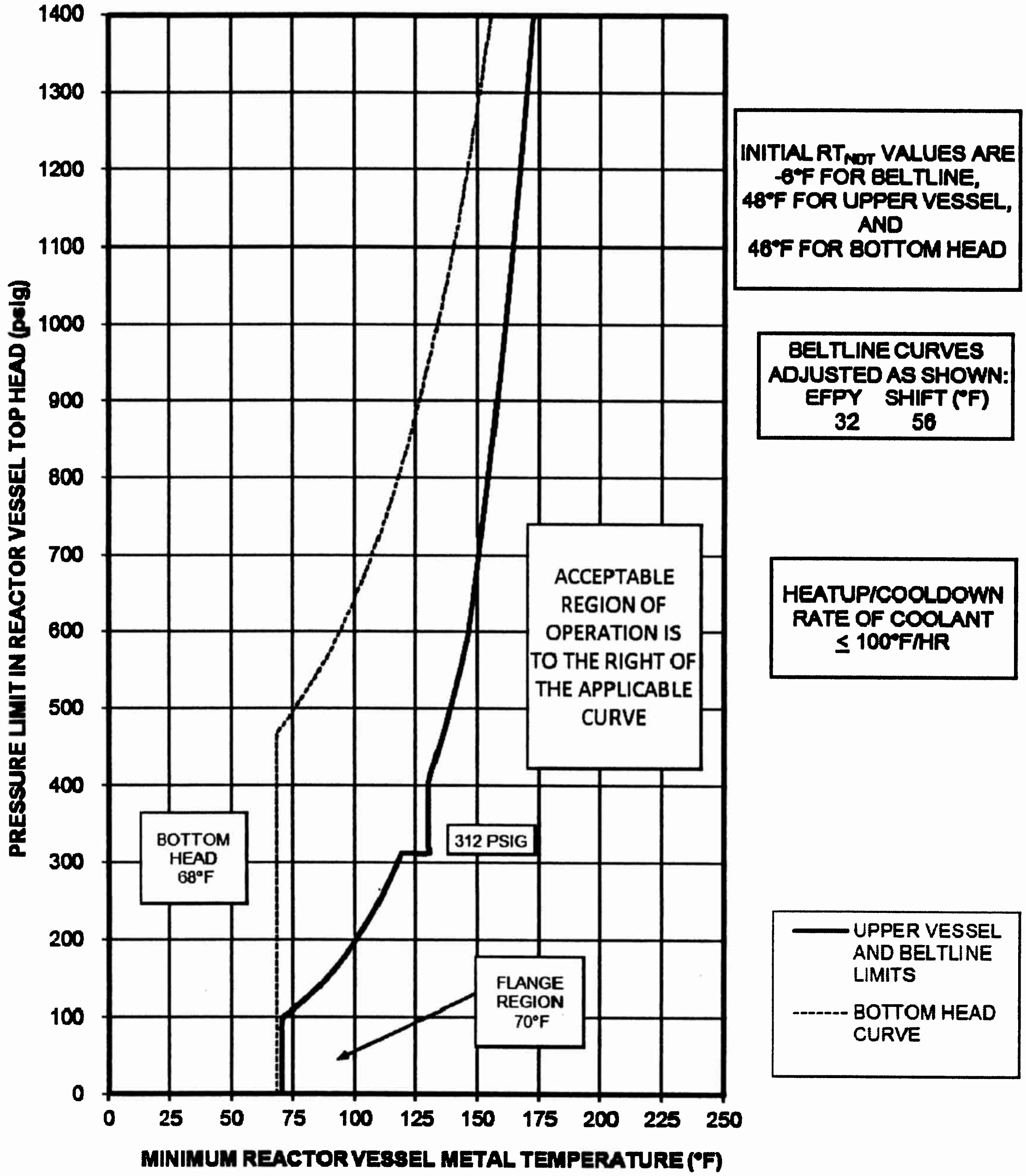

Figure 3 – Unit 2 Limiting Curve C Core Critical P-T Curves Effective for up to 32 EFPY

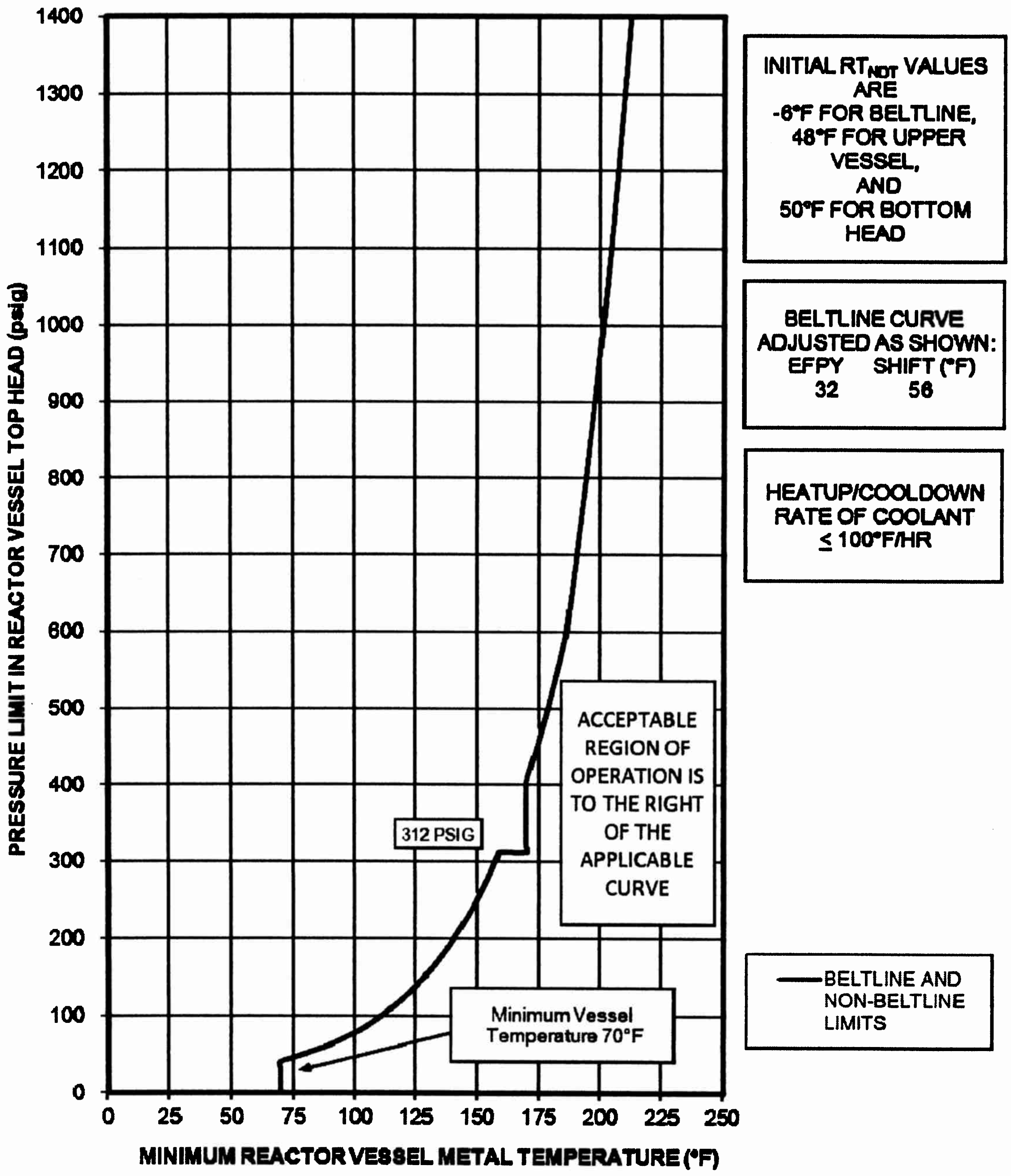

Figure 4 – Unit 2 Composite Curve A Pressure Test P-T Curves Effective for up to 54 EFPY

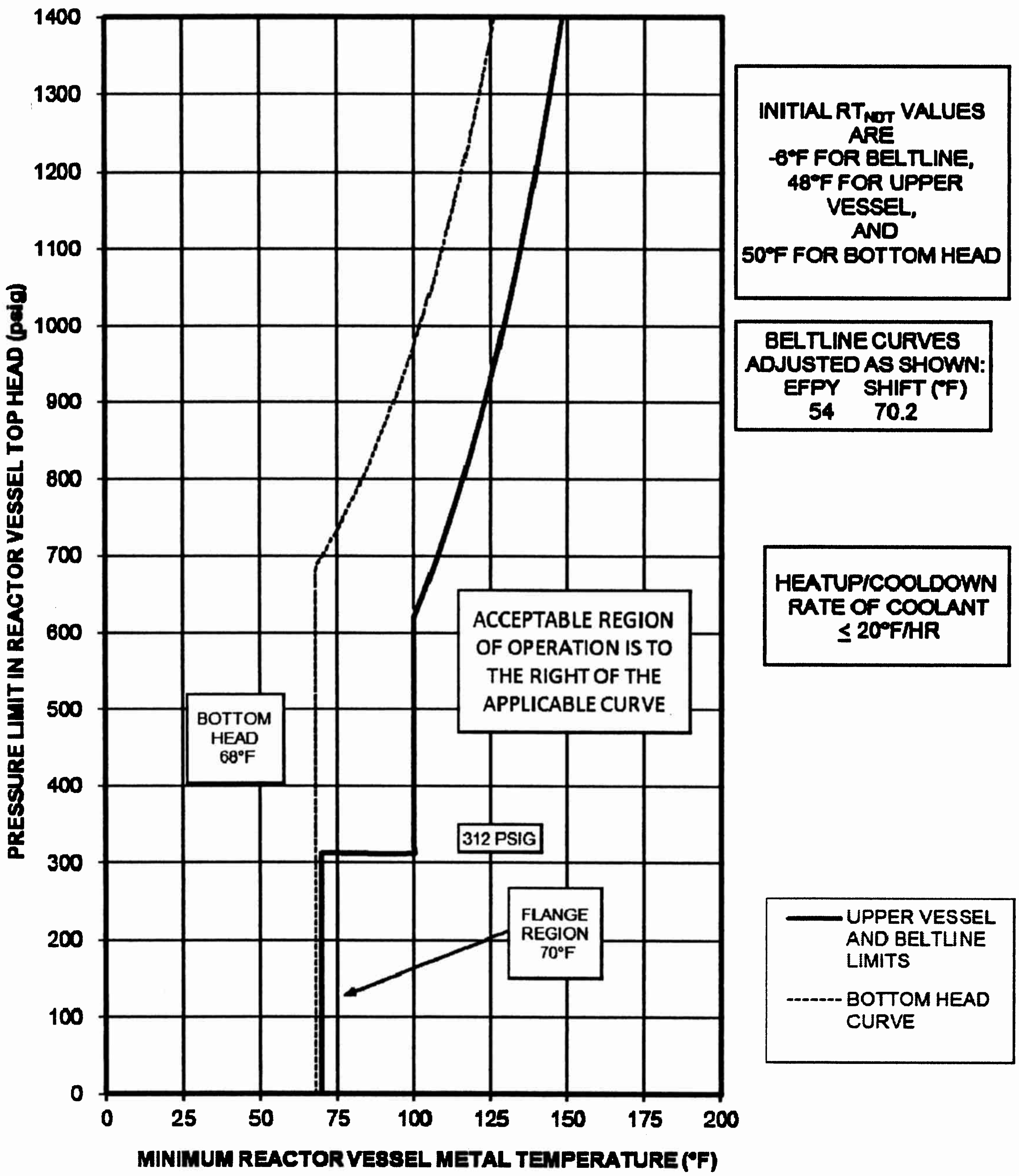

Figure 5 – Unit 2 Composite Curve B Core Not Critical P-T Curves Effective for up to 54 EFPY

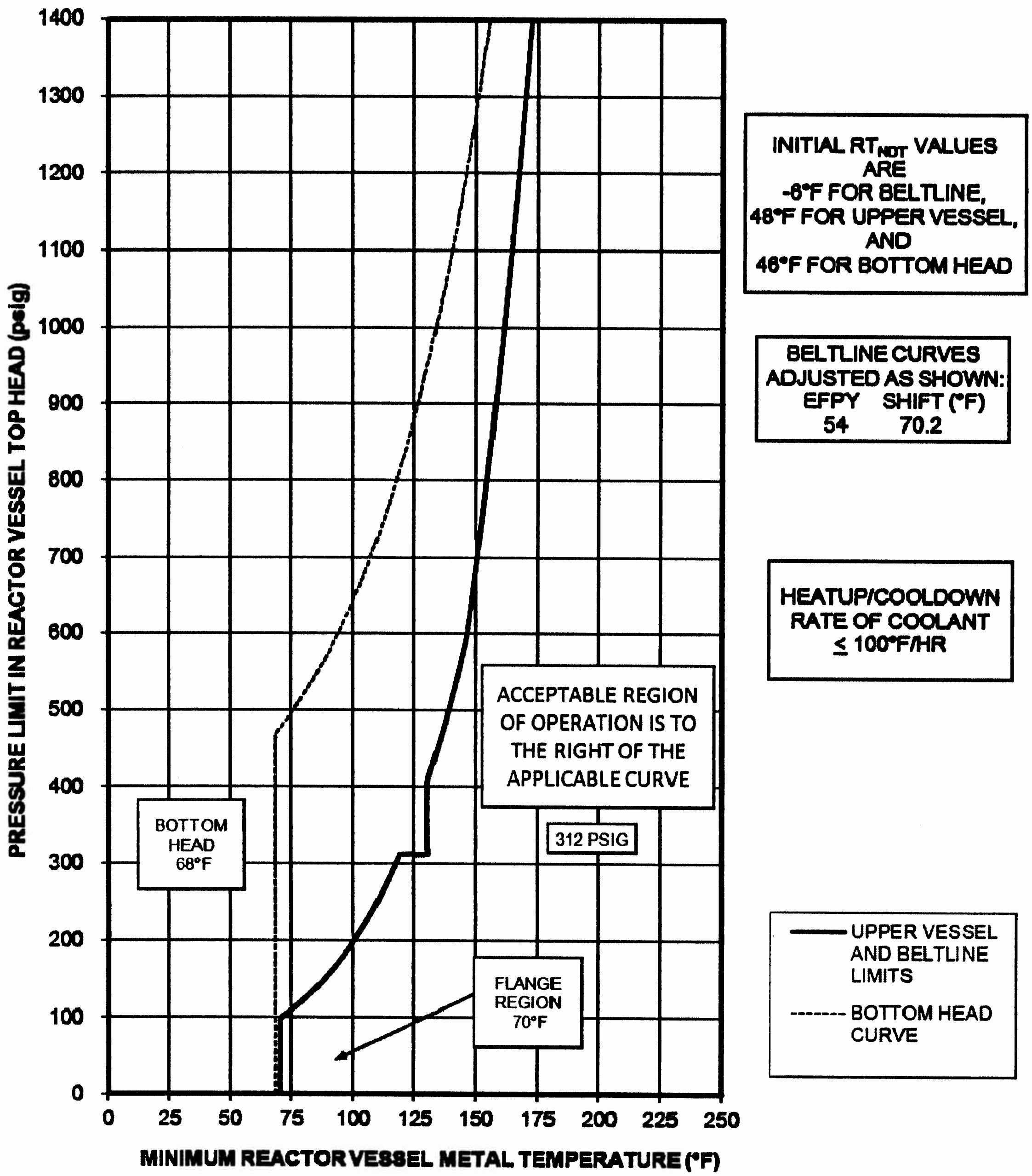

Figure 6 – Unit 2 Limiting Curve C Core Critical P-T Curves Effective for up to 54 EFPY

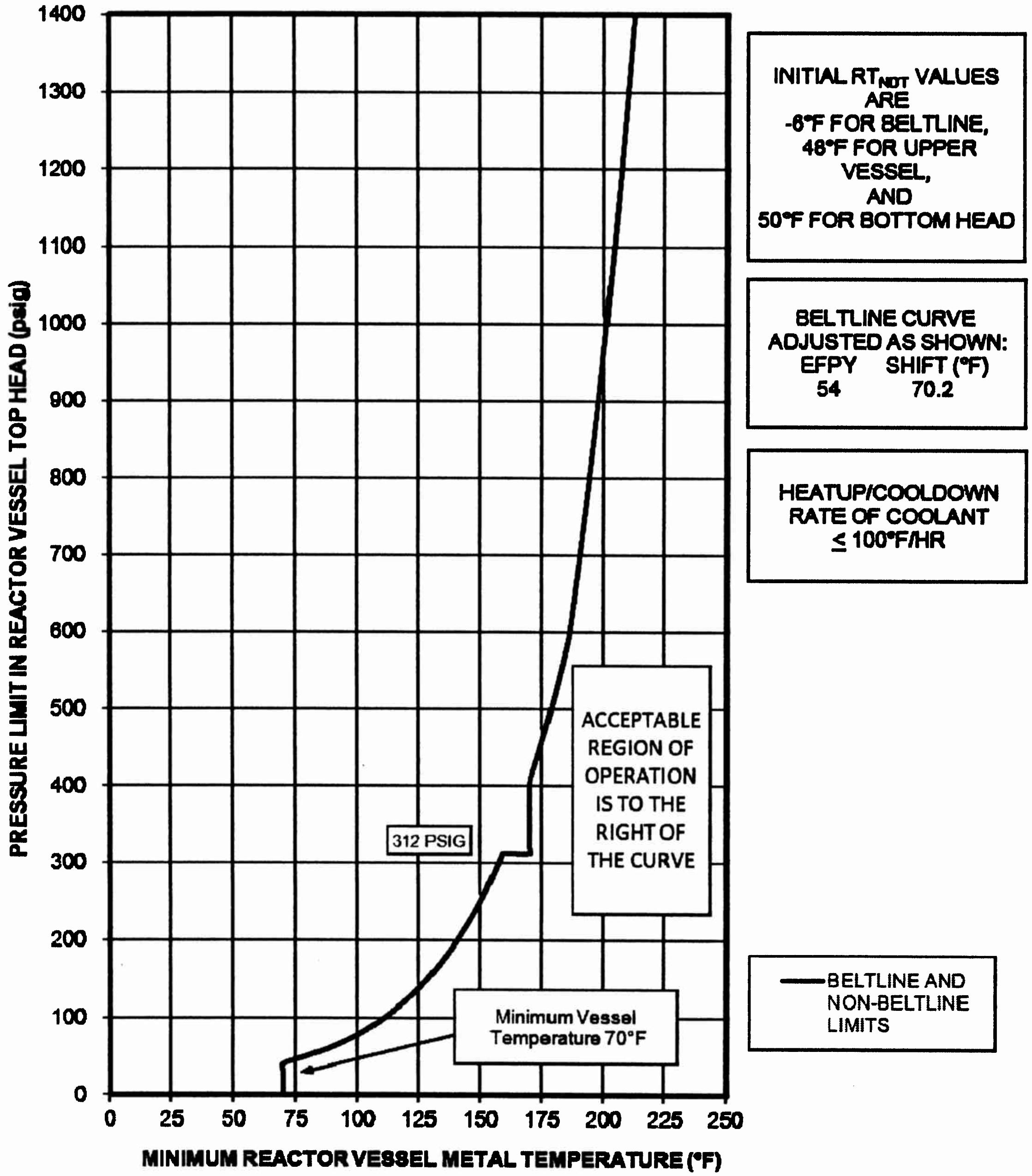

Figure 7 – Unit 3 Composite Curve A Pressure Test P-T Curves Effective for up to 32 EFPY

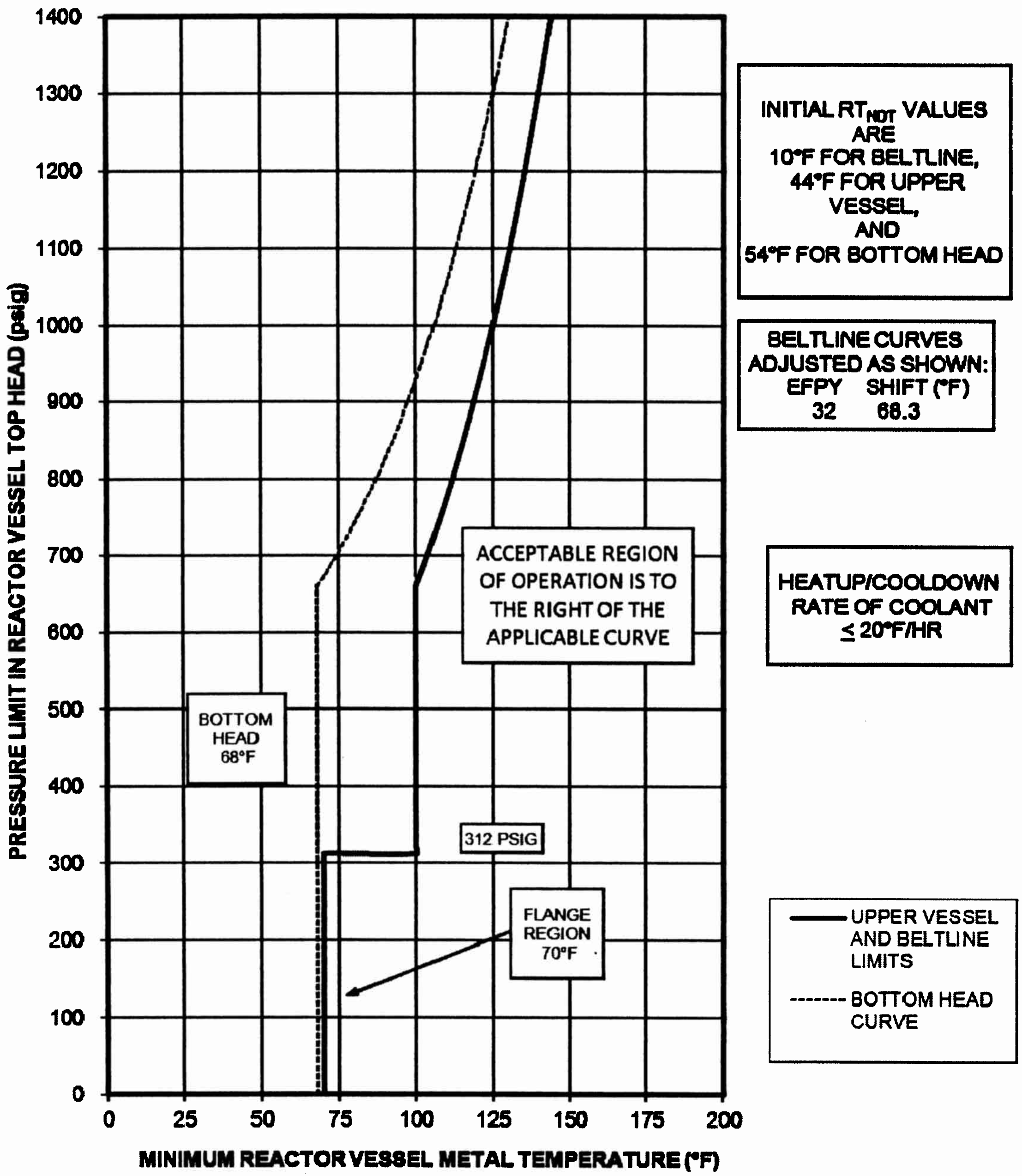

Figure 8 – Unit 3 Composite Curve B Core Not Critical P-T Curves Effective for up to 32 EFPY

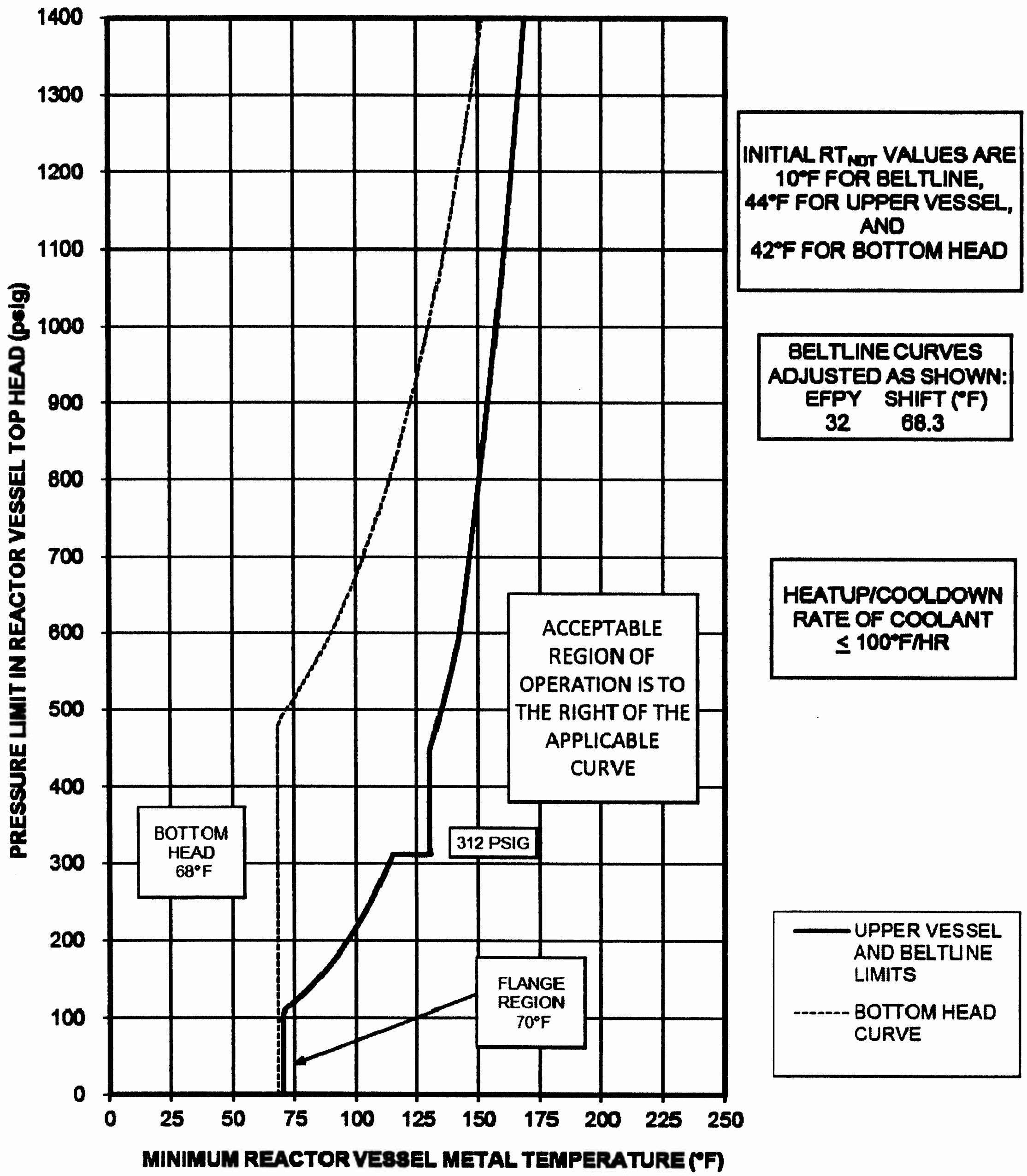

Figure 9 – Unit 3 Limiting Curve C Core Critical P-T Curves Effective for up to 32 EFPY

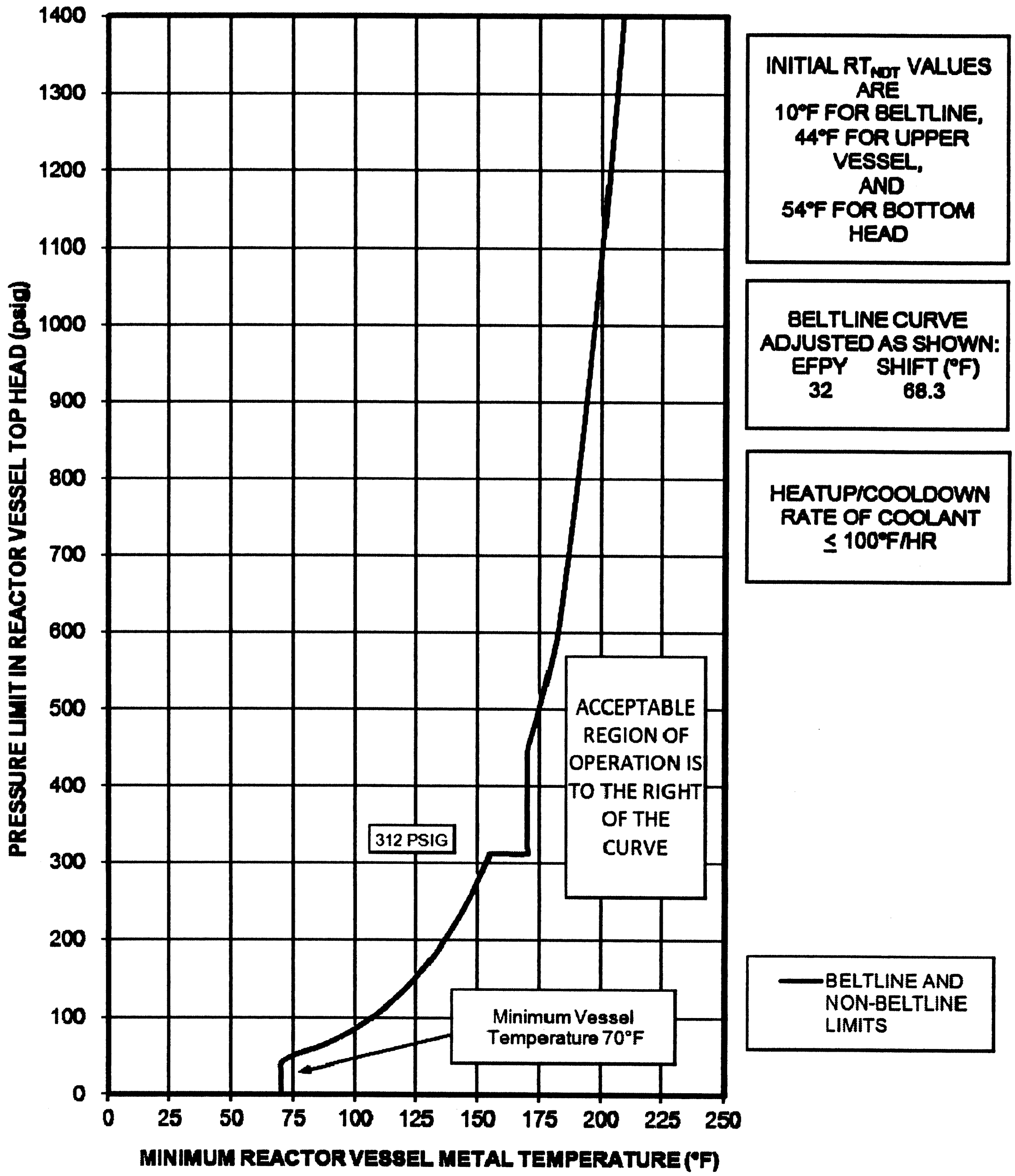

Figure 10 – Unit 3 Composite Curve A Pressure Test P-T Curves Effective for up to 54 EFPY

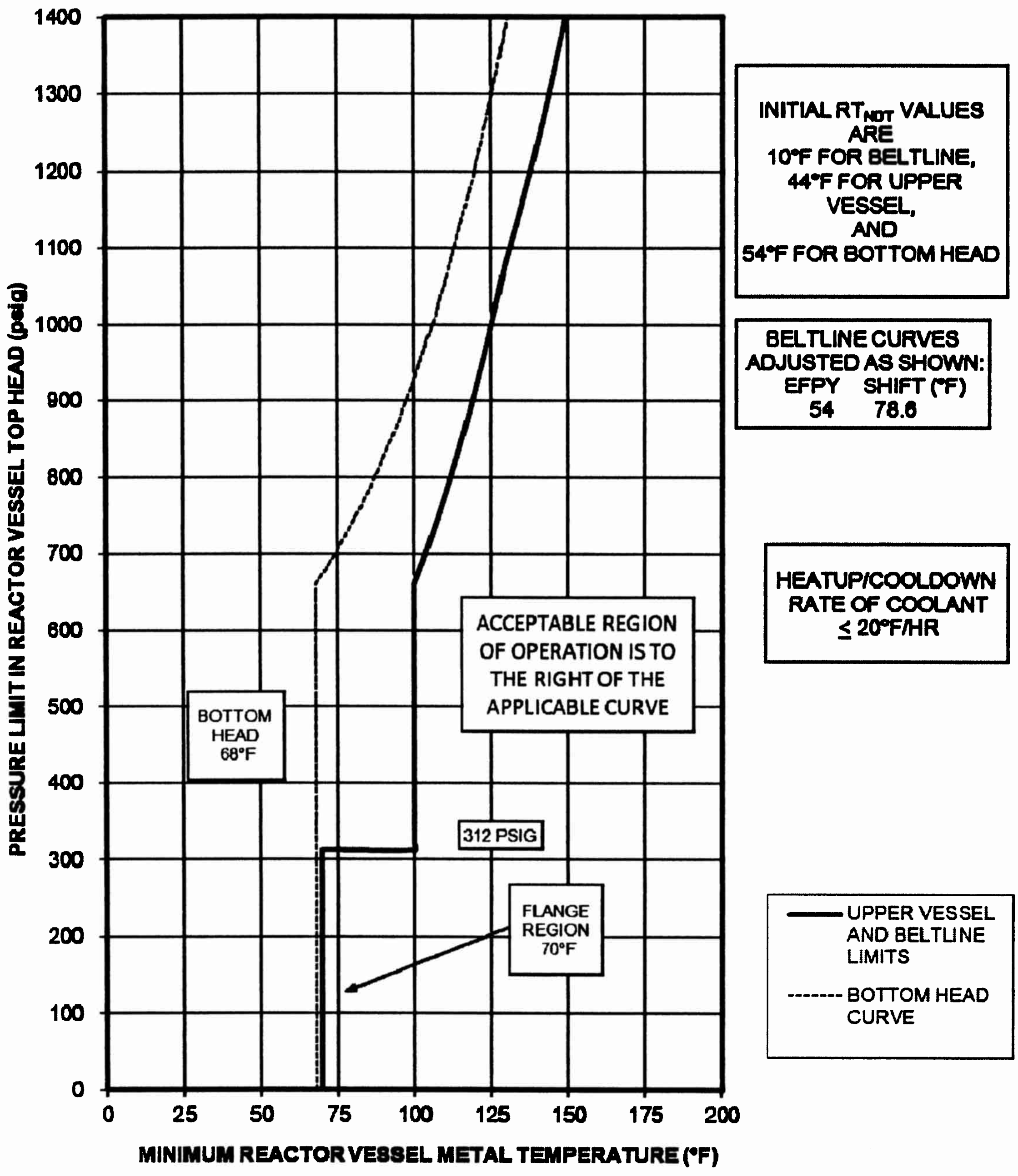

Figure 11 – Unit 3 Composite Curve B Core Not Critical P-T Curves Effective for up to 54 EFPY

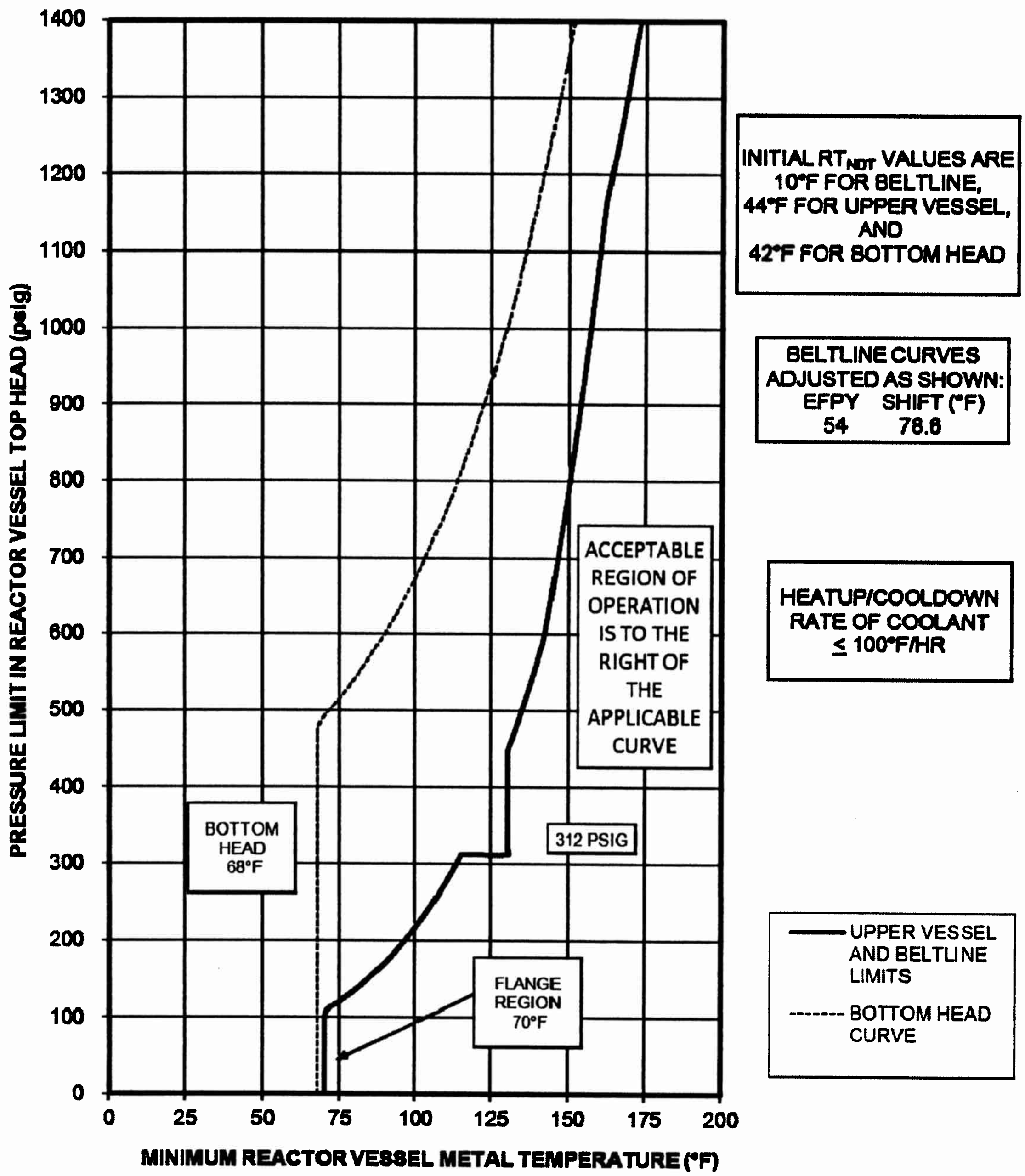

Figure 12 – Unit 3 Limiting Curve C Core Critical P-T Curves Effective for up to 54 EFPY

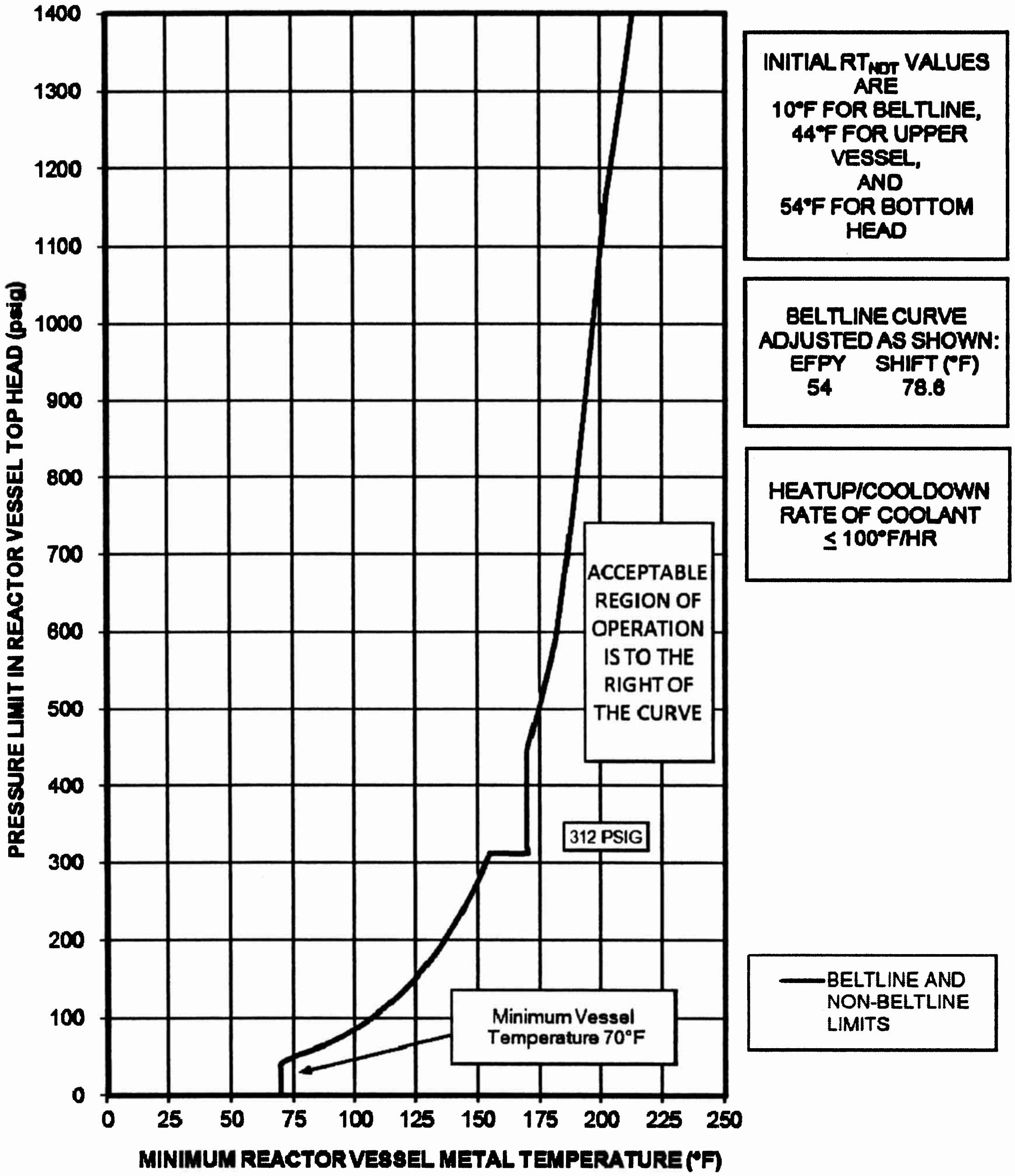

Table 1 – Unit 2 Tabulation of Curves – 32 EFPY

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	79.1
60	68.0	70.0	68.0	70.0	88.0
70	68.0	70.0	68.0	70.0	95.2
80	68.0	70.0	68.0	70.0	101.2
90	68.0	70.0	68.0	70.0	106.3
100	68.0	70.0	68.0	70.8	110.8
110	68.0	70.0	68.0	74.9	114.9
120	68.0	70.0	68.0	78.7	118.7
130	68.0	70.0	68.0	82.2	122.2
140	68.0	70.0	68.0	85.4	125.4
150	68.0	70.0	68.0	88.2	128.2
160	68.0	70.0	68.0	90.9	130.9
170	68.0	70.0	68.0	93.5	133.5
180	68.0	70.0	68.0	95.9	135.9
190	68.0	70.0	68.0	98.2	138.2
200	68.0	70.0	68.0	100.3	140.3
210	68.0	70.0	68.0	102.3	142.3
220	68.0	70.0	68.0	104.3	144.3
230	68.0	70.0	68.0	106.1	146.1
240	68.0	70.0	68.0	107.9	147.9
250	68.0	70.0	68.0	109.6	149.6
260	68.0	70.0	68.0	111.2	151.2
270	68.0	70.0	68.0	112.8	152.8
280	68.0	70.0	68.0	114.3	154.3
290	68.0	70.0	68.0	115.8	155.8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
300	68.0	70.0	68.0	117.2	157.2
310	68.0	70.0	68.0	118.5	158.5
312.5	68.0	70.0	68.0	118.9	158.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.2	170.2
420	68.0	100.0	68.0	131.2	171.2
430	68.0	100.0	68.0	132.2	172.2
440	68.0	100.0	68.0	133.2	173.2
450	68.0	100.0	68.0	134.1	174.1
460	68.0	100.0	68.0	135.1	175.1
470	68.0	100.0	68.6	136.0	176.0
480	68.0	100.0	71.1	136.9	176.9
490	68.0	100.0	73.4	137.7	177.7
500	68.0	100.0	75.6	138.6	178.6
510	68.0	100.0	77.8	139.4	179.4
520	68.0	100.0	79.8	140.2	180.2
530	68.0	100.0	81.8	141.0	181.0
540	68.0	100.0	83.7	141.8	181.8
550	68.0	100.0	85.5	142.6	182.6
560	68.0	100.0	87.3	143.4	183.4
570	68.0	100.0	89.0	144.1	184.1
580	68.0	100.0	90.6	144.9	184.9
590	68.0	100.0	92.2	145.6	185.6
600	68.0	100.0	93.8	146.1	186.1

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
610	68.0	100.0	95.3	146.6	186.6
620	68.0	100.0	96.7	147.0	187.0
630	68.0	101.0	98.1	147.4	187.4
640	68.0	102.0	99.5	147.8	187.8
650	68.0	103.0	100.8	148.2	188.2
660	68.0	104.0	102.1	148.7	188.7
670	68.0	104.9	103.4	149.1	189.1
680	68.0	105.9	104.7	149.5	189.5
690	68.7	106.8	105.9	149.9	189.9
700	70.2	107.7	107.0	150.3	190.3
710	71.7	108.6	108.2	150.7	190.7
720	73.1	109.4	109.3	151.1	191.1
730	74.5	110.3	110.4	151.5	191.5
740	75.8	111.1	111.5	151.9	191.9
750	77.1	112.0	112.6	152.2	192.2
760	78.4	112.8	113.6	152.6	192.6
770	79.6	113.6	114.6	153.0	193.0
780	80.8	114.3	115.6	153.4	193.4
790	82.0	115.1	116.6	153.8	193.8
800	83.2	115.9	117.5	154.1	194.1
810	84.3	116.6	118.5	154.5	194.5
820	85.4	117.4	119.4	154.9	194.9
830	86.5	118.1	120.3	155.2	195.2
840	87.5	118.8	121.2	155.6	195.6
850	88.6	119.5	122.0	155.9	195.9
860	89.6	120.2	122.9	156.3	196.3
870	90.6	120.9	123.7	156.6	196.6
880	91.5	121.6	124.6	157.0	197.0
890	92.5	122.3	125.4	157.3	197.3
900	93.4	122.9	126.2	157.7	197.7
910	94.4	123.6	127.0	158.0	198.0
920	95.3	124.2	127.7	158.4	198.4
930	96.1	124.9	128.5	158.7	198.7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
940	97.0	125.5	129.3	159.0	199.0
950	97.9	126.1	130.0	159.4	199.4
960	98.7	126.7	130.7	159.7	199.7
970	99.6	127.3	131.5	160.0	200.0
980	100.4	127.9	132.2	160.4	200.4
990	101.2	128.5	132.9	160.7	200.7
1000	102.0	129.1	133.6	161.0	201.0
1010	102.7	129.7	134.2	161.3	201.3
1015	103.1	130.0	134.6	161.5	201.5
1020	103.5	130.2	134.9	161.6	201.6
1030	104.3	130.8	135.6	162.0	202.0
1035	104.6	131.1	135.9	162.1	202.1
1040	105.0	131.4	136.2	162.3	202.3
1050	105.7	131.9	136.9	162.6	202.6
1055	106.1	132.2	137.2	162.7	202.7
1060	106.4	132.5	137.5	162.9	202.9
1070	107.2	133.0	138.1	163.2	203.2
1080	107.9	133.5	138.8	163.5	203.5
1090	108.6	134.1	139.4	163.8	203.8
1100	109.2	134.6	140.0	164.1	204.1
1105	109.6	134.8	140.3	164.3	204.3
1110	109.9	135.1	140.6	164.4	204.4
1120	110.6	135.6	141.2	164.7	204.7
1130	111.2	136.1	141.8	165.0	205.0
1140	111.9	136.6	142.3	165.3	205.3
1150	112.5	137.1	142.9	165.6	205.6
1160	113.1	137.6	143.5	165.9	205.9
1170	113.8	138.1	144.0	166.2	206.2
1180	114.4	138.6	144.6	166.5	206.5
1190	115.0	139.1	145.1	166.7	206.7
1200	115.6	139.5	145.7	167.0	207.0
1210	116.2	140.0	146.2	167.3	207.3
1220	116.8	140.5	146.8	167.6	207.6

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
1230	117.3	140.9	147.3	167.9	207.9
1240	117.9	141.4	147.8	168.2	208.2
1250	118.5	141.8	148.3	168.4	208.4
1260	119.0	142.3	148.8	168.7	208.7
1270	119.6	142.7	149.3	169.0	209.0
1280	120.1	143.2	149.8	169.2	209.2
1290	120.7	143.6	150.3	169.5	209.5
1300	121.2	144.0	150.8	169.8	209.8
1310	121.7	144.5	151.3	170.1	210.1
1320	122.3	144.9	151.8	170.3	210.3
1330	122.8	145.3	152.2	170.6	210.6
1340	123.3	145.7	152.7	170.8	210.8
1350	123.8	146.1	153.2	171.1	211.1
1360	124.3	146.6	153.6	171.4	211.4
1370	124.8	147.0	154.1	171.6	211.6
1380	125.3	147.4	154.5	171.9	211.9
1390	125.8	147.8	155.0	172.1	212.1
1400	126.3	148.2	155.4	172.4	212.4

Table 2 – Unit 2 Tabulation of Curves – 54 EFPY

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	79.1
60	68.0	70.0	68.0	70.0	88.0
70	68.0	70.0	68.0	70.0	95.2
80	68.0	70.0	68.0	70.0	101.2
90	68.0	70.0	68.0	70.0	106.3
100	68.0	70.0	68.0	70.8	110.8
110	68.0	70.0	68.0	74.9	114.9
120	68.0	70.0	68.0	78.7	118.7
130	68.0	70.0	68.0	82.2	122.2
140	68.0	70.0	68.0	85.4	125.4
150	68.0	70.0	68.0	88.2	128.2
160	68.0	70.0	68.0	90.9	130.9
170	68.0	70.0	68.0	93.5	133.5
180	68.0	70.0	68.0	95.9	135.9
190	68.0	70.0	68.0	98.2	138.2
200	68.0	70.0	68.0	100.3	140.3
210	68.0	70.0	68.0	102.3	142.3
220	68.0	70.0	68.0	104.3	144.3
230	68.0	70.0	68.0	106.1	146.1
240	68.0	70.0	68.0	107.9	147.9
250	68.0	70.0	68.0	109.6	149.6
260	68.0	70.0	68.0	111.2	151.2
270	68.0	70.0	68.0	112.8	152.8
280	68.0	70.0	68.0	114.3	154.3

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
290	68.0	70.0	68.0	115.8	155.8
300	68.0	70.0	68.0	117.2	157.2
310	68.0	70.0	68.0	118.5	158.5
312.5	68.0	70.0	68.0	118.9	158.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.2	170.2
420	68.0	100.0	68.0	131.2	171.2
430	68.0	100.0	68.0	132.2	172.2
440	68.0	100.0	68.0	133.2	173.2
450	68.0	100.0	68.0	134.1	174.1
460	68.0	100.0	68.0	135.1	175.1
470	68.0	100.0	68.6	136.0	176.0
480	68.0	100.0	71.1	136.9	176.9
490	68.0	100.0	73.4	137.7	177.7
500	68.0	100.0	75.6	138.6	178.6
510	68.0	100.0	77.8	139.4	179.4
520	68.0	100.0	79.8	140.2	180.2
530	68.0	100.0	81.8	141.0	181.0
540	68.0	100.0	83.7	141.8	181.8
550	68.0	100.0	85.5	142.6	182.6
560	68.0	100.0	87.3	143.4	183.4
570	68.0	100.0	89.0	144.1	184.1
580	68.0	100.0	90.6	144.9	184.9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
590	68.0	100.0	92.2	145.6	185.6
600	68.0	100.0	93.8	146.1	186.1
610	68.0	100.0	95.3	146.6	186.6
620	68.0	100.0	96.7	147.0	187.0
630	68.0	101.0	98.1	147.4	187.4
640	68.0	102.0	99.5	147.8	187.8
650	68.0	103.0	100.8	148.2	188.2
660	68.0	104.0	102.1	148.7	188.7
670	68.0	104.9	103.4	149.1	189.1
680	68.0	105.9	104.7	149.5	189.5
690	68.7	106.8	105.9	149.9	189.9
700	70.2	107.7	107.0	150.3	190.3
710	71.7	108.6	108.2	150.7	190.7
720	73.1	109.4	109.3	151.1	191.1
730	74.5	110.3	110.4	151.5	191.5
740	75.8	111.1	111.5	151.9	191.9
750	77.1	112.0	112.6	152.2	192.2
760	78.4	112.8	113.6	152.6	192.6
770	79.6	113.6	114.6	153.0	193.0
780	80.8	114.3	115.6	153.4	193.4
790	82.0	115.1	116.6	153.8	193.8
800	83.2	115.9	117.5	154.1	194.1
810	84.3	116.6	118.5	154.5	194.5
820	85.4	117.4	119.4	154.9	194.9
830	86.5	118.1	120.3	155.2	195.2
840	87.5	118.8	121.2	155.6	195.6
850	88.6	119.5	122.0	155.9	195.9
860	89.6	120.2	122.9	156.3	196.3
870	90.6	120.9	123.7	156.6	196.6
880	91.5	121.6	124.6	157.0	197.0
890	92.5	122.3	125.4	157.3	197.3
900	93.4	122.9	126.2	157.7	197.7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
910	94.4	123.6	127.0	158.0	198.0
920	95.3	124.2	127.7	158.4	198.4
930	96.1	124.9	128.5	158.7	198.7
940	97.0	125.5	129.3	159.0	199.0
950	97.9	126.1	130.0	159.4	199.4
960	98.7	126.7	130.7	159.7	199.7
970	99.6	127.3	131.5	160.0	200.0
980	100.4	127.9	132.2	160.4	200.4
990	101.2	128.5	132.9	160.7	200.7
1000	102.0	129.1	133.6	161.0	201.0
1010	102.7	129.7	134.2	161.3	201.3
1015	103.1	130.0	134.6	161.5	201.5
1020	103.5	130.2	134.9	161.6	201.6
1030	104.3	130.8	135.6	162.0	202.0
1035	104.6	131.1	135.9	162.1	202.1
1040	105.0	131.4	136.2	162.3	202.3
1050	105.7	131.9	136.9	162.6	202.6
1055	106.1	132.2	137.2	162.7	202.7
1060	106.4	132.5	137.5	162.9	202.9
1070	107.2	133.0	138.1	163.2	203.2
1080	107.9	133.5	138.8	163.5	203.5
1090	108.6	134.1	139.4	163.8	203.8
1100	109.2	134.6	140.0	164.1	204.1
1105	109.6	134.8	140.3	164.3	204.3
1110	109.9	135.1	140.6	164.4	204.4
1120	110.6	135.6	141.2	164.7	204.7
1130	111.2	136.1	141.8	165.0	205.0
1140	111.9	136.6	142.3	165.3	205.3
1150	112.5	137.1	142.9	165.6	205.6
1160	113.1	137.6	143.5	165.9	205.9
1170	113.8	138.1	144.0	166.2	206.2
1180	114.4	138.6	144.6	166.5	206.5

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
1190	115.0	139.1	145.1	166.7	206.7
1200	115.6	139.5	145.7	167.0	207.0
1210	116.2	140.0	146.2	167.3	207.3
1220	116.8	140.5	146.8	167.6	207.6
1230	117.3	140.9	147.3	167.9	207.9
1240	117.9	141.4	147.8	168.2	208.2
1250	118.5	141.8	148.3	168.4	208.4
1260	119.0	142.3	148.8	168.7	208.7
1270	119.6	142.7	149.3	169.0	209.0
1280	120.1	143.2	149.8	169.2	209.2
1290	120.7	143.6	150.3	169.5	209.5
1300	121.2	144.0	150.8	169.8	209.8
1310	121.7	144.5	151.3	170.1	210.1
1320	122.3	144.9	151.8	170.3	210.3
1330	122.8	145.3	152.2	170.6	210.6
1340	123.3	145.7	152.7	170.8	210.8
1350	123.8	146.1	153.2	171.1	211.1
1360	124.3	146.6	153.6	171.4	211.4
1370	124.8	147.0	154.1	171.6	211.6
1380	125.3	147.4	154.5	171.9	211.9
1390	125.8	147.8	155.0	172.1	212.1
1400	126.3	148.2	155.4	172.4	212.4

Table 3 – Unit 3 Tabulation of Curves – 32 EFPY

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	75.1
60	68.0	70.0	68.0	70.0	84.0
70	68.0	70.0	68.0	70.0	91.2
80	68.0	70.0	68.0	70.0	97.2
90	68.0	70.0	68.0	70.0	102.3
100	68.0	70.0	68.0	70.0	106.8
110	68.0	70.0	68.0	70.9	110.9
120	68.0	70.0	68.0	74.7	114.7
130	68.0	70.0	68.0	78.2	118.2
140	68.0	70.0	68.0	81.4	121.4
150	68.0	70.0	68.0	84.2	124.2
160	68.0	70.0	68.0	86.9	126.9
170	68.0	70.0	68.0	89.5	129.5
180	68.0	70.0	68.0	91.9	131.9
190	68.0	70.0	68.0	94.2	134.2
200	68.0	70.0	68.0	96.3	136.3
210	68.0	70.0	68.0	98.3	138.3
220	68.0	70.0	68.0	100.3	140.3
230	68.0	70.0	68.0	102.1	142.1
240	68.0	70.0	68.0	103.9	143.9
250	68.0	70.0	68.0	105.6	145.6
260	68.0	70.0	68.0	107.2	147.2
270	68.0	70.0	68.0	108.8	148.8
280	68.0	70.0	68.0	110.3	150.3
290	68.0	70.0	68.0	111.8	151.8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
300	68.0	70.0	68.0	113.2	153.2
310	68.0	70.0	68.0	114.5	154.5
312.5	68.0	70.0	68.0	114.9	154.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.0	170.0
420	68.0	100.0	68.0	130.0	170.0
430	68.0	100.0	68.0	130.0	170.0
440	68.0	100.0	68.0	130.0	170.0
450	68.0	100.0	68.0	130.1	170.1
460	68.0	100.0	68.0	131.1	171.1
470	68.0	100.0	68.0	132.0	172.0
480	68.0	100.0	68.0	132.9	172.9
490	68.0	100.0	69.4	133.7	173.7
500	68.0	100.0	71.6	134.6	174.6
510	68.0	100.0	73.8	135.4	175.4
520	68.0	100.0	75.8	136.2	176.2
530	68.0	100.0	77.8	137.0	177.0
540	68.0	100.0	79.7	137.8	177.8
550	68.0	100.0	81.5	138.6	178.6
560	68.0	100.0	83.3	139.4	179.4
570	68.0	100.0	85.0	140.1	180.1
580	68.0	100.0	86.6	140.9	180.9
590	68.0	100.0	88.2	141.6	181.6
600	68.0	100.0	89.8	142.1	182.1

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
610	68.0	100.0	91.3	142.6	182.6
620	68.0	100.0	92.7	143.0	183.0
630	68.0	100.0	94.1	143.4	183.4
640	68.0	100.0	95.5	143.8	183.8
650	68.0	100.0	96.8	144.2	184.2
660	68.0	100.0	98.1	144.7	184.7
670	69.6	100.9	99.4	145.1	185.1
680	71.2	101.9	100.7	145.5	185.5
690	72.7	102.8	101.9	145.9	185.9
700	74.2	103.7	103.0	146.3	186.3
710	75.7	104.6	104.2	146.7	186.7
720	77.1	105.4	105.3	147.1	187.1
730	78.5	106.3	106.4	147.5	187.5
740	79.8	107.1	107.5	147.9	187.9
750	81.1	108.0	108.6	148.2	188.2
760	82.4	108.8	109.6	148.6	188.6
770	83.6	109.6	110.6	149.0	189.0
780	84.8	110.3	111.6	149.4	189.4
790	86.0	111.1	112.6	149.8	189.8
800	87.2	111.9	113.5	150.1	190.1
810	88.3	112.6	114.5	150.5	190.5
820	89.4	113.4	115.4	150.9	190.9
830	90.5	114.1	116.3	151.2	191.2
840	91.5	114.8	117.2	151.6	191.6
850	92.6	115.5	118.0	151.9	191.9
860	93.6	116.2	118.9	152.3	192.3
870	94.6	116.9	119.7	152.6	192.6
880	95.5	117.6	120.6	153.0	193.0
890	96.5	118.3	121.4	153.3	193.3
900	97.4	118.9	122.2	153.7	193.7
910	98.4	119.6	123.0	154.0	194.0
920	99.3	120.2	123.7	154.4	194.4
930	100.1	120.9	124.5	154.7	194.7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
940	101.0	121.5	125.3	155.0	195.0
950	101.9	122.1	126.0	155.4	195.4
960	102.7	122.7	126.7	155.7	195.7
970	103.6	123.3	127.5	156.0	196.0
980	104.4	123.9	128.2	156.4	196.4
990	105.2	124.5	128.9	156.7	196.7
1000	106.0	125.1	129.6	157.0	197.0
1010	106.7	125.7	130.2	157.3	197.3
1015	107.1	126.0	130.6	157.5	197.5
1020	107.5	126.2	130.9	157.6	197.6
1030	108.3	126.8	131.6	158.0	198.0
1035	108.6	127.1	131.9	158.1	198.1
1040	109.0	127.4	132.2	158.3	198.3
1050	109.7	127.9	132.9	158.6	198.6
1055	110.1	128.2	133.2	158.7	198.7
1060	110.4	128.5	133.5	158.9	198.9
1070	111.2	129.0	134.1	159.2	199.2
1080	111.9	129.5	134.8	159.5	199.5
1090	112.6	130.1	135.4	159.8	199.8
1100	113.2	130.6	136.0	160.1	200.1
1105	113.6	130.8	136.3	160.3	200.3
1110	113.9	131.1	136.6	160.4	200.4
1120	114.6	131.6	137.2	160.7	200.7
1130	115.2	132.1	137.8	161.0	201.0
1140	115.9	132.6	138.3	161.3	201.3
1150	116.5	133.1	138.9	161.6	201.6
1160	117.1	133.6	139.5	161.9	201.9
1170	117.8	134.1	140.0	162.2	202.2
1180	118.4	134.6	140.6	162.5	202.5
1190	119.0	135.1	141.1	162.7	202.7
1200	119.6	135.5	141.7	163.0	203.0
1210	120.2	136.0	142.2	163.3	203.3
1220	120.8	136.5	142.8	163.6	203.6

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	LIMITING 32 EFPY CURVE C (°F)
1230	121.3	136.9	143.3	163.9	203.9
1240	121.9	137.4	143.8	164.2	204.2
1250	122.5	137.8	144.3	164.4	204.4
1260	123.0	138.3	144.8	164.7	204.7
1270	123.6	138.7	145.3	165.0	205.0
1280	124.1	139.2	145.8	165.2	205.2
1290	124.7	139.6	146.3	165.5	205.5
1300	125.2	140.0	146.8	165.8	205.8
1310	125.7	140.5	147.3	166.1	206.1
1320	126.3	140.9	147.8	166.3	206.3
1330	126.8	141.3	148.2	166.6	206.6
1340	127.3	141.7	148.7	166.8	206.8
1350	127.8	142.1	149.2	167.1	207.1
1360	128.3	142.6	149.6	167.4	207.4
1370	128.8	143.0	150.1	167.6	207.6
1380	129.3	143.4	150.5	167.9	207.9
1390	129.8	143.8	151.0	168.1	208.1
1400	130.3	144.2	151.4	168.4	208.4

Table 4 – Unit 3 Tabulation of Curves – 54 EFPY

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	75.1
60	68.0	70.0	68.0	70.0	84.0
70	68.0	70.0	68.0	70.0	91.2
80	68.0	70.0	68.0	70.0	97.2
90	68.0	70.0	68.0	70.0	102.3
100	68.0	70.0	68.0	70.0	106.8
110	68.0	70.0	68.0	70.9	110.9
120	68.0	70.0	68.0	74.7	114.7
130	68.0	70.0	68.0	78.2	118.2
140	68.0	70.0	68.0	81.4	121.4
150	68.0	70.0	68.0	84.2	124.2
160	68.0	70.0	68.0	86.9	126.9
170	68.0	70.0	68.0	89.5	129.5
180	68.0	70.0	68.0	91.9	131.9
190	68.0	70.0	68.0	94.2	134.2
200	68.0	70.0	68.0	96.3	136.3
210	68.0	70.0	68.0	98.3	138.3
220	68.0	70.0	68.0	100.3	140.3
230	68.0	70.0	68.0	102.1	142.1
240	68.0	70.0	68.0	103.9	143.9
250	68.0	70.0	68.0	105.6	145.6
260	68.0	70.0	68.0	107.2	147.2
270	68.0	70.0	68.0	108.8	148.8
280	68.0	70.0	68.0	110.3	150.3
290	68.0	70.0	68.0	111.8	151.8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
300	68.0	70.0	68.0	113.2	153.2
310	68.0	70.0	68.0	114.5	154.5
312.5	68.0	70.0	68.0	114.9	154.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.0	170.0
420	68.0	100.0	68.0	130.0	170.0
430	68.0	100.0	68.0	130.0	170.0
440	68.0	100.0	68.0	130.0	170.0
450	68.0	100.0	68.0	130.1	170.1
460	68.0	100.0	68.0	131.1	171.1
470	68.0	100.0	68.0	132.0	172.0
480	68.0	100.0	68.0	132.9	172.9
490	68.0	100.0	69.4	133.7	173.7
500	68.0	100.0	71.6	134.6	174.6
510	68.0	100.0	73.8	135.4	175.4
520	68.0	100.0	75.8	136.2	176.2
530	68.0	100.0	77.8	137.0	177.0
540	68.0	100.0	79.7	137.8	177.8
550	68.0	100.0	81.5	138.6	178.6
560	68.0	100.0	83.3	139.4	179.4
570	68.0	100.0	85.0	140.1	180.1
580	68.0	100.0	86.6	140.9	180.9
590	68.0	100.0	88.2	141.6	181.6
600	68.0	100.0	89.8	142.1	182.1

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
610	68.0	100.0	91.3	142.6	182.6
620	68.0	100.0	92.7	143.0	183.0
630	68.0	100.0	94.1	143.4	183.4
640	68.0	100.0	95.5	143.8	183.8
650	68.0	100.0	96.8	144.2	184.2
660	68.0	100.0	98.1	144.7	184.7
670	69.6	100.9	99.4	145.1	185.1
680	71.2	101.9	100.7	145.5	185.5
690	72.7	102.8	101.9	145.9	185.9
700	74.2	103.7	103.0	146.3	186.3
710	75.7	104.6	104.2	146.7	186.7
720	77.1	105.4	105.3	147.1	187.1
730	78.5	106.3	106.4	147.5	187.5
740	79.8	107.1	107.5	147.9	187.9
750	81.1	108.0	108.6	148.2	188.2
760	82.4	108.8	109.6	148.6	188.6
770	83.6	109.6	110.6	149.0	189.0
780	84.8	110.3	111.6	149.4	189.4
790	86.0	111.1	112.6	149.8	189.8
800	87.2	111.9	113.5	150.1	190.1
810	88.3	112.6	114.5	150.5	190.5
820	89.4	113.4	115.4	150.9	190.9
830	90.5	114.1	116.3	151.2	191.2
840	91.5	114.8	117.2	151.6	191.6
850	92.6	115.5	118.0	151.9	191.9
860	93.6	116.2	118.9	152.3	192.3
870	94.6	116.9	119.7	152.6	192.6
880	95.5	117.6	120.6	153.0	193.0
890	96.5	118.3	121.4	153.3	193.3
900	97.4	118.9	122.2	153.7	193.7
910	98.4	119.6	123.0	154.0	194.0
920	99.3	120.2	123.7	154.4	194.4
930	100.1	120.9	124.5	154.7	194.7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
940	101.0	121.5	125.3	155.0	195.0
950	101.9	122.1	126.0	155.4	195.4
960	102.7	122.7	126.7	155.7	195.7
970	103.6	123.3	127.5	156.0	196.0
980	104.4	123.9	128.2	156.4	196.4
990	105.2	124.5	128.9	156.7	196.7
1000	106.0	125.1	129.6	157.0	197.0
1010	106.7	125.7	130.2	157.3	197.3
1015	107.1	126.0	130.6	157.5	197.5
1020	107.5	126.2	130.9	157.6	197.6
1030	108.3	126.8	131.6	158.0	198.0
1035	108.6	127.1	131.9	158.1	198.1
1040	109.0	127.4	132.2	158.3	198.3
1050	109.7	127.9	132.9	158.6	198.6
1055	110.1	128.2	133.2	158.7	198.7
1060	110.4	128.5	133.5	158.9	198.9
1070	111.2	129.0	134.1	159.2	199.2
1080	111.9	129.5	134.8	159.5	199.5
1090	112.6	130.2	135.4	159.8	199.8
1100	113.2	130.9	136.0	160.1	200.1
1105	113.6	131.3	136.3	160.3	200.3
1110	113.9	131.7	136.6	160.4	200.4
1120	114.6	132.4	137.2	160.7	200.7
1130	115.2	133.1	137.8	161.0	201.0
1140	115.9	133.8	138.3	161.3	201.3
1150	116.5	134.5	138.9	161.6	201.6
1160	117.1	135.1	139.5	161.9	201.9
1170	117.8	135.8	140.0	162.3	202.3
1180	118.4	136.5	140.6	162.8	202.8
1190	119.0	137.1	141.1	163.3	203.3
1200	119.6	137.8	141.7	163.8	203.8
1210	120.2	138.4	142.2	164.3	204.3
1220	120.8	139.0	142.8	164.8	204.8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 54 EFPY CURVE B (°F)	LIMITING 54 EFPY CURVE C (°F)
1230	121.3	139.6	143.3	165.3	205.3
1240	121.9	140.2	143.8	165.8	205.8
1250	122.5	140.8	144.3	166.3	206.3
1260	123.0	141.4	144.8	166.7	206.7
1270	123.6	142.0	145.3	167.2	207.2
1280	124.1	142.6	145.8	167.7	207.7
1290	124.7	143.2	146.3	168.1	208.1
1300	125.2	143.7	146.8	168.6	208.6
1310	125.7	144.3	147.3	169.1	209.1
1320	126.3	144.9	147.8	169.5	209.5
1330	126.8	145.4	148.2	170.0	210.0
1340	127.3	146.0	148.7	170.4	210.4
1350	127.8	146.5	149.2	170.8	210.8
1360	128.3	147.0	149.6	171.3	211.3
1370	128.8	147.5	150.1	171.7	211.7
1380	129.3	148.1	150.5	172.1	212.1
1390	129.8	148.6	151.0	172.6	212.6
1400	130.3	149.1	151.4	173.0	213.0

Appendix A: Reactor Vessel Material Surveillance Program

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, the first surveillance capsule was removed from the PBAPS Unit 2 reactor vessel after Cycle 7, during refueling outage (RFO) 7. The surveillance capsule contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The flux wires and test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of testing are presented in Reference 6.7, as required by 10 CFR 50, Appendices G and H. After testing, a reconstituted capsule was prepared and installed in the vessel during RFO 8.

The first surveillance capsule was removed from the PBAPS Unit 3 reactor vessel after Cycle 7, during refueling outage (RFO) 7. The surveillance capsule contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The flux wires and test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of testing are presented in Reference 6.8, as required by 10 CFR 50, Appendices G and H. After testing, a reconstituted capsule was prepared and installed in the vessel during RFO 8.

As described in PBAPS Unit 2 and PBAPS Unit 3 Updated Final Safety Analysis Report (UFSAR) Section 4.2.6, Inspection and Testing, the Integrated Surveillance Program (ISP) will determine the removal schedule for the remaining PBAPS Unit 2 and Unit 3 surveillance capsules. The PBAPS material surveillance program is administered in accordance with the BWR Vessel and Internals Project (VIP) ISP. The ISP combines the US BWR surveillance programs into a single integrated program. This program uses similar heats of materials in the surveillance programs of BWRs to represent the limiting materials in other vessels. It also adds data from the BWR Supplemental Surveillance Program (SSP). Per the BWRVIP ISP, Unit 2 is a host plant; the second Unit 2 capsule is scheduled to be removed at 33.7 EFPY and the remaining capsules are classified as "Standby". Per the BWRVIP ISP, Unit 3 is not a host plant, and the remaining capsules are classified as "Standby".

Appendix B: PBAPS Unit 2 Reactor Pressure Vessel P-T Curve Supporting Plant-Specific Information

Figure B-1: PBAPS Unit 2 Reactor Pressure Vessels

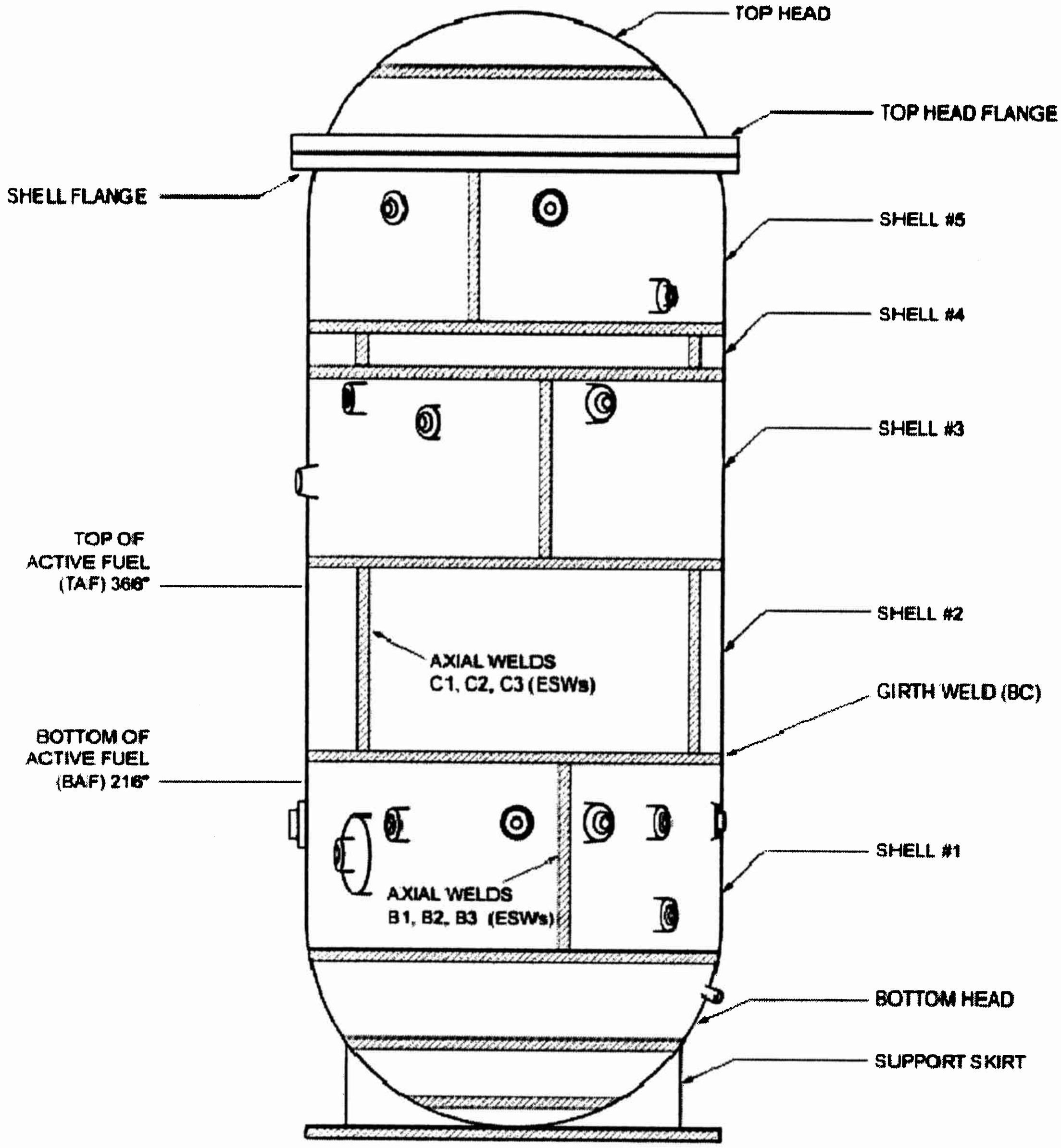

Table B-1: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Top Head and Cylindrical Shell Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
PLATES									
Top Head & Flange									
Shell Flange									
Mk. 48-139-1	124-V-201 ASB-87	10	L	141	123	111	40	10	10
Head Flange									
Mk. 209-139-1	5P-2744 ABU 134	10	L	57	78	96	40	10	10
Top Head Dollar									
Mk. 201-127-1	B5842-2	40	L	66	67	72	70	40	40
Top Head Side Plates									
Mk. 202-145-1	C3131-3	10	L	75	46	91	48	10	10
Mk. 202-145-5	C3042-3	10	L	80	84	79	40	10	10
Mk. 202-146-3	C3262-1	10	L	97	94	79	40	10	10
Mk. 202-145-6	C3042-3	10	L	87	87	84	40	10	10
Mk. 202-146-5	C3262-3	10	L	105	103	92	40	10	10
Mk. 202-146-6	C3262-3	10	L	72	78	93	40	10	10
Shell Courses									
Shell 5									
Mk. 6-139-4	C2796-2	10	L	43	53	49	54	10	10
Mk. 6-139-6	C2773-1	10	L	45	50	59	50	10	10
Mk. 6-139-5	C2863-2	10	L	65	62	72	40	10	10
Shell 4									
Mk. 15-139-1	C2789-3	10	L	68	41	65	58	10	10
Mk. 15-139-2	C2775-3	10	L	66	71	51	40	10	10
Mk. 15-139-3	B6776-1	10	L	77	69	65	40	10	10
Shell 3									
Mk. 6-139-1	C3042-2	10	L	50	56	59	40	10	10
Mk. 6-139-2	C2796-1	10	L	65	54	47	46	10	10
Mk. 6-139-3	C2859-2	10	L	62	59	62	40	10	10
Shell 2									
Mk. 6-139-17	C2761-2	10	L	57	69	58	40	-20	-20
Mk. 6-139-18	C2873-1	10	L	43	52	48	54	-20	-6
Mk. 6-139-23	C2894-2	10	L	57	52	54	40	-30	-20
Shell 1									
Mk. 6-139-13	C2761-1	10	L	47	59	63	46	-30	-14
Mk. 6-139-14	C2791-2	10	L	61	55	44	52	-30	-8
Mk. 6-139-15	C2873-2	10	L	60	69	64	40	-30	-20

Table B-2: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Bottom Head and Support Skirt Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Bottom Head									
Upper Torus									
Mk. 2-122-12	A0931-2	40	L	79	86	77	70	40	40
Mk. 2-122-9	A0942-2	40	L	90	72	60	70	40	40
Mk. 2-127-5	B5825-2	40	L	57	50	68	70	40	40
Mk. 2-127-6	B5825-2	40	L	57	55	54	70	40	40
Mk. 2-127-4	A0985-2	40	L	74	75	50	70	40	40
Mk. 2-127-3	A0985-2	40	L	62	54	61	70	40	40
Lower Torus									
Mk. 4-139-5	A1907-2	40	L	65	33	40	104	40	44
Mk. 4-139-6	A1907-2	40	L	45	44	60	82	40	40
Mk. 4-139-7	C3042-1	40	L	45	65	40	90	40	40
Mk. 4-139-8	C3042-1	40	L	66	60	71	70	40	40
Bottom Head Dollar									
Mk. 1-139-2	C2781-2	40	L	32	32	55	106	40	46
Support Skirt									
Knuckle									
Mk. 24-127-1 through -4	B7478-3	40	L	56	61	69	70	40	40
Skirt									
Mk. 40-139-1	C3885-3C	40	L	100	103	104	70	40	40
Mk. 40-139-2	C3885-4C	40	L	94	66	81	70	40	40
Base Ring									
Mk. 41-139-5 thru 8	C3888-3	40	L	113	115	92	70	40	40

Table B-3: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Nozzle N1 through N3 Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Nozzles:									
N1 Recirculation Outlet Nozzle									
N1A	AV-1595, Forging 7I-6387	40	L	58	61	84	70	40	40
N1B	AV-1834, Forging 7K-6310	40	L	32	48	76	106	40	46
N2 Recirculation Inlet Nozzle									
N2A	EV9947, Forging 7G-6200	40	L	76	64	60	70	-10	10
N2B	EV9934, Forging 7G-6197	40	L	34	44	36	102	0	42
N2C	EV9947, Forging 7G-6205	40	L	72	75	112	70	-10	10
N2D	EV9947, Forging 7G-6203	40	L	56	95	84	70	-10	10
N2E	AV1662, Forging 7I-6158A	40	L	92	96	89	70	40	40
N2F	EV9934, Forging 7G-6194	40	L	30	40	31	110	0	50
N2G	AV1662, Forging 7I-6158B	40	L	92	96	89	70	40	40
N2H	EV9947, Forging 7G-6202	40	L	56	95	84	70	-10	10
N2J	EV9947, Forging 7G-6201	40	L	76	64	60	70	0	10
N2K	EV9934, Forging 7G-6196	40	L	34	44	36	102	-10	42
N3 Steam Outlet Nozzle									
N3A	AV1670, Forging 7J-6163	40	L	55	46	45	80	40	40
N3B	AV1684, Forging 7J-6270	40	L	55	69	79	70	40	40
N3C	AV1844, Forging 7K-6012	40	L	74	34	54	102	40	42
N3D	AV1671, Forging 7J-6164	40	L	80	74	54	70	40	40

Table B-4: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Nozzle N4 through N16 Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
N4 Feedwater Nozzle									
N4A	EV9964, Forging 7H-6028B	40	L	31	53	55	108	10	48
N4B	AV1736, Forging 7I-6415A	40	L	60	60	60	70	40	40
N4C	AV1736, Forging 7I-6415B	40	L	60	60	60	70	40	40
N4D	EV9964, Forging 7H-6027B	40	L	31	34	33	108	10	48
N4E	AV1671, Forging 7I-6301B	40	L	83	63	65	70	40	40
N4F	AV1671, Forging 7I-6301A	40	L	83	63	65	70	40	40
N5 Core Spray Nozzle									
N5A	AV1607, Forging N7H6232A	40	L	95	104	80	70	0	10
N5B	AV1607, Forging N7H6232B	40	L	95	104	80	70	0	10
N6 Top Head Spray Nozzle									
N6A	ZT3043-4	40	L	170	158	142	70	40	40
N6B	BT2615-4	40	L	123	143	144	70	40	40
N7 Top Head Vent Nozzle	ZT3043- 3	40	L	113	122	146	70	40	40
N8 Jet Pump Instrumentation Nozzle									
N8A	BT2615-2, Forging 9584B	40	L	132	118	120	70	40	40
N8B	BT2615-4, Forging 9584D	40	L	132	118	120	70	40	40
N9 CRD HYD System Return Nozzle, Ser # 13-127	E23VW, Forging 438H-2	40	L	120	112	114	70	40	40
N10 Core Delta P& Liq Cont. Noz, Ser # 17-127-2	ZT3043-1	40	L	106	136	111	70	40	40
N11 Instrumentation Nozzle									
N11A & B (Alloy 600)	071708-1 & 8601-1								
N12 Instrumentation Nozzle									
N12A & B (Alloy 600)	071708-1 & 8601-1								
N13 & N14 High & Low Pressure Seal Leak Nozzle	A276N (Alloy 600)								
N15 Drain Nozzle	213099-1	40	L	39	42	44	92	-	32
N16 Instrumentation Nozzle									
N16A & B (Alloy 600)	071708-1 & 8601-1								

Table B-5: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Shell Weld Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Vertical and Meridional Welds									
Bottom Head Lower Torus (Z1, Z2, Z3, Z4)	N/A								
Bottom Head Upper Torus (A1, A2, A3, A4, A5, A6)	N/A								
Shell Course 1 (B1, B2, B3)	37C065 (Electroslag)								-45
Shell Course 2 (C1, C2, C3)	37C065 (Electroslag)								-45
Shell Course 3 (D1, D2, D3)	Electroslag								
Shell Course 4 (D4, D5, D6)	Electroslag								
Shell Course 5 (E1, E2, E3)	Electroslag								
Top Head Torus (G1, G2, G3, G4, G5, G6)	01R496/S925A27A	10	L	62	64	68	10	-	-50
Girth Welds									
Bottom Head Dollar to Lower Torus	432Z0471/B003A27A	10	L	100	102	106	10	-	-50
Bottom Head Dollar to Lower Torus	01R496/S925A27A	10	L	62	64	68	10	-	-50
Bottom Head Dollar to Lower Torus	CTY538/A027A27A	10	L	94	95	95	10	-	-50
Bottom Head Lower to Upper Torus	N/A								
Bottom Head Torus to Shell # 1	N/A								
Shell # 1 to Shell # 2 / Shell # 2 to Shell # 3	S-3986 Linde 124	10	L	41	45	46	28	-	-32
Shell # 2 to Shell # 3	88E081/F920A27A	10	L	85	90	91	10	-	-50
Shell # 3 to Shell # 4	N/A								
Shell # 4 to Shell # 5	CTY538/A027A27A	10	L	94	95	95	10	-	-50
Shell # 4 to Shell # 5	CTY538/B012A27A	10	L	77	81	87	10	-	-50
Shell # 4 to Shell # 5	08R4818/S922A27A	10	L	65	68	69	10	-	-50
Shell # 4 to Shell # 5	S-3986 Linde 124	10	L	41	45	46	28	-	-32
Shell #5 to Shell Flange	N/A								
Top Head Flange to Torus / Torus to Dollar	08R481/J908A27A	10	L	81	81	82	10	-	-50
Top Head Flange to Torus / Torus to Dollar	01R496/S925A27A	10	L	62	64	68	10	-	-50
Top Head Flange to Torus / Torus to Dollar	82D913/D908A27A	10	L	80	94	83	10	-	-50
Top Head Flange to Torus / Torus to Dollar	S-3986 Linde 124	10	L	41	45	46	28	-	-32
Top Head Torus to Dollar	88E081/F920A27A	10	L	85	90	91	10	-	-50

Table B-6: PBAPS Unit 2 Initial RT_{NDT} Values for RPV Materials
Nozzle and Appurtenance Weld and Bolting Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Nozzle Welds									
N1 Recirculation Outlet Nozzle	N/A								
N2 Recirculation Inlet Nozzle	N/A								
N3 Steam Outlet Nozzle	01R496/S925A27A	10	L	62	64	68	10	-	-50
	82D913/D908A27A	10	L	80	94	83	10	-	-50
	601221/E916A27A	10	L	107	108	108	10	-	-50
	601090/D901A27A	10	L	62	48	51	14	-	-46
	649T273/I726A27A	10	L	81	67	81.5	10	-	-50
	650X006/J807A27A	10	L	89	93	97	10	-	-50
	601382/S923A27A	10	L	77	78	78	10	-	-50
N4 Feedwater Nozzle	N/A								
N5 Core Spray Nozzle	Inconel								
N6, N7 Top Head Instrument & Vent Nozzle	432Z0471/B003A27A	10	L	100	102	106	10	-	-50
	05R938/A027A27A	10	L	66	84	86	10	-	-50
	CTY538/A027A27A	10	L	94	95	95	10	-	-50
N8 Jet Pump Instrumentation Nozzle	Stainless Steel								
N9 CRD HSR Nozzle	Inconel								
N10 Core ΔP and Liquid Control Nozzle	N/A								
N11, N12 Instrumentation Nozzle	Inconel								
N13, N14 High & Low Pressure Seal Leak Detector	N/A								
N15 Drain Nozzle	N/A								
N16 Instrumentation Nozzle	Inconel								
CRD Stub Tubes	Inconel								
Appurtenance Welds									
Shroud Support	Inconel								
Support Skirt Knuckle to Bottom Head	N/A								
Feedwater Sparger Bracket Pads / Guide Rod Brackets	Stainless Steel								
Surveillance Brackets / Core Spray Brackets	Stainless Steel								
Steam Dryer Support Brackets	Stainless Steel								
Dryer Hold Down Brackets	N/A								
Thermocouple Pads / Top Head Lifting Lugs	88E081/F920A27A	10	L	85	90	91	10	-	-50
Thermocouple Pads / Top Head Lifting Lugs	01R496/S925A27A	10	L	62	64	68	10	-	-50
Thermocouple Pads	432Z4521/B020A27A	10	L	83	94	96	10	-	-50
Thermocouple Pads	06R885/D001A27A	10	L	56	58	67	10	-	-50
Top Head Lifting Lugs	432Z0471/B003A27A	10	L	100	102	106	10	-	-50
Jet Pump Riser Pads	Stainless Steel								
Refueling Bellows Skirt	915L65	10	L	130	80	129	10	-	-50
Stabilizer Brackets	401Z8811/K914A27A	10	L	111	119	120	10	-	-50
Stabilizer Brackets	08R4818/S922A27A	10	L	65	68	69	10	-	-50
Stabilizer Brackets	402A0462/B023A27A	10	L	75	77	86	10	-	-50
Insulation Brackets	04R976/C004H1A	10	L	76	79	81	10	-	-50
Insulation Brackets	04R976/C006H1A	10	L	41	122	123	28	-	-32
STUDS:									LST
MK-61	6720443	10	n/a	35	36	37	n/a		70
	6780382	10	n/a	41	41	41	n/a		70

Table B-7: PBAPS Unit 2 Adjusted Reference Temperatures for up to 32 EFPY

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 1.61E+18 n/cm²
32 EFPY Peak 1/4 T fluence = 6.61E+17 n/cm²

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 1.23E+18 n/cm²
32 EFPY Peak 1/4 T fluence = 5.05E+17 n/cm²

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 5.69E+17 n/cm²
32 EFPY Peak 1/4 T fluence = 2.33E+17 n/cm²

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm²	32 EFPY Δ RT_{NDT} °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES													
PLATES:													
Lower Shell Mark 57	C2791-2	0.12	0.52	81.4		-8	5.05E+17	23.9	0	12.0	23.9	47.9	39.9
	C2761-1	0.11	0.54	73.4		-14	5.05E+17	21.6	0	10.8	21.6	43.2	29.2
	C2873-2	0.12	0.57	82.4		-20	5.05E+17	24.2	0	12.1	24.2	48.5	28.5
Lower-Intermediate Shell Mark 58	C2894-2	0.13	0.42	85.6		-20	6.61E+17	29.0	0	14.5	29.0	58.1	38.1
	C2873-1	0.12	0.57	82.4		-6	6.61E+17	27.9	0	14.0	27.9	55.9	49.9
	C2761-2	0.11	0.54	73.4		-20	6.61E+17	24.9	0	12.4	24.9	49.8	29.8
AXIAL WELDS:													
Lower Shell B1,B2,B3	37C065	0.182	0.181	94.5		-45	5.05E+17	27.8	16	13.9	42.4	70.2	25.2
Lower-Int Shell C1,C2,C3	37C065	0.182	0.181	94.5		-45	6.61E+17	32.0	16	16.0	45.3	77.3	32.3
CIRCUMFERENTIAL WELDS:													
BC	S-3986 Linde 124 Lot 3876	0.056	0.96	76.4		-32	5.05E+17	22.5	0	11.2	22.5	45.0	13.0
NOZZLES:													
N16 Forging [1] N16 Weld [1]	C2873-1 Alloy 600	0.12	0.57	82.4		-6	2.33E+17	15.6	0	7.8	15.6	31.2	25.2
BEST ESTIMATE CHEMISTRIES from BWRVIP-135 R1													
BC	[		]	79.2		-32	5.05E+17	23.3	0	11.7	23.3	46.6	14.6
INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):													
Plate [2]	[		]	65.0		-20	6.61E+17	22.0	0	11.0	22.0	44.1	24.1
Weld [3]	[		]	84.2		-45	6.61E+17	28.6	0	14.3	28.6	57.1	12.1

Notes:

[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur. The weld connecting the forging to the vessel shell is also Alloy 600 material, and is not required to be evaluated.

[2] The ISP plate material is not the vessel target material, but does occur within the Unit 2 beltline region (Lower-Intermediate Shell). Therefore, this material is considered in determining the limiting ART. Only one set of surveillance data is currently available; therefore, upon testing of a second ISP capsule scheduled for 2018, the CF can be reviewed.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 2 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

Table B-8: PBAPS Unit 2 Adjusted Reference Temperatures for up to 54 EFPY

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 1.61E+18 n/cm²
54 EFPY Peak 1/4 T fluence = 1.11E+18 n/cm²

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 1.23E+18 n/cm²
54 EFPY Peak 1/4 T fluence = 8.52E+17 n/cm²

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 5.69E+17 n/cm²
54 EFPY Peak 1/4 T fluence = 3.94E+17 n/cm²

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RTndt °F	1/4 T Fluence n/cm²	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES													
PLATES:													
Lower Shell Mark 57	C2791-2	0.12	0.52	81.4		-8	8.52E+17	31.4	0	15.7	31.4	62.8	54.8
	C2761-1	0.11	0.54	73.4		-14	8.52E+17	28.3	0	14.1	28.3	56.6	42.6
	C2873-2	0.12	0.57	82.4		-20	8.52E+17	31.8	0	15.9	31.8	63.5	43.5
Lower-Intermediate Shell Mark 58	C2894-2	0.13	0.42	85.6		-20	1.11E+18	37.6	0	17.0	34.0	71.6	51.6
	C2873-1	0.12	0.57	82.4		-6	1.11E+18	36.2	0	17.0	34.0	70.2	64.2
	C2761-2	0.11	0.54	73.4		-20	1.11E+18	32.2	0	16.1	32.2	64.4	44.4
AXIAL WELDS:													
Lower Shell B1,B2,B3	37C065	0.182	0.181	94.5		-45	8.52E+17	36.4	16	18.2	48.5	84.9	39.9
Lower-Int Shell C1,C2,C3	37C065	0.182	0.181	94.5		-45	1.11E+18	41.5	16	20.7	52.4	93.9	48.9
CIRCUMFERENTIAL WELDS:													
BC	S-3986 Linde 124 Lot 3876	0.056	0.96	76.4		-32	8.52E+17	29.5	0	14.7	29.5	58.9	26.9
NOZZLES:													
N16 Forging [1] N16 Weld [1]	C2873-1 Alloy 600	0.12	0.57	82.4		-6	3.94E+17	21.2	0	10.6	21.2	42.3	36.3
BEST ESTIMATE CHEMISTRIES from BWRVIP-135 R1													
BC	[		]	79.2		-32	8.52E+17	30.5	0	15.3	30.5	61.1	29.1
INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):													
Plate [2]	[		]	65.0		-20	1.11E+18	28.5	0	14.3	28.5	57.1	37.1
Weld [3]	[		]	84.2		-45	1.11E+18	37.0	0	18.5	37.0	73.9	28.9

Notes:
[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur. The weld connecting the forging to the vessel shell is also Alloy 600 material, and is not required to be evaluated.
[2] The ISP plate material is not the vessel target material, but does occur within the Unit 2 beltline region (Lower-Intermediate Shell). Therefore, this material is considered in determining the limiting ART. Only one set of surveillance data is currently available; therefore, upon testing of a second ISP capsule scheduled for 2018, the CF can be reviewed.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 2 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

Table B-9: PBAPS Unit 2 RPV Beltline P-T Curve Input Values for 54 EFPY

Adjusted RT_{NDT} = Initial RT_{NDT} + Shift	A = -6 + 70.2 = 64.2°F (Based on ART values)
Vessel Height	H = 875.125 inches
Bottom of Active Fuel Height	B = 216.3 inches
Vessel Radius (to base metal)	R = 125.7 inches
Minimum Vessel Thickness (without clad)	t = 6.125 inches

Table B-10: PBAPS Unit 2 Definition of RPV Beltline Region[1]

Component	Elevation (inches from RPV "0")
Shell # 2 - Top of Active Fuel (TAF)	366.31"
Shell # 1 - Bottom of Active Fuel (BAF)	216.31"
Shell # 2 – Top of Extended Beltline Region (54 EFPY)	381.1"
Shell # 1 – Bottom of Extended Beltline Region (54 EFPY)	205.8"
Circumferential Weld Between Shell #1 and Shell #2	258.69"
Circumferential Weld Between Shell #2 and Shell #3	391.69"
Centerline of Recirculation Outlet Nozzle in Shell # 1	161.5"
Top of Recirculation Outlet Nozzle N1 in Shell # 1	188.0"
Centerline of Recirculation Inlet Nozzle N2 in Shell # 1	181.0"
Top of Recirculation Inlet Nozzle N2 in Shell # 1	193.5"
Centerline of Water Level Instrumentation Nozzle in Shell # 2	366.0"
Bottom of Water Level Instrumentation Nozzle in Shell # 2	364.6"

[1] The beltline region is defined as any location where the peak neutron fluence is expected to exceed or equal 1.0e17 n/cm^2.

Based on the above, it is concluded that none of the PBAPS Unit 2 reactor vessel plates, nozzles, or welds, other than those included in the Adjusted Reference Temperature Table, are in the beltline region.

Appendix C: PBAPS Unit 3 Reactor Pressure Vessel P-T Curve Supporting Plant-Specific Information

Figure C-1: PBAPS Unit 3 Reactor Pressure Vessels

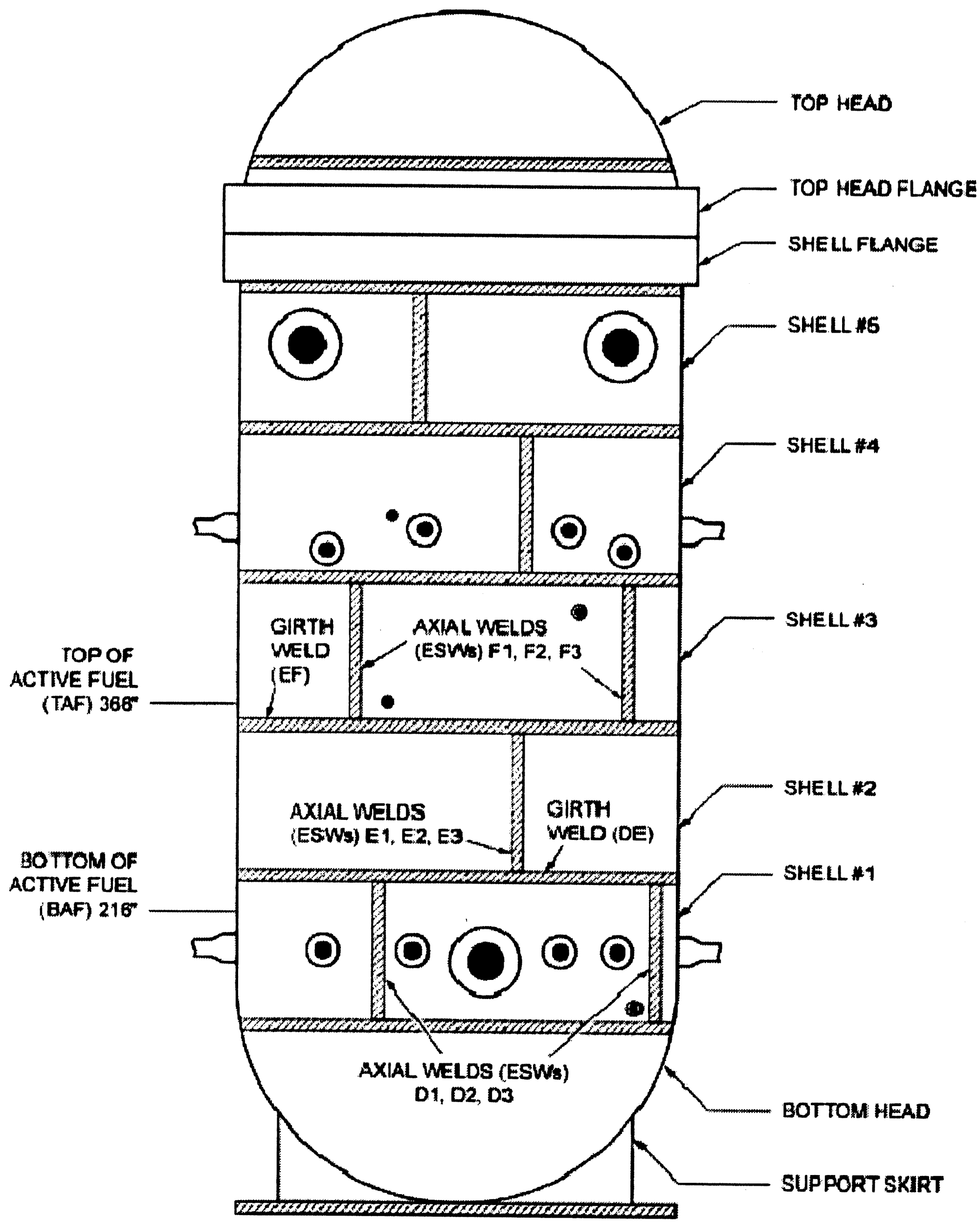

Table C-1: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Top Head and Cylindrical Shell Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
PLATES & FORGINGS:									
Top Head & Flange									
Shell Flange									
Mk. 40-146-0	123V245 ACN-97	10	L	127	109	96	40	10	10
Head Flange									
Mk. 209-139-2	124T-609 AAL-95	10	L	54	81	87	40	10	10
Top Head Dollar									
Mk. 201-127-2	C2426-4	40	L	75	74	64	70	40	40
Top Head Side Plates									
Mk. 202-139-8	C2737-1	10	L	73	64	55	40	10	10
Mk. 202-139-9	C2765-4	10	L	75	77	77	40	10	10
Mk. 202-139-10	C2765-4	10	L	74	82	78	40	10	10
Mk. 202-139-11	C1982-1	10	L	90	60	45	50	10	10
Mk. 202-139-12	C1982-1	10	L	76	74	69	40	10	10
Mk. 202-146-4	C3262-1	10	L	82	110	72	40	10	10
Shell Courses									
Shell 5									
Mk. 6-146-2	C4598-1	10	L	42	80	80	56	10	10
Mk. 6-146-3	C4679-1	10	L	65	71	80	40	10	10
Mk. 6-146-5	C4684-1	10	L	81	51	63	40	10	10
Shell 4									
Mk. 15-146-1	C4613-2	10	L	78	75	72	40	10	10
Mk. 15-146-4	C4613-1	10	L	69	78	82	40	10	10
Mk. 15-146-6	C4608-2	10	L	73	85	84	40	10	10
Shell 3									
Mk. 6-146-2	C4654-1	10	L	55	77	79	40	10	10
Mk. 6-146-4	C4689-1	10	L	96	105	60	40	10	10
Mk. 6-146-5	C4608-1	10	L	102	90	82	40	10	10
Shell 2									
Mk. 6-139-10	C2773-2	10	L	44	56	48	52	10	10
Mk. 6-139-11	C2775-1	10	L	58	66	68	40	10	10
Mk. 6-139-12	C3103-1	10	L	57	66	76	40	10	10
Shell 1									
Mk. 6-146-1	C4689-2	10	L	101	78	69	40	-10	-10
Mk. 6-146-3	C4684-2	10	L	63	60	60	40	-20	-20
Mk. 6-146-7	C4627-1	10	L	68	82	90	40	-20	-20

Table C-2: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Bottom Head and Support Skirt Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Bottom Head									
Upper Torus									
Mk. 2-127-13	C2393-3	40	L	104	124	133	70	40	40
Mk. 2-145-1	C2521-2	40	L	125	105	129	70	40	40
Mk. 2-145-4	C3069-1	40	L	65	60	68	70	40	40
Mk. A2-146-2	B7267-2	40	L	103	91	106	70	40	40
Mk. A2-146-4	B7255-2	40	L	81	28	64	114	40	54
Mk. A2-146-6	B7291-2	40	L	100	101	94	70	40	40
Lower Torus									
Mk. 4-146-3	C3436-3	40	L	72	82	75	70	40	40
Mk. 4-146-1	C2123-50	40	L	59	57	63	70	40	40
Mk. 4-146-2	C2123-50	40	L	60	50	43	84	40	40
Mk. 4-146-4	C3436-3	40	L	34	48	55	102	40	42
Bottom Head Dollar									
Mk. 1-146-1	C3354-2	40	L	65	55	66	70	40	40
Support Skirt									
Knuckle									
Mk. 24-146-1 through -4	A4846-1	40	L	122	109	127	70	30	30
Skirt									
Mk. 40-1	A4973-5BA	40	L	110	115	110	70	40	40
Mk. 30-2	A4973-5BB	40	L	132	110	110	70	40	40
Base Ring									
Mk. 41	C5489-2B	40	L	95	96	99	70	30	30

Table C-3: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Nozzle N1 through N3 Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Nozzles:									
N1 Recirc Outlet Nozzle									
N1A	AV1650 Lot N7I-6388	40	L	50	69	69	70	40	40
N1B	E31VW Lot 431H-2	40	L	73	90	95	70	40	40
N2 Recirc Inlet Nozzle									
N2A	AV1809 Lot 7J-6260B	40	L	64	78	42	86	0	26
N2B	AV1684 Lot 7I-6305A	40	L	68	116	102	70	40	40
N2C	EV9934 Lot 7G-6195	40	L	30	40	31	110	0	50
N2D	AV1660 Lot 7I-6110B	40	L	31	34	42	108	40	48
N2E	AV1809 Lot 7J-6259B	40	L	85	58	88	70	0	10
N2F	AV1660 Lot 7I-6110A	40	L	31	34	42	108	40	48
N2G	EV9947 Lot 7G-6204	40	L	72	75	112	70	0	10
N2H	AV1809 Lot 7J-6259A	40	L	85	58	88	70	0	10
N2J	AV1684 Lot 7I-6305B	40	L	68	116	102	70	40	40
N2K	AV1809 Lot 7J-6260A	40	L	64	78	42	86	0	26
N3 Steam Outlet Nozzle									
N3A	AV1973 Lot N8A-6059	40	L	35	58	74	100	40	40
N3B	AV2018 Lot N8A-6185	40	L	66	41	46	88	40	40
N3C	AV2029 Lot N8A-6186	40	L	78	34	60	102	40	42
N3D	AV1998 Lot N8A-6184	40	L	56	38	34	102	40	42

Table C-4: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Nozzle N4 through N16 Materials

COMPONENT	HEAT	TEST TEMP (°F)	Trans or Long	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
N4 Feedwater Nozzle									
N4A	AV1909 Lot 7K-6126A	40	L	85	67	89	70	40	40
N4B	AV1951 Lot 7K-6247B	40	L	85	91	68	70	40	40
N4C	EV9812 Lot 7J-6153B	40	L	62	112	76	70	40	40
N4D	AV1970 Lot 7K-6350B	40	L	80	110	98	70	40	40
N4E	AV1945 Lot 7K-6246B	40	L	80	68	106	70	40	40
N4F	AV1945 Lot N8D-6144	10	L	47	116	86	46	40	40
N5 Core Spray Nozzle									
N5A	EV26VW Lot 437H-4	40	L	107	71	89	70	40	40
N5B	EV9964 Lot N7H-6029	40	L	38	42	44	94	0	34
N6 Top Head Spray Nozzle									
N6A	BT2615 Lot 4	40	L	123	143	144	70	40	40
N6B	ZT3043 Lot 4	40	L	170	158	142	70	40	40
N7 Top Head Vent Nozzle	ZT3043 Lot 3	40	L	102	130	117	70	40	40
N8 Jet Pump Instrumentation Nozzle									
N8A & N8B	BT2615 Lot 2	40	L	132	118	120	70	40	40
N9 CRD HYD System Return Nozzle	EV9143 Lot N7H-6020A	40	L	94	92	96	70	-10	10
N10 Core Delta P& Liq Cont. Nozzle	ZT3403 Lot 1	40	L	155	154	156	70	40	40
N11 Instrumentation Nozzle									
N11A & N11B	8601 Lot 1 (Inconel)								
N12 Instrumentation Nozzle									
N12A & N12B	8601 Lot 1 (Inconel)								
N13 High Pressure Seal Leak Detector	A276N (Inconel)								
N14 Low Pressure Seal Leak Detector	A276N (Inconel)								
N15 Drain Nozzle	7579	40	L	39	34	55	102	40	42
N16 Instrumentation Nozzle									
N16A	8601 Lot 1 (Inconel)								
N16B	54316 Lot 4 (SB166)								

Table C-5: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Vertical Weld Materials

COMPONENT	HEAT	TEST TEMP (°F)	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
WELDS:								
Vertical Welds								
Bottom head Torus Ring 1 - B1, B2, B3, B4	N/A							40
Bottom Head Torus Ring 2 - C1, C2, C3, C4, C5, C6	N/A							40
Shell Course 1 - D1, D2, D3	37C065 (Electroslag Weld)							-45
Shell Course 2 - E1, E2, E3	37C065 (Electroslag Weld)							-45
Shell Course 3 - F1, F2, F3	37C065 (Electroslag Weld)							-45
Shell Course 4 - G1, G2, G3, G4, G5, G6	N/A							40
Shell Course 5 - H1, H2, H3	N/A							10
Top Head Torus - K1, K2, K3, K4, K5, K6	06R885 / D001A27A	10	56	58	67	10	-	-50
	78B743 / C727A27A	10	100	92	119	10	-	-50
	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
	06L165 / F017A27A	10	60	61	62	10	-	-50
	06L165 / F011A27A	10	84	89	92	10	-	-50
	01L333 / L908A27A	10	50	53	56	10	-	-50
	421Z0611 / L908A27A	10	66	79	80	10	-	-50

Table C-6: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Girth Weld Materials

COMPONENT	HEAT	TEST TEMP (°F)	CHARPY ENERGY (FT-LB)			(T_{50T}) (°F)	DROP WEIGHT NDT (°F)	RT_{NDT} (°F)
Girth Welds								
Dollar Plate to Torus Ring 1 - AB	88E081 / F920A27A	10	85	90	91	10	-	-50
	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
	CTY538 / A027A27A	10	94	95	95	10	-	-50
	CTY538 / B012A27A	10	77	81	87	10	-	-50
	432A6871 / B003A27A							40
	04R976 / C004A27A	10	76	79	81	10	-	-50
Torus Ring 1 to Torus Ring 2 - BC	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
Torus Ring 1 to Shell 1 - CD	78B743 / C727A27A	10	100	92	119	10	-	-50
	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
Shell 1 to Shell 2 - DE	3P4000 / 3932 / 989 Tandem	10	97	96	90	10	-	-50
	3P4000 / 3932 / 989 Single	10	97	95	88	10	-	-50
	07L669 / K004A27A	10	50	50	54	10	-	-50
Shell 2 to Shell 3 - EF	411A3531 / H004A27A	10	60	60	68	10	-	-50
	421A6811 / F023A27A	10	75	77	80	10	-	-50
	1P4217 / 3929 / 989 Tandem	10	62	68	63	10	-	-50
	1P4217 / 3929 / 989 Single	10	56	71	60	10	-	-50
Shell 3 to Shell 4 - FG	08R481 / J908A27A	10	81	81	82	10	-	-50
	01R496 / S925A27A							40
	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
	432Z0471 / B003A27A	10	100	102	106	10	-	-50
	601382 / S923A27A	10	77	78	78	10	-	-50
Shell 4 to Shell 5 - GH	1P4218 / 3929 / 989 Single	10	98	100	102	10	-	-50
	411A3531 / H004A27A	10	60	60	68	10	-	-50
Shell 5 to Shell Flange - HJ	S3986 / 3876 / 934	-20	123	92	158	-20	-	-50
	432Z0471 / B003A27A	10	100	102	106	10	-	-50
	05R938 / A020A27A							10
Head Flange to Top Head Torus - JK	CTY538 / A027A27A	10	94	95	95	10	-	-50
	08R4818 / S922A27A							10
	401Z9711 / A022A27A	10	98	99	104	10	-	-50
	1P4217 / 3929 / 989 Tandem	10	62	68	63	10	-	-50
	1P4217 / 3929 / 989 Single	10	56	71	60	10	-	-50
	411A3531 / H004A27A	10	60	60	68	10	-	-50
	421A6811 / F023A27A	10	75	77	80	10	-	-50
	06L165 / F017A27A	10	60	61	62	10	-	-50
	01L333 / F025A27A	10	50	53	56	10	-	-50
Top Head Torus to Dollar Plate - KM	01L333 / F025A27A	10	50	53	56	10	-	-50
	1P4217 / 3929 / 989 Tandem	10	62	68	63	10	-	-50
	1P4217 / 3929 / 989 Single	10	56	71	60	10	-	-50
	421A6811 / F023A27A	10	75	77	80	10	-	-50
	402A041 / B026A27A	10	86	87	95	10	-	-50
	06L165 / F017A27A	10	60	61	62	10	-	-50

Table C-7: PBAPS Unit 3 Initial RT_{NDT} Values for RPV Materials
Bolting Materials

COMPONENT	HEAT	TEST TEMP (°F)	CHARPY ENERGY (FT-LB)			LST (°F)
STUDS:						
MK-61	6720443	10	35	37	40	70
	6780382	10	43	43	43	70
	5P3629	10	43	37	30	70
	6788210	10	40	45	46	70
NUTS:	6780382	10	42	42	42	70

Table C-8: PBAPS Unit 3 Adjusted Reference Temperatures for up to 32 EFPY

Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 9.54E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 3.91E+17 n/cm^2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 1.53E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 6.28E+17 n/cm^2

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 9.48E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 3.89E+17 n/cm^2

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches= 6.125

54 EFPY Peak I.D. fluence = 5.69E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.33E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	32 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES													
PLATES:													
Lower Shell													
6-146-1	C4689-2	0.12	0.56	82.2		-10	3.89E+17	21.0	0	10.5	21.0	41.9	31.9
6-146-3	C4684-2	0.13	0.58	90.4		-20	3.89E+17	23.0	0	11.5	23.0	46.1	26.1
6-146-7	C4627-1	0.12	0.57	82.4		-20	3.89E+17	21.0	0	10.5	21.0	42.0	22.0
Lower-Intermediate Shell													
6-139-10	C2773-2	0.15	0.49	104.0		10	6.28E+17	34.3	0	17.0	34.0	68.3	78.3
6-139-11	C2775-1	0.13	0.46	86.8		10	6.28E+17	28.7	0	14.3	28.7	57.3	67.3
6-139-12	C3103-1	0.14	0.6	100.0		10	6.28E+17	33.0	0	16.5	33.0	66.1	76.1
Intermediate Shell													
6-146-5	C4608-1	0.12	0.55	82.0		10	3.91E+17	21.0	0	10.5	21.0	42.0	52.0
6-146-4	C4689-1	0.12	0.56	82.2		10	3.91E+17	21.0	0	10.5	21.0	42.1	52.1
6-146-2	C4654-1	0.11	0.55	73.5		10	3.91E+17	18.8	0	9.4	18.8	37.6	47.6
AXIAL WELDS:													
Lower Shell D1,D2,D3	37C065	0.182	0.181	94.5		-45	3.89E+17	24.1	16	12.0	40.1	64.1	19.1
Lower-Int Shell E1,E2,E3	37C065	0.182	0.181	94.5		-45	6.28E+17	31.2	16	15.6	44.7	75.9	30.9
Intermediate Shell F1,F2,F3	37C065	0.182	0.181	94.5		-45	3.91E+17	24.2	16	12.1	40.1	64.3	19.3
CIRCUMFERENTIAL WELDS:													
Lower to Lower-Int DE	3P4000 Linde 124 Lot 3932	0.020	0.934	27.0		-50	3.89E+17	6.9	0	3.4	6.9	13.8	-36.2
Lower-Int to Intermediate EF	1P4217 Linde 124 Lot 3929	0.102	0.942	136.9		-50	3.91E+17	35.0	0	17.5	35.0	70.0	20.0
NOZZLES:													
N16 [1]	C4689-1	0.12	0.56	82.2		10	2.33E+17	15.5	0	7.8	15.5	31.1	41.1
BEST ESTIMATE CHEMISTRIES from BWRVIP-135 R1													
DE	[		]	27.0		-50	3.89E+17	6.9	0	3.4	6.9	13.8	-36.2
EF	[		]	139.3		-50	3.91E+17	35.6	0	17.8	35.6	71.3	21.3
INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):													
Plate [2]	[		]	111.25		10	6.28E+17	36.7	0	17.0	34.0	70.7	80.7
Weld [3]	[		]	82.0		-45	6.28E+17	27.1	0	13.5	27.1	54.2	9.2
Weld [3]	[		]	108.0		-45	6.28E+17	35.7	0	17.8	35.7	71.3	26.3

Notes:

[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur. The weld connecting the forging to the vessel shell is also Alloy 600 material, and is not required to be evaluated.

[2] The ISP plate material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

[4] The ISP best estimate chemistry is used.

Table C-9: PBAPS Unit 3 Adjusted Reference Temperatures for up to 54 EFPY

Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125
54 EFPY Peak I.D. fluence = 9.54E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 6.61E+17 n/cm^2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches= 6.125
54 EFPY Peak I.D. fluence = 1.53E+18 n/cm^2
54 EFPY Peak 1/4 T fluence = 1.06E+18 n/cm^2

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches= 6.125
54 EFPY Peak I.D. fluence = 9.48E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 6.56E+17 n/cm^2

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches= 6.125
54 EFPY Peak I.D. fluence = 5.69E+17 n/cm^2
54 EFPY Peak 1/4 T fluence = 3.94E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Adjusted CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	54 EFPY RT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES													
PLATES:													
Lower Shell													
6-146-1	C4689-2	0.12	0.56	82.2		-10	6.56E+17	27.8	0	13.9	27.8	55.6	45.6
6-146-3	C4684-2	0.13	0.58	90.4		-20	6.56E+17	30.6	0	15.3	30.6	61.1	41.1
6-146-7	C4627-1	0.12	0.57	82.4		-20	6.56E+17	27.9	0	13.9	27.9	55.7	35.7
Lower-Intermediate Shell													
6-139-10	C2773-2	0.15	0.49	104.0		10	1.06E+18	44.6	0	17.0	34.0	78.6	88.6
6-139-11	C2775-1	0.13	0.46	86.8		10	1.06E+18	37.2	0	17.0	34.0	71.2	81.2
6-139-12	C3103-1	0.14	0.6	100.0		10	1.06E+18	42.9	0	17.0	34.0	76.9	86.9
Intermediate Shell													
6-146-5	C4608-1	0.12	0.55	82.0		10	6.61E+17	27.8	0	13.9	27.8	55.6	65.6
6-146-4	C4689-1	0.12	0.56	82.2		10	6.61E+17	27.9	0	13.9	27.9	55.7	65.7
6-146-2	C4654-1	0.11	0.55	73.5		10	6.61E+17	24.9	0	12.5	24.9	49.8	59.8
AXIAL WELDS:													
Lower Shell D1,D2,D3	37C065	0.182	0.181	94.5		-45	6.56E+17	31.9	16	16.0	45.2	77.2	32.2
Lower-Int Shell E1,E2,E3	37C065	0.182	0.181	94.5		-45	1.06E+18	40.5	16	20.2	51.6	92.1	47.1
Intermediate Shell F1,F2,F3	37C065	0.182	0.181	94.5		-45	6.61E+17	32.0	16	16.0	45.3	77.3	32.3
CIRCUMFERENTIAL WELDS:													
Lower to Lower-Int DE	3P4000 Linde 124 Lot 3932	0.020	0.934	27.0		-50	6.56E+17	9.1	0	4.6	9.1	18.3	-31.7
Lower-Int to Intermediate EF	1P4217 Linde 124 Lot 3929	0.102	0.942	136.9		-50	6.61E+17	46.4	0	23.2	46.4	92.8	42.8
NOZZLES:													
N16 [1]	C4689-1	0.12	0.56	82.2		10	3.94E+17	21.1	0	10.6	21.1	42.2	52.2
BEST ESTIMATE CHEMISTRIES from BWRVIP-135 R1													
DE	[		]	27.0		-50	6.56E+17	9.1	0	4.6	9.1	18.3	-31.7
EF	[		]	139.3		-50	6.61E+17	47.2	0	23.6	47.2	94.5	44.5
INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):													
Plate [2]	[		]	111.25		10	1.06E+18	47.7	0	17.0	34.0	81.7	91.7
Weld [3]	[		]	82.0		-45	1.06E+18	35.1	0	17.6	35.1	70.3	25.3
Weld [3]	[		]	108.0		-45	1.06E+18	46.3	0	23.1	46.3	92.6	47.6

Notes:

[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur. The weld connecting the forging to the vessel shell is also Alloy 600 material, and is not required to be evaluated.

[2] The ISP plate material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG1.99 for the ISP chemistry.

[4] The ISP best estimate chemistry is used.

Table C-10: PBAPS Unit 3 RPV Beltline P-T Curve Input Values for 54 EFPY

Adjusted RT_{NDT} = Initial RT_{NDT} + Shift	A = 10 + 78.6 = 88.6°F (Based on ART values)
Vessel Height	H = 875.3125 inches
Bottom of Active Fuel Height	B = 216.3 inches
Vessel Radius (to base metal)	R = 125.7 inches
Minimum Vessel Thickness (without clad)	t = 6.125 inches

Table C-11: PBAPS Unit 3 Definition of RPV Beltline Region[1]

Component	Elevation (inches from RPV "0")
Shell # 2 - Top of Active Fuel (TAF)	366.31"
Shell # 1 - Bottom of Active Fuel (BAF)	216.31"
Shell # 2 – Top of Extended Beltline Region (54 EFPY)	381.1"
Shell # 1 – Bottom of Extended Beltline Region (54 EFPY)	205.8"
Circumferential Weld Between Shell #1 and Shell #2	258.69"
Circumferential Weld Between Shell #2 and Shell #3	391.69"
Centerline of Recirculation Outlet Nozzle in Shell # 1	161.5"
Top of Recirculation Outlet Nozzle N1 in Shell # 1	188.0"
Centerline of Recirculation Inlet Nozzle N2 in Shell # 1	181.0"
Top of Recirculation Inlet Nozzle N2 in Shell # 1	193.5"
Centerline of Water Level Instrumentation Nozzle in Shell # 2	366.0"
Bottom of Water Level Instrumentation Nozzle in Shell # 2	364.6"

[1] The beltline region is defined as any location where the peak neutron fluence is expected to exceed or equal 1.0e17 n/cm^2.

Based on the above, it is concluded that none of the PBAPS Unit 3 reactor vessel plates, nozzles, or welds, other than those included in the Adjusted Reference Temperature Table, are in the beltline region.

Appendix D: PBAPS Unit 2 and Unit 3 Reactor Pressure Vessel P-T Curve Checklist

Table D-1: PBAPS Unit 2 Checklist

Parameter	Completed	Comments/Resolutions/Clarifications
Initial RT_{NDT}		
Initial RT_{NDT} has been determined for PBAPS Unit 2 for all vessel materials including plates, flanges, forgings, studs, nuts, bolts, welds. Include explanation (including methods/sources) of any exceptions, resolution of discrepant data (e.g., deviation from originally reported values).	☒	Additional plant-specific information was located for girth welds, nozzle welds, and appurtenance welds. This information is included in Appendix B. All other information remains unchanged from previous submittals.
Appendix B contains tables of all Initial RT_{NDT} values for PBAPS Unit 2	☒	
Has any non-PBAPS Unit 2 initial RT_{NDT} information (e.g., ISP, comparison to other plant) been used?	☒	Plate heat C2761-2 and weld heat PB2 ESW information was obtained from the ISP database. These materials are not identical heats to the target vessel material and, in accordance with the ISP guidance, this data was not used in determining the limiting ART.
If deviation from the LTR process occurred, sufficient supporting information has been included (e.g., Charpy V-Notch data used to determine an Initial RT_{NDT}).	☒	No deviations from the LTR process.
All previously published Initial RT_{NDT} values from sources such as the GL88-01, RVID, FSAR, etc., have been reviewed.	☒	RVID was reviewed; all initial RT_{NDT} values agree; no further review was performed
Adjusted Reference Temperature (ART)		
Sigma I (standard deviation for Initial RT_{NDT}) is 0°F unless the RT_{NDT} was obtained from a source other than CMTRs. If σ_I is not equal to 0, reference/basis has been provided.	☒	Sigma I for the electroslag weld (ESW) axial shell welds is equal to 16°F, and is consistent with previous submittals and the PBAPS UFSAR.
Sigma Δ (standard deviation for ΔRT_{NDT}) is determined per RG 1.99, Rev. 2	☒	

Parameter	Completed	Comments/Resolutions/Clarifications
Chemistry has been determined for all vessel beltline materials including plates, forgings (if applicable), and welds for PBAPS Unit 2. Include explanation (including methods/sources) of any exceptions, resolution of discrepant data (e.g., deviation from originally reported values).	☒	No deviations from previously reported values.
Non-PBAPS Unit 2 chemistry information (e.g., ISP, comparison to other plant) used has been adequately defined and described.	☒	Heat S3986 has been evaluated using best estimate chemistry from the ISP. Chemistry information for heats C2761-2 and PB2 ESW was obtained from the ISP database.
For any deviation from the LTR process, sufficient information has been included.	☒	No deviations from the LTR process.
All previously published chemistry values from sources such as the GL88-01, RVID, FSAR, etc., have been reviewed.	☒	RVID was reviewed; all initial RT_{NDT} values agree; no further review was performed
The fluence used for determination of ART and any extended beltline region was obtained using an NRC-approved methodology.	☒	
The fluence calculation provides an axial distribution to allow determination of the vessel elevations that experience fluence of 1.0e17 n/cm^2 both above and below active fuel.	☒	
The fluence calculation provides an axial distribution to allow determination of the fluence for intermediate locations such as the beltline girth weld (if applicable) or for any nozzles within the beltline region.	☒	
All materials within the elevation range where the vessel experiences a fluence ≥1.0e17 n/cm^2 have been included in the ART calculation. All initial RT_{NDT} and chemistry information is available or explained.	☒	The N16 nozzle forging is Alloy 600 and does not require evaluation for fracture toughness. Therefore, the initial RT_{NDT} and chemistry information of the shell that includes the N16 nozzle is used.

Parameter	Completed	Comments/Resolutions/Clarifications
Discontinuities		
The discontinuity comparison has been performed as described in Section 4.3.2.1 of the LTR. Any deviations have been explained.	☒	
Discontinuities requiring additional components (such as nozzles) to be considered part of the beltline have been adequately described. It is clear which curve is used to bound each discontinuity.	☒	
Appendix G of the LTR describes the process for considering a thickness discontinuity, both beltline and non-beltline. If there is a discontinuity in the PBAPS Unit 2 vessel that requires such an evaluation, the evaluation was performed. The affected curve was adjusted to bound the discontinuity, if required.	☒	The thickness discontinuity evaluation demonstrated that no additional adjustment is required; the curves bound the discontinuity stresses.
Appendix H of the LTR defines the basis for the CRD Penetration curve discontinuity and the appropriate transient application. The PBAPS Unit 2 evaluation bounds the requirements of Appendix H.	☒	
Appendix J of the LTR defines the basis for the Water Level Instrumentation Nozzle curve discontinuity and the appropriate transient application. The PBAPS Unit 2 evaluation bounds the requirements of Appendix J.	☒	

Table D-2: PBAPS Unit 3 Checklist

Parameter	Completed	Comments/Resolutions/Clarifications
Initial RT_{NDT}		
Initial RT_{NDT} has been determined for PBAPS Unit 3 for all vessel materials including plates, flanges, forgings, studs, nuts, bolts, welds. Include explanation (including methods/sources) of any exceptions, resolution of discrepant data (e.g., deviation from originally reported values).	☒	
Appendix C contains tables of all Initial RT_{NDT} values for PBAPS Unit 3	☒	
Has any non-PBAPS Unit 3 initial RT_{NDT} information (e.g., ISP, comparison to other plant) been used?	☒	Plate heat B0673-1 and weld heat 5P6756 information was obtained from the ISP database. These materials are not identical heats to the target vessel material and, in accordance with the ISP guidance, this data was not used in determining the limiting ART.
If deviation from the LTR process occurred, sufficient supporting information has been included (e.g., Charpy V-Notch data used to determine an Initial RT_{NDT}).	☒	No deviations from the LTR process.
All previously published Initial RT_{NDT} values from sources such as the GL88-01, RVID, FSAR, etc., have been reviewed.	☒	RVID was reviewed; all initial RT_{NDT} values agree; no further review was performed
Adjusted Reference Temperature (ART)		
Sigma I (standard deviation for Initial RT_{NDT}) is 0°F unless the RT_{NDT} was obtained from a source other than CMTRs. If σ_I is not equal to 0, reference/basis has been provided.	☒	Sigma I for the electroslag weld (ESW) axial shell welds is equal to 16°F, and is consistent with previous submittals and the PBAPS UFSAR.
Sigma Δ (standard deviation for ΔRT_{NDT}) is determined per RG 1.99, Rev. 2	☒	

Parameter	Completed	Comments/Resolutions/Clarifications
Chemistry has been determined for all vessel beltline materials including plates, forgings (if applicable), and welds for PBAPS Unit 3 Include explanation (including methods/sources) of any exceptions, resolution of discrepant data (e.g., deviation from originally reported values).	☒	No deviations from previously reported values.
Non- PBAPS Unit 3 chemistry information (e.g., ISP, comparison to other plant) used has been adequately defined and described.	☒	Heats 3P4000 and 1P4217 have been evaluated using best estimate chemistry from the ISP. Chemistry information for heats B0673-1 and 5P6756 was obtained from the ISP database.
For any deviation from the LTR process, sufficient information has been included.	☒	No deviations from the LTR process.
All previously published chemistry values from sources such as the GL88-01, RVID, FSAR, etc., have been reviewed.	☒	RVID was reviewed; all initial RT_{NDT} values agree; no further review was performed
The fluence used for determination of ART and any extended beltline region was obtained using an NRC-approved methodology.	☒	
The fluence calculation provides an axial distribution to allow determination of the vessel elevations that experience fluence of 1.0e17 n/cm^2 both above and below active fuel.	☒	
The fluence calculation provides an axial distribution to allow determination of the fluence for intermediate locations such as the beltline girth weld (if applicable) or for any nozzles within the beltline region.	☒	
All materials within the elevation range where the vessel experiences a fluence ≥1.0e17 n/cm^2 have been included in the ART calculation. All initial RT_{NDT} and chemistry information is available or explained.	☒	The N16 nozzle forging is Alloy 600 and does not require evaluation for fracture toughness. Therefore, the initial RT_{NDT} and chemistry information of the shell that includes the N16 nozzle is used.

Parameter	Completed	Comments/Resolutions/Clarifications
Discontinuities		
The discontinuity comparison has been performed as described in Section 4.3.2.1 of the LTR. Any deviations have been explained.	☒	
Discontinuities requiring additional components (such as nozzles) to be considered part of the beltline have been adequately described. It is clear which curve is used to bound each discontinuity.	☒	
Appendix G of the LTR describes the process for considering a thickness discontinuity, both beltline and non-beltline. If there is a discontinuity in the PBAPS Unit 3 vessel that requires such an evaluation, the evaluation was performed. The affected curve was adjusted to bound the discontinuity, if required.	☒	The thickness discontinuity evaluation demonstrated that no additional adjustment is required; the curves bound the discontinuity stresses.
Appendix H of the LTR defines the basis for the CRD Penetration curve discontinuity and the appropriate transient application. The PBAPS Unit 3 evaluation bounds the requirements of Appendix H.	☒	
Appendix J of the LTR defines the basis for the Water Level Instrumentation Nozzle curve discontinuity and the appropriate transient application. The PBAPS Unit 3 evaluation bounds the requirements of Appendix J.	☒	

ATTACHMENT 6

Non-Proprietary Version - Responses to Requests for Additional Information

Attachment 2

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

NEIL WILMSHURST
Vice President and
Chief Nuclear Officer

April 17, 2012

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Andrew Hon

Subject: Request for Withholding of the following Proprietary Information Included in:

Enclosure 1
7491-1-2S83W9-HE0-1
Peach Bottom RAI Response – Proprietary

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Report and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence for informational purposes regarding a submittal to the NRC by Exelon Corporation. The Proprietary Information is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Proprietary Information provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 704-595-2732. Questions on the content of the Proprietary Information should be directed to Randy Stark of EPRI at (650) 855-2122.

Sincerely,

c: Sheldon Stuchell, NRC (Sheldon.stuchell@nrc.gov)

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ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In::

Enclosure 1
7491-1-2S83W9-HE0-1
Peach Bottom RAI Response – Proprietary

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information. EPRI made a substantial economic investment to develop the Proprietary Information and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

c. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

"'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

e. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 4-17-2012.

[signature]

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 17th day of April, 2012 by Neil Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016.

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Louis M. Quintana, state as follows:

(1) I am the Program Manager, EHS, GE-Hitachi Nuclear Energy Americas LLC (GEH). I have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH letter, 7491-1-2S83W9-HEO-1, Larry Beese (GEH) to Jeremy Searer (Exelon), "GEH Responses to Potential RAIs," dated April 4, 2012. The proprietary information in Enclosure 1 entitled, "Peach Bottom RAI Response-Proprietary," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]]. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH and/or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains results of an analysis performed by GEH to support the Peach Bottom Pressure and Temperature Limits Report (PTLR) Technical Specification 5.6.7 changes. This analysis is part of the GEH PTLR methodology. Development of the PTLR methodology and the supporting analysis techniques and information, and their application to the design, modification, and processes were achieved at a significant cost to GEH.

The development of the evaluation methodology along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 4th day of April 2012.

Louis M. Quintana

Louis M. Quintana
Program Manager, EHS
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd.
Wilmington, NC 28401
Louis.Quintana@ge.com

GGNS RAI #4

Do the PT limit curves include a hydrostatic pressure adjustment for the column of water in a full RPV? If so, provide the pressure head used in the PT limit curve analysis.

Response

Yes, the pressure head for PBAPS is 24 psig. This is determined using the height of the vessel and the elevation of the bottom of active fuel. Similarly, the full vessel pressure head is 30 psig. This value and the equation used can be found in *Development of Reactor Pressure Vessel Pressure-Temperature Curves*, NEDC-33178P-A, Revision 1, Sections 4.3.2.1.1 and 4.3.2.2.2, respectively.

GGNS RAI #5

Address inconsistencies between the statement that "the PT curves are beltline (A1224-1 plate) limited above 1330 psig for Curve A for 35 EFPY..." and the NRC staff determination that the PT curves are beltline (A1224-1 plate) limited ~1360 psig from data in Table 1 of GNRO-2010/00056.

Response

This RAI is not applicable to PBAPS as it addresses a plant-specific GGNS statement.

GGNS RAI #6

Provide the surveillance data and the analysis of the surveillance data used to determine ART from Reference 6.3 (BWRVIP-135, Revision 1 "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations"), as required by NEDC-33178P-A.

Response

BWRVIP-135, Revision 1 provides the surveillance data considered in determining the chemistry and any adjusted Chemistry Factors (CF) for the beltline materials.

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]

For PBAPS Unit 2, the Integrated Surveillance Program (ISP) representative plate, heat [ ], is not the target plate, but is a plate heat in the beltline region. This heat was contained in one (1) PBAPS Unit 2 capsule that has been tested and analyzed. Data was obtained from two (2) specimens from the capsule and one data set from the plant-specific Certified Material Test Report (CMTR). The resultant chemistry is [ ] Cu and [ ] Ni. The CF from Regulatory Guide 1.99, Revision 2 (RG1.99) is 65°F. No fitted CF has been determined as there has only been one (1) capsule tested. Therefore, the Adjusted Reference Temperature (ART) table evaluated the ISP plate material using the RG1.99 CF. This material was considered in determining the limiting ART for the PT curves; however, this is not the limiting material.

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]
[ ]
[ ]

For PBAPS Unit 2, the ISP representative weld, heat [ ], is not the target weld. This heat was contained in one (1) PBAPS Unit 2 capsule that has been tested and analyzed. Data was obtained from two (2) specimens from the capsule. The resultant chemistry is [ ] Cu and [ ] Ni. The CF from RG1.99 is 84.2°F. No fitted CF has been determined as there has only been one (1) capsule tested. Therefore, the ART table evaluated the ISP weld material using the RG1.99 CF. This material was not considered in determining the limiting ART for the PT curves.

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]
[ ]

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]

For PBAPS Unit 3, the ISP representative plate, heat [ ], is not the target plate. This heat was contained in two (2) Duane Arnold capsules that have been tested and analyzed. Data was obtained from five (5) specimens from the capsules and one (1) data set from the plant-specific Certified Material Test Report (CMTR). The resultant chemistry is [ ] Cu and [ ] Ni. The CF from RG1.99 is 111.25°F; the fitted CF is 148.71°F. The fitted CF for the plate material is not applicable for the PBAPS Unit 3 ART evaluation because the ISP plate heat, [ ], is not the target vessel plate. Therefore, the ART table evaluated the ISP plate material using the RG1.99 CF. This material was not considered in determining the limiting ART for the PT curves.

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]
[ ]
[ ]
[ ]
[ ]
[ ]

For PBAPS Unit 3, the ISP representative weld, heat [ ], is not the target weld. This heat was contained in one (1) River Bend and two (2) Supplemental Surveillance Program (SSP) capsules that have been tested and analyzed. Data was obtained from three (3) specimens from the capsules. The resultant chemistry is [ ] Cu and [ ] Ni. The CF from RG1.99 is 82°F; the fitted CF is 116.9°F. An adjusted CF for the weld material is not applicable for the PBAPS Unit 3 ART evaluation because the ISP weld heat, [ ], is not the target vessel weld. Therefore, the ART table evaluated the ISP weld material using the RG1.99 CF. This material was not considered in determining the limiting ART for the PT curves.

Excerpt from BWRVIP-135, Revision 1 (used by permission)

[ ]
[ ]
[ ]
[ ]
[ ]
[ ]
[ ]

GGNS RAI #7

Provide additional detail for the non-beltline analysis conducted in the following areas in order for the NRC staff to complete independent verification of the proposed PT limits:

a. Identify limiting materials for the Reference Temperature for Nil Ductility Transition (RT_{NDT}) values used to shift the generic Bottom Head and Upper Vessel PT curves when applying NEDC-33178P-A.
b. The NRC staff identified a limiting RT_{NDT} of 10°F for the Bottom Head Torus Plates, while GGNS assumed a RT_{NDT} of 24.6°F for the Bottom Head Curve B. Support all RT_{NDT} values reported by providing details of any plant-specific analysis conducted.
c. Explain minor differences in assumed RT_{NDT} values for the Bottom Head. Specifically Curves A and C assume a limiting RT_{NDT} of 19°F, while Curve B assumes a limiting RT_{NDT} of 24.6°F.
d. Which region of the RPV is limiting for Curve C < 312 psig.

Response

In order to determine how much to shift the Pressure-Temperature (PT) curves, an evaluation is performed using Tables 4-4a and 4-5a from NEDC-33178P-A. These tables define the required Temperature minus Reference Temperature of Nil Ductility Transition ($T\text{-}RT_{NDT}$) that is used to adjust the non-shifted curves. Each component listed in these tables is evaluated using the plant-specific initial RT_{NDT} for each component. The required temperature is then determined by adding the $T\text{-}RT_{NDT}$ to the plant-specific RT_{NDT}, thereby resulting in the required T for the curve. As the upper vessel curve is initially based on the non-shifted feedwater (FW) nozzle $T\text{-}RT_{NDT}$, all resulting T values are compared to the FW nozzle T. The difference between the maximum T and the FW nozzle $T\text{-}RT_{NDT}$ is used to shift the upper vessel curve. The same method is applied for the Control Rod Drive (CRD) curve. In this manner, it is assured that each curve bounds the maximum discontinuity that is represented.

For the PBAPS Unit 2 upper vessel curve, the maximum T value from the method described above is [[ ]]. The initial required $T\text{-}RT_{NDT}$ for the [[ ]]; this is then adjusted by the PBAPS Unit 2-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the next limiting value, the [[ ]], the required $T\text{-}RT_{NDT}$ is [[ ]], which is added to the [[ ]]. It is seen that the resulting T required for the [[ ]]. As [[ ]] is limiting, the PBAPS Unit 2 upper vessel curve is based on an RT_{NDT} of [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-4a; only the limiting case is presented here.

For the PBAPS Unit 2 bottom head or CRD [[ ]], respectively), the maximum T value from the method described above is [[ ]]. The required T-RT_{NDT} for the [[ ]]; this is adjusted by the PBAPS Unit 2-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the CRD values, the required T-RT_{NDT} is [[ ]], which is added to the [[ ]]. It is seen that the resulting T required for the bottom head is [[ ]]. As [[ ]] than [[ ]], the PBAPS Unit 2 bottom head (CRD) curve is based on an [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-5a; only the limiting case is presented here.

Appendix H of NEDC-33178P-A contains the details of an analysis performed to determine the baseline requirement (non-shifted) for the [[ ]]. It can be seen in Section H.5 of Appendix H that the stresses developed in this finite element analysis demonstrated that the [[ ]], resulting in a baseline non-shifted required T-RT_{NDT} of [[ ]]. Therefore, considering the determination of the required shift from the paragraph above for [[ ]], calculations for all components listed in Table 4-5a were compared to the CRD T, which is [[ ]] (where [[ ]] materials). Therefore, the shift for the bottom head [[ ]].

For Curve C, the upper vessel and beltline regions are bounding at pressures up to 40 psig. For pressures between 40 psig and 312.5 psig, the upper vessel is bounding; this is true for both 32 and 54 EFPY.

For the PBAPS Unit 3 upper vessel curve, the maximum T value from the method described above is [[ ]]. The initial required T-RT_{NDT} for the [[ ]]; this is then adjusted by the PBAPS Unit 3-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the [[ ]], the required T-RT_{NDT} is [[ ]], which is added to the [[

]]. It is seen that the resulting T required for the [[ ]] is [[ ]]. As [[ ]] is limiting, the PBAPS Unit 3 upper vessel curve is based on an RT_{NDT} of [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-4a; only the limiting case is presented here.

For the PBAPS Unit 3 bottom head or CRD [[ ]], respectively), the maximum T value from the method described above is [[ ]]. The required T-RT_{NDT} for the [[ ]]; this is adjusted by the PBAPS Unit 3-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the next limiting value, the [[ ]], the required T-RT_{NDT} is [[ ]], which is added to the [[ ]]. It is seen that the resulting T required for the bottom head is [[ ]]. As [[ ]] than [[ ]], the PBAPS Unit 3 bottom head (CRD) curve is based on an [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-5a; only the limiting case is presented here.

Appendix H of NEDC-33178P-A contains the details of an analysis performed to determine the baseline requirement (non-shifted) for the [[ ]]. It can be seen in Section H.5 of Appendix H that the stresses developed in this finite element analysis demonstrated that the [[ ]], resulting in a baseline non-shifted required T-RT_{NDT} of [[ ]]. Therefore, considering the determination of the required shift from the paragraph above for [[ ]], calculations for all components listed in Table 4-5a were compared to the CRD T, which is [[ ]] (where [[ ]] materials). Therefore, the shift for the bottom head [[ ]].

For Curve C, the upper vessel and beltline regions are bounding at pressures up to 40 psig. For pressures between 40 psig and 312.5 psig, the upper vessel is bounding; this is true for both 32 and 54 EFPY.

GGNS RAI #8

Attachment 7 identifies nozzle N12 as a beltline water level instrument nozzle and notes that an evaluation was conducted using the limiting material properties for the adjoining shell ring, which appears to be appropriate as nozzle N12 is identified as austenitic. Provide details of this evaluation which demonstrates that the drill hole for the beltline water level instrument nozzle is not limiting.

Response

The PBAPS Unit 2 and Unit 3 N12 nozzle is fabricated from Alloy 600 material. Appendix J of NEDC-33178P-A provides detailed results of an analysis performed for the water level instrumentation nozzle that provides the required stresses for the drill hole in the shell plate. These stresses were used to generate a specific curve applicable for the water level instrumentation nozzle to assure that this location is bounded in the PT curves. For PBAPS Unit 2 and Unit 3, the water level instrumentation nozzles are [[ ]] for 32 and 54 EFPY.

GGNS RAI #9

Provide details on any plant-specific feedwater nozzle evaluation conducted in support of the proposed PT limits or explain why plant-specific evaluation was not required.

Response

An evaluation was performed for the feedwater nozzle as described in Section 4.3.2.1.3 of NEDC-33178P-A. This evaluation confirmed that the feedwater discontinuity bounds the other discontinuities defined in Table 4-4a of NEDC-33178P-A. The first part of the evaluation is as described in the response to RAI #7, where it is assured that the limiting component that is represented by the upper vessel nozzle curve is bounded. A second evaluation was performed using the PBAPS-specific feedwater nozzle dimensions; this evaluation is shown below to demonstrate that the [[ ]] curve is applicable to PBAPS Unit 2:

Vessel radius to base metal, R_v	[[
Vessel thickness, t_v	
Vessel pressure, P_v	
Pressure stress = PR/t = [[ ]]	
Dead Weight + Thermal Restricted Free End stress	
Total Stress = [[ ]]	]]

The factor F (a/r_n) from Figure A5-1 of "PVRC Recommendations on Toughness Requirements for Ferritic Materials", Welding Research Council Bulletin 175, August 1972 (WRC-175) is determined where:

$a = \frac{1}{4} (t_n^2 + t_v^2)^{\frac{1}{2}}$	[[
t_n = thickness of nozzle	
t_v = thickness of vessel	
r_n = apparent radius of nozzle = $r_i + 0.29*r_c$	
r_i = actual inner radius of nozzle	
r_c = nozzle radius (nozzle corner radius)	]]

Therefore, a/r_n = [[ ]]. The value F (a/r_n), taken from Figure A5-1 of WRC-175 for an [[ ]]. Including the safety factor of 1.5, the stress intensity factor, K_I, is $1.5\ \sigma\ (\pi a)^{1/2} * F(a/r_n)$:

Nominal K_I = 1.5 * [[ ]]

A detailed upper vessel example calculation for core not critical conditions is provided in Section 4.3.2.1.4 of NEDC-33178P-A. Section 4.3.2.1.3 of NEDC-33178P-A, presents the [[ ]] FW nozzle evaluation upon which the baseline non-shifted upper vessel PT curve is based. It can be seen that the nominal K_I from this evaluation is [[ ]]. Therefore, it has been shown that the nominal K_I for the PBAPS Unit 2-specific FW nozzle is equal to the [[ ]] K_I, demonstrating applicability of the FW nozzle curve for PBAPS Unit 2.

The same evaluation is shown below to demonstrate that the [[ ]] curve is applicable to PBAPS Unit 3:

Vessel radius to base metal, R_v	[[
Vessel thickness, t_v	
Vessel pressure, P_v	
Pressure stress = PR/t = [[ ]]	
Dead Weight + Thermal Restricted Free End stress	
Total Stress = [[ ]]	]]

The factor F (a/r_n) from Figure A5-1 of "PVRC Recommendations on Toughness Requirements for Ferritic Materials", Welding Research Council Bulletin 175, August 1972 (WRC-175) is determined where:

$a = ¼ (t_n^2 + t_v^2)^{½}$	[[
t_n = thickness of nozzle	
t_v = thickness of vessel	
r_n = apparent radius of nozzle = $r_i + 0.29*r_c$	
r_i = actual inner radius of nozzle	
r_c = nozzle radius (nozzle corner radius)	]]

Therefore, a/r_n = [[ ]]. The value F (a/r_n), taken from Figure A5-1 of WRC-175 for an [[ ]]. Including the safety factor of 1.5, the stress intensity factor, K_I, is $1.5\ \sigma\ (\pi a)^{1/2} * F(a/r_n)$:

Nominal K_I = 1.5 * [[ ]]

A detailed upper vessel example calculation for core not critical conditions is provided in Section 4.3.2.1.4 of NEDC-33178P-A. Section 4.3.2.1.3 of NEDC-33178P-A, presents the [[ ]] FW nozzle evaluation upon which the baseline non-shifted upper vessel PT curve is based. It can be seen that the nominal K_I from this evaluation is

[[ ]]. Therefore, it has been shown that the nominal K_I for the PBAPS Unit 3-specific FW nozzle is bounded by the [[ ]] K_I, demonstrating applicability of the FW nozzle curve for PBAPS Unit 3.