FENOC™

FirstEnergy Nuclear Operating Company

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June 23, 2014
L-14-150

10 CFR 50.90

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

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
<u>Request for Licensing Action to Amend Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits"</u>

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) is requesting an amendment to the Perry Nuclear Power Plant (PNPP) Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

An evaluation of the proposed amendment is provided as an enclosure. FENOC is requesting Nuclear Regulatory Commission (NRC) staff approval by June 25, 2015. Implementation of the amendment by FENOC is planned within 90 days of its approval.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 23, 2014.

Sincerely,

Ernest J. Harkness

Enclosure: Evaluation of Proposed Request for Licensing Action

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Subject: Request for Licensing Action to Amend Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

Attachments:

1. Proposed Technical Specification Changes (MARK-UP)
2. Proposed Technical Specification Changes (RETYPED)
3. Proposed Technical Specification Bases Changes
4. Development of P/T Limit Curves for Technical Specification 3.4.11

1.0 SUMMARY DESCRIPTION

This evaluation supports a FirstEnergy Nuclear Operating Company (FENOC) request to amend Perry Nuclear Power Plant (PNPP) Technical Specification (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," by updating the pressure and temperature limit requirements. The figures affected by the proposed amendment include the following:

- Figure 3.4.11-1(a): Pressure Test Curve (Curve A) (Valid up to 22 EFPY - Unit 1)
- Figure 3.4.11-1(b): Non-Nuclear Heatup/Cooldown (Curve B) (Valid up to 22 EFPY - Unit1)
- Figure 3.4.11-1(c): Core Critical Operation (Curve C) (Valid up to 22 EFPY - Unit 1)
- Figure 3.4.11-1(d): Pressure Test Curve (Curve A) (Valid up to 32 EFPY - Unit 1)
- Figure 3.4.11-1(e): Non-Nuclear Heatup/Cooldown (Curve B) (Valid up to 32 EFPY - Unit 1)
- Figure 3.4.11-1(f): Core Critical Operation (Curve C) (Valid up to 32 EFPY - Unit 1)

The three figures that are valid up to 22 effective full power years (EFPY) are deleted per this license amendment request, and the three figures that are valid up to 32 EFPY become Figures 3.4.11-1(a), (b), and (c).

By updating the figures, the proposed amendment addresses two issues related to the pressure and temperature (P/T) limit curves. First, it was determined that an existing water level instrument nozzle (WLIN) in the reactor pressure vessel (RPV) beltline region was not included in the data used to develop the current PNPP P/T curves. Second, it was determined that the reactor coolant system (RCS) experiences a vacuum under certain conditions. The current TS 3.4.11 figures reflect gauge pressure, do not include a vacuum region, and do not depict P/T limit curves that extend into the vacuum region.

In addition, editorial changes were made to Surveillance Requirement (SR) 3.4.11.1.b and to Figures 3.4.11-1(a) through 3.4.11-1(c). For SR 3.4.11.1.b, the required heatup and cooldown rate limits will be determined by consulting the P/T limit figures, a format identical to SR 3.4.11-1(a). For the figures, clarifications and updates were made to the titles, labeling, and notes.

In summary, the proposed amendment updates the TS 3.4.11 figures using an NRC approved methodology to adjust the P/T limit curves for the previously missing beltline WLIN data, addresses the RCS vacuum condition that can occur under certain conditions, and aligns the heatup/cooldown requirements of SR 3.4.11.1.b. with the limits in the associated P/T figures.

The proposed TS changes are marked in Attachment 1; the TS retyped pages incorporating the proposed changes are provided in Attachment 2. For the TS Bases, the planned changes are marked in Attachment 3. A summary of the calculation to develop the P/T limit curves is provided as Attachment 4.

2.0 DETAILED DESCRIPTION

The proposed amendment to TS 3.4.11, "RCS Pressure and Temperature (P/T) Limits," will provide three updated figures, with specific updates as noted:

Figure 3.4-11-1(a): Pressure Test Curve (Curve A) (Valid up to 22 EFPY - Unit 1)

This 22 EFPY figure will be deleted.

Figure 3.4.11-1(b): Non-Nuclear Heatup/Cooldown (Curve B) (Valid up to 22 EFPY - Unit 1)

This 22 EFPY figure will be deleted.

Figure 3.4.11-1(c): Core Critical Operation (Curve C) (Valid up to 22 EFPY – Unit 1)

This 22 EFPY figure will be deleted.

Figure 3.4.11-1(d): Pressure Test Curve (Curve A) (Valid up to 32 EFPY – Unit 1)

This figure will be updated and retitled as Figure 3.4.11-1(a): Pressure Test Curves (Valid up to 32 EFPY). The words "Curve A" and "Unit 1" were dropped from the title; Curve A was a carryover from when one P/T limit figure had multiple curves, and Unit 1 is unnecessary as PNPP is a single unit site. The word "Curve" was pluralized as the figure depicts two curves. The word "Limit" was dropped from the Y axis label; this axis depicts pressure in the reactor vessel top head and the curves depict the actual pressure limits. The word "Minimum" was dropped from the X axis label; this axis depicts reactor vessel metal temperature and the curves depict the actual minimum required temperatures. The word "Maximum" was added to the figure's note box describing heatup/cooldown rate; this note intended the heatup/cooldown rate to be an "up to a maximum" rate rather than a single specific rate as currently depicted. The clarification words "Minimum metal temperature at given top head pressure" were added to the curve legend box as a result of the editorial changes made to the X and Y axis labels.

With the update described in the technical evaluation below, portions of the upper vessel and beltline P/T limit curve will generally shift downward and to the right, and will extend into the vacuum region with a new endpoint (70 F, -14.7 psig). Both P/T limit curves will now include temperature limits at their existing upper (1400 psig) endpoints.

Figure 3.4.11-1(e): Non-Nuclear Heatup/Cooldown (Curve B) (Valid up to 32 EFPY - Unit 1)

This figure will be updated and retitled as Figure 3.4.11-1(b): Non-Nuclear Heatup/Cooldown Curves (Valid up to 32 EFPY). The words "Curve B" and "Unit 1" were dropped from the title; Curve B was a carryover from when one P/T limit figure had multiple curves, and Unit 1 is unnecessary as PNPP is a single unit site. The word "Curves" was added to the title as the figure depicts two curves. The word "Limit" was dropped from the Y axis label; this axis depicts pressure in the reactor vessel top head and the curves depict the actual pressure limits. The word "Minimum" was dropped from the X axis label; this axis depicts reactor vessel metal temperature and the curves depict the actual minimum required temperatures. The word "Maximum" was added to the figure's note box describing heatup/cooldown rate; this note intended the heatup/cooldown rate to be an "up to a maximum" rate rather than a single specific rate as currently depicted. The clarification words "Minimum metal temperature at given top head pressure" were added to the curve legend box as a result of the editorial changes made to the X and Y axis labels.

With the update described in the technical evaluation below, portions of the upper vessel and beltline P/T limit curve will generally shift downward and to the right, and will extend into the vacuum region with a new endpoint (70 F, -14.7 psig). Both P/T limit curves will now include temperature limits at their existing upper (1400 psig) endpoints.

Figure 3.4.11-1(f): Core Critical Operation (Curve C) (Valid up to 32 EFPY – Unit 1)

This figure will be updated and retitled as Figure 3.4.11-1(c): Core Critical Operation Curves (Valid up to 32 EFPY). The words "Curve C" and "Unit 1" were dropped from the title; Curve C was a carryover from when one P/T limit figure had multiple curves, and Unit 1 is unnecessary as PNPP is a single unit site. The word "Curves" was added to the title as the figure depicts two curves. The word "Limit" was dropped from the Y axis label; this axis depicts pressure in the reactor vessel top head and the curves depict the actual pressure limits. The word "Minimum" was dropped from the X axis label; this axis depicts reactor vessel metal temperature and the curves depict the actual minimum required temperatures. The word "Maximum" was added to the figure's note box describing heatup/cooldown rate; this note intended the heatup/cooldown rate to be an "up to a maximum" rate rather than a single specific rate as currently depicted. The clarification words "Minimum metal temperature at given top head pressure" were added to the curve legend box as a result of the editorial changes made to the X and Y axis labels.

With the update described in the technical evaluation below, portions of the upper vessel and beltline P/T limit curve will generally shift downward and to the right, and will extend into the vacuum region with a new endpoint (70 F, -14.7 psig). Both P/T limit curves will now include temperature limits at their existing upper (1400 psig) endpoints.

Surveillance Requirement (SR) 3.4.11.1.b, Heatup/Cooldown Limits

The wording of this SR will be revised to delete the specific value of ≤ 100°F currently stated in the SR, and instead will require verification that both the 20°F and 100°F heatup/cooldown rate limits in new Figures 3.4.11-1(a), (b), and (c) are met. This eliminates a human performance error trap from this specification. The SR is revised to read "Verify: RCS heatup and cooldown rates are within the limits of Figure 3.4.11-1."

3.0 TECHNICAL EVALUATION

4.1 Basis for P/T Limits

All components of the RCS are designed to withstand the effects of loads due to system pressure and temperature changes. These loads are introduced by normal and anticipated operational occurrences (for example, startup, heatup, shutdown, cooldown, and reactor trips). The limiting condition of operation of TS 3.4.11 limits the pressure and temperature changes within design assumptions and stress limits for operation during RCS heatup and cooldown by providing specific P/T limit curves.

The TS 3.4.11 figures contain P/T limit curves that are applicable to nuclear and non-nuclear heatup, cooldown, and inservice leak and hydrostatic testing. The provided PNPP figures are valid for 32 EFPY, since the 22 EFPY figures are expected to expire approximately June 2015 (consistent with the requested approval date for this license amendment).

Each P/T limit curve defines an acceptable region for operation. The figures provide guidance during vessel heatup and cooldown. The surveillance requirements, as implemented per plant procedures, provide direction on when pressure and temperature indications are monitored.

The primary purpose of the operating limits is to provide margin to nonductile, brittle failure of the reactor pressure vessel (RPV). The fracture toughness of the RPV materials may decrease over time in the presence of neutron radiation. The curves provide guidance for operators to maintain the margins of safety above brittle fracture limits for affected RPV materials.

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, brittle failure of the RPV, a condition that is unanalyzed. 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.

3.2 Background and Acceptability – Water Level Instrument Nozzle

During a 2009 NRC review of the General Electric Hitachi (GEH) P/T limits licensing topical report (LTR), the NRC issued GEH a request for additional information regarding the adequacy of the methodology used to evaluate the WLINs for the P/T curves, both beltline and non-beltline. To support its response to the NRC, GEH performed a detailed finite element analysis (FEA) of a representative and bounding J-weld penetration WLIN. The NRC reviewed the FEA and subsequently issued a final safety evaluation (SE) [Reference 1] endorsing use of the LTR. The NRC then requested that GEH perform an assessment of the WLIN impact on the P/T limit curves provided to its customers.

The GEH assessment determined that, for PNPP:

- Non-beltline WLINs, as analyzed, are bounded and require no adjustment.
- The PNPP reactor pressure vessel has a J-weld penetration WLIN in the beltline region, which was not included in the last PNPP P/T curve report.
- The pressure test curve requirements for 22 and 32 EFPY for the WLIN are not entirely bounded by the upper vessel curves, and operation must be restricted to use of the beltline or composite curves unless the P/T curves are revised.
- The non-nuclear heatup/cooldown curve requirements for 22 and 32 EFPY for the WLIN are not entirely bounded by the upper vessel curves, and operation must be restricted to use of the beltline or composite curves unless the P/T curves are revised. In addition, there is one area of concern that is not bounded by the beltline or composite curves for the 32 EFPY curve only. Specifically, between the pressures of 290 and 312.5 pounds per square inch gauge (psig), the temperature maximum is up to 5.2°F lower than required.
- The core critical operation curve requirements for 22 EFPY for the WLIN are not entirely bounded by the upper vessel curves, and operation must be restricted to use of the beltline or composite curves unless the P/T curves are revised. In addition, there is one area of concern that is not bounded by the beltline or composite. Specifically, between the pressures of 180 and 312.5 psig, the temperature maximum is up to 19°F lower than required. However, during core critical operation the vessel is at saturation temperature, well to the right of the minimum required curve.
- The core critical operation curve requirements for 32 EFPY for the WLIN are not entirely bounded by the upper vessel curves, and operation must be restricted to use of the beltline or composite curves unless the P/T curves are revised. In addition, there is one area of concern that is not bounded by the beltline or composite. Specifically, between the pressures of 180 and 312.5 psig, the temperature maximum is up to 24.3°F lower than required. However, during core critical operation the vessel is at saturation temperature, well to the right of the minimum required curve.

Upon receipt of the assessment results, FENOC entered the issue into the corrective action program for evaluation. The evaluation determined that P/T limit curve changes were necessary and required administrative control as a nonconservative technical specification.

In 2013, the boiling water reactor owners group (BWROG) issued "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations." This LTR, which is the culmination of BWROG work that began in 2010, included PNPP-specific geometry, loads, and thermal transients in its analyses. In its SE [Reference 2] for the LTR, the NRC stated that no conditions or limitations are necessary for future applicants to address in their application of this LTR to their plant-specific submittals. Additionally, the LTR provides an acceptable methodology for BWR licensees to obtain plant-specific stress intensity factors for use in developing plant-specific P/T limit curves for WLINs.

In this proposed amendment, FENOC used the LTR and supporting FEA to develop the updated P/T limit curves provided herein. The FENOC calculation that applied the LTR's supporting FEA results assumed the current fluence and material adjusted fracture toughness properties for the applicable limiting beltline material. The limiting beltline material for the PNPP reactor pressure vessel, which considered plates and welds affected by the WLIN, was determined to be plate MK 22-1-1 for shell 2, heat number C2557-1. The 32 EFPY adjusted reference temperature (ART) for this material is 59°F.

The FEA, which was performed by the Boiling Water Reactor Vessel Internal Program (BWRVIP) contractor utilizing ANSYS software, determined a unit (1000 psig internal pressure) pressure stress intensity factor of 69.4 ksi√in[1] and a maximum thermal stress intensity factor of 38.6 ksi√in (100°F/hour thermal transient). The K1c methodology, which incorporated the ART and the stress intensity factors from the FEA, resulted in the new upper vessel and beltline curves provided herein. The same K1c methodology was used to develop the existing P/T limit curves. Therefore, no change in methodology.

3.3 Background and Acceptability – Reactor Coolant System Vacuum

During an October 2011 plant startup, a concern was raised regarding the acceptability of a vacuum condition in the RCS with a TS 3.4.11 non-nuclear heatup P/T limit curve that does not indicate or address RCS pressure values below zero (0) pounds per square inch gage (psig). This raised a concern that the RCS was operated outside the governing bounds of the P/T limits, potentially rendering the RCS inoperable.

An investigation determined that during plant startups with the main steam lines and drains open to the main condenser, which is a normal system line-up during startups, a vacuum condition in the RCS is possible. This line-up is specifically used to evacuate air and condensate from the lines leading to the main condenser, and the vacuum condition assists in the air and condensate removal. The investigation also determined that when the RCS is evaluated against the main condenser, the structural adequacy and capability of the main condenser was most limiting with no adverse effect to the RCS.

Subsequently, an engineering calculation to assess the effects of a vacuum in the RCS determined the maximum allowable external pressure on the reactor pressure vessel (RPV). The calculation established the maximum permitted external pressure using applicable rules from the American Society of Mechanical Engineers (ASME) Code. In the calculation, maximum external pressures were obtained for both the reactor shell and spherical head with no stiffeners credited. The calculation results indicate a limiting external pressure on the reactor shell of 475 pounds per square inch (psig) and 694 psig on the spherical head. With the limiting external pressure of 475 psig, the calculation concluded that the RPV would not be adversely affected under the vacuum condition with an external pressure of 0 psig and internal pressure of -14.7 psig. The calculation also concluded there were no vacuum or RCS operational concerns or limitations, that a nonductile brittle failure of the reactor coolant pressure boundary was not credible, and that a vacuum in the RCS was acceptable provided an RCS temperature of 70°F or greater is maintained. To inform and assist operators in the control room, notes were added into applicable operating procedures regarding a vacuum in the RPV under certain conditions.

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[1] ksi√in = kips per square inch square root inches

The proposed amendment will update the P/T limit figures and extend the P/T limits below 0 psig and into the vacuum region. The update of the figures will also include the limiting temperature during vacuum conditions. Therefore, the proposed amendment will account for vacuum conditions in the RCS that can occur during certain periods of plant operation, such as startup or cooldown operations.

3.4 <u>10 CFR 50, Appendix G - Fracture Toughness Requirements</u>

As part of its introduction and scope, Appendix G states:

> This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The scope of the proposed amendment is limited to updating P/T limits due to issues involving the WLIN and vacuum conditions in the RCS. For the proposed amendment, compliance with the NRC approved methodology for developing P/T limits, including Appendix G, were maintained.

4.0 REGULATORY EVALUATION

4.1 <u>Significant Hazards Consideration</u>

The proposed amendment would modify Technical Specification (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," by updating the pressure and temperature limit requirements. Changes to the figures include extending the P/T limit curves below zero (0) pounds per square inch gauge to account for vacuum conditions in the reactor coolant system (RCS), and adjusting the upper vessel and beltline P/T limits due to an updated analysis related to the water level instrument nozzles. Surveillance Requirement (SR) 3.4.11.1.b. is also revised to align the heatup/cooldown requirements of the SR with the limits in the associated P/T figures.

The proposed amendment extends the operating range of the RCS to account for vacuum conditions in the RCS. The proposed amendment does not involve a design modification or physical change to the plant, and does not change methods of plant operation when using the P/T limit curves or maintenance of equipment important to safety.

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

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

Response: No

The P/T limits define RCS operational limits to avoid encountering pressure, temperature, and temperature rate of change conditions that reduce safety margins with respect to nonductile brittle failure of the reactor coolant pressure boundary (RCPB). The figures are not accident initiators or accident mitigating features, but preclude operation in an unanalyzed condition.

This proposed amendment does not change the design function of the RCS or RCPB and does not change the way the plant is maintained or operated when using the P/T limit curves. This proposed amendment does not affect any plant systems that are accident initiators and does not affect any accident mitigating feature.

The proposed amendment does not affect the operability requirements for the RCS, as verification of operating within the P/T limits will continue to be performed, as required. Compliance with and continued verification of the P/T limits support the capability of the RCS to perform its required design functions, consistent with the plant safety analyses.

Changing the figures will not change any of the dose analyses associated with the USAR Chapter 15 accidents because they do not affect the source term, containment isolation or radiological release assumptions used in any accident previously evaluated. Plant accident mitigation functions and requirements remain unchanged.

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

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

Response: No

The P/T limits define RCS operational parameters to protect the RCPB and are not accident initiators or accident mitigating features. The limits are conservatively calculated using an NRC approved methodology. This proposed amendment does not change the design function of the RCS or RCPB, and does not change the way the plant is operated or maintained. This proposed amendment does not affect any plant systems that are accident initiators, does not affect any accident mitigating feature, and does not create a new or different kind of accident.

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

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

Response: No.

The P/T limits define RCS operational parameters, which are established to protect the reactor vessel. The analysis supporting the curve changes utilize methods previously reviewed and approved by the NRC.

Margin of safety is related to the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. This proposed amendment does not directly involve or physically affect fuel cladding or the primary containment. The amendment request proposes to update the P/T limit figures using an NRC approved methodology. The curves maintain the margin of safety for RCPB materials that are exposed to neutron radiation.

The proposed amendment does not involve a physical change to the plant, does not change methods of plant operation within prescribed limits, and does not change methods of maintenance on equipment important to safety.

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

Based on the responses to the three questions above, FENOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements/Criteria

The proposed amendment has been reviewed against General Design Criteria (GDC), the Standard Review Plan (SRP), branch technical position (BTP) documents, and 10 CFR 50 Appendix G to determine whether applicable regulations and requirements would continue to be met. Specifically:

- 10 CFR 50, Appendix G, Fracture Toughness Requirements
- GDC 14, Reactor Coolant Pressure Boundary
- GDC 15, Reactor Coolant System Design
- GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary
- SRP 5.2.3, Reactor Coolant Pressure Boundary Materials
- SRP 5.3.1, Reactor Vessel Materials
- SRP 5.3.2, Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock
- SRP 5.3.3, Reactor Vessel Integrity
- SRP 5.4, Reactor Coolant System Component and Subsystem Design
- BTP 5-3, Fracture Toughness Requirements

FENOC has determined that the proposed amendment maintains conformance with the criteria and requirements described in the above cited documents and in PNPP's Updated Safety Analysis Report (USAR).

4.3 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

A review 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, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. U.S. Nuclear Regulatory Commission letter to Mr. Doug Coleman, BWROG Chair, Subject: 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.
[Accession Nos. ML091100139, ML091100117 and ML091110060]

2. U.S. Nuclear Regulatory Commission letter to Mr. Frederick Schiffley, BWROG Chairman, Subject: Final Safety Evaluation for Boiling Water Reactors Owners' Group Topical Report BWROG-TP-11-023, Revision 0, November 2011, "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," (TAC No. ME7650), March 14, 2013.
[Accession No. ML13056A157]

Attachment 1

Proposed Technical Specification Changes

(MARK-UP)

(7 Pages Follow)

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 3.4.11.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
	Verify: a. RCS pressure and RCS temperature are within the limits of Figure 3.4.11-1; and b. RCS heatup and cooldown rates are ~~≤ 100°F in any one hour period.~~	30 minutes

(continued)

INSERT: within the limits of Figure 3.4.11-1.

REMOVE this [22 EFPY] FIGURE.

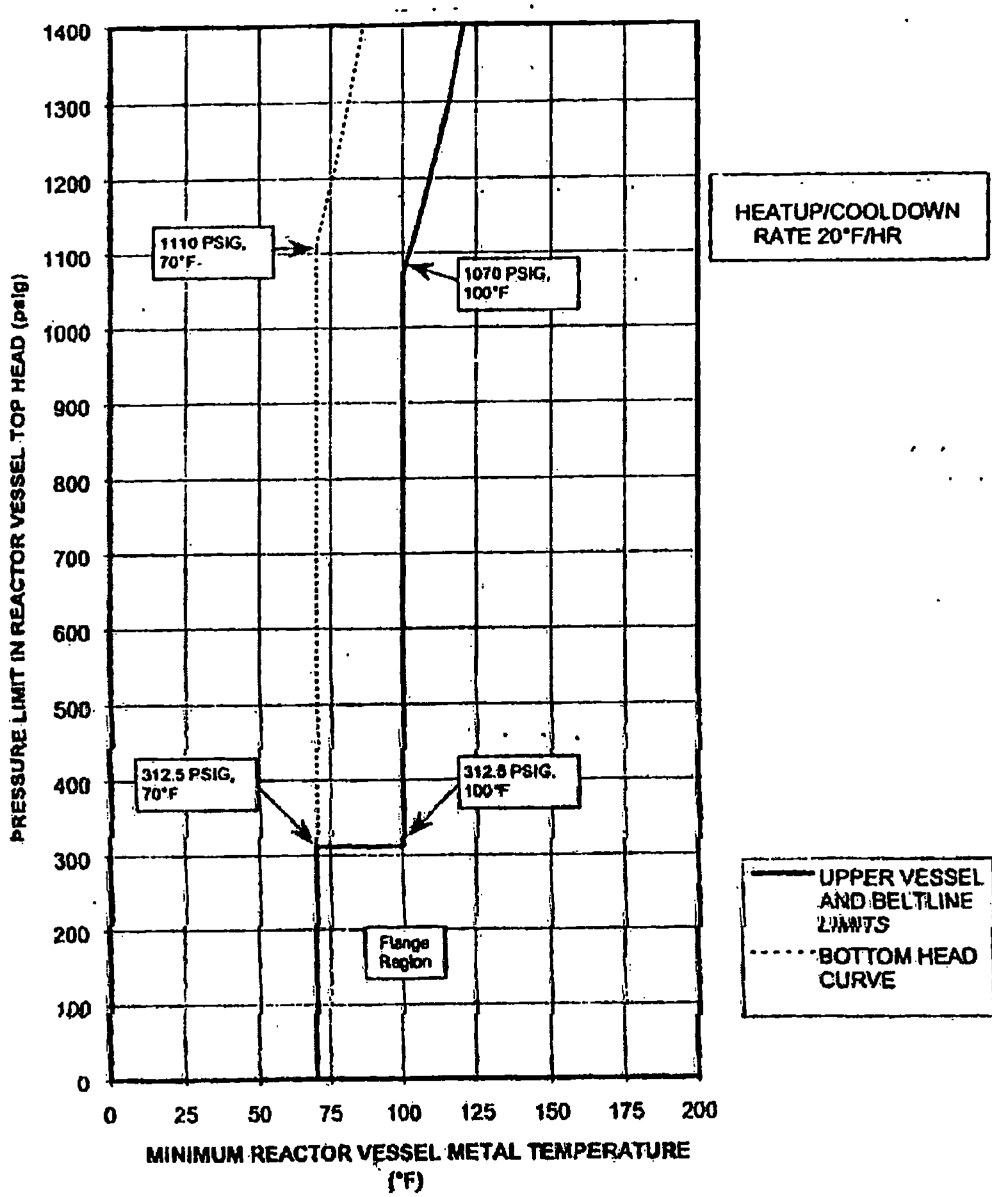


Figure 3.4.11-1(a): *Pressure Test Curve (Curve A) (Valid Up to 22 EFPY – Unit 1)*

REMOVE this [22 EFPY] FIGURE.

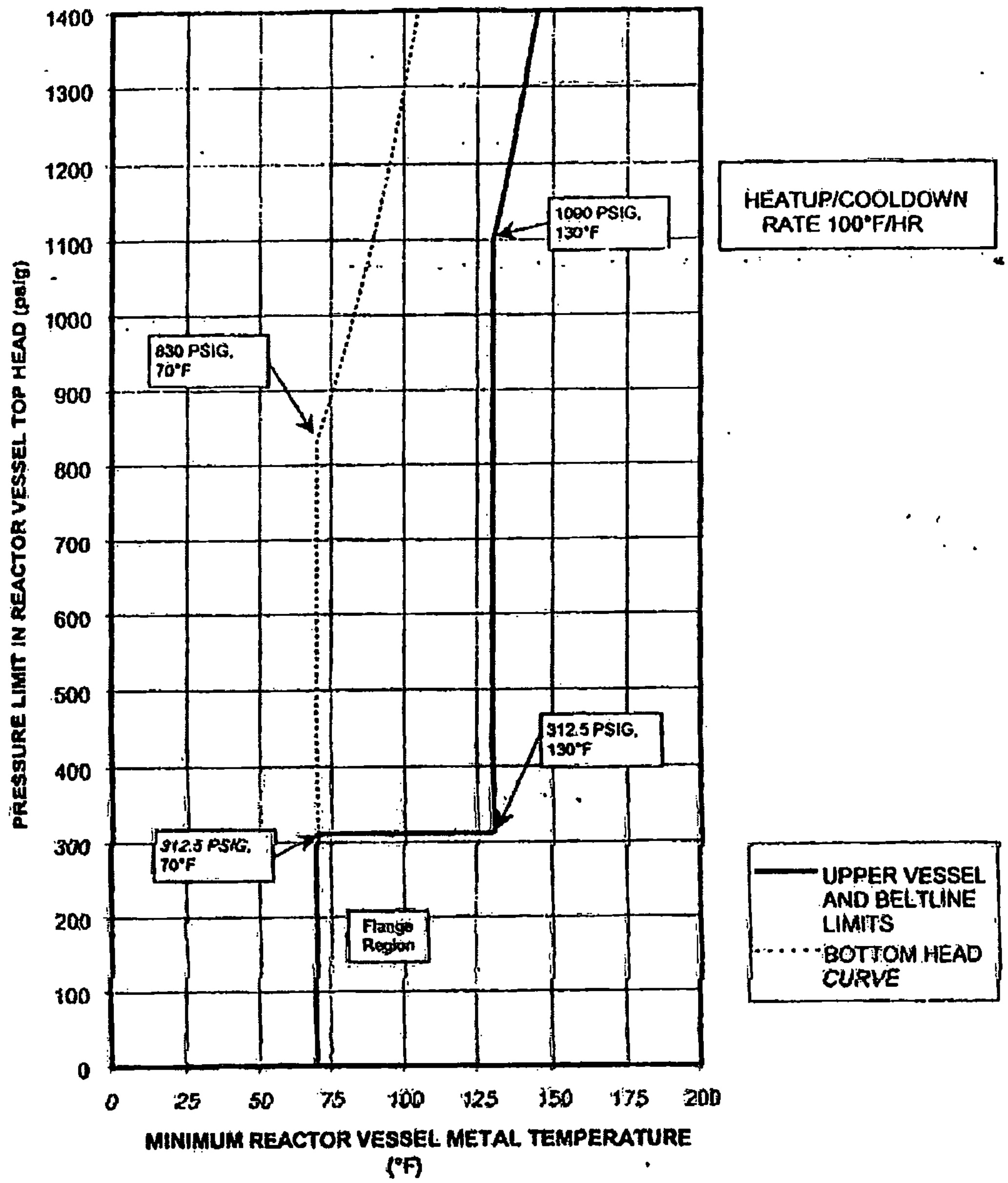


Figure 3.4.11-1(b): Non-Nuclear Heatup/Cooldown (Curve B) (Valid Up to 22 EFPY – Unit 1)

REMOVE this [22 EFPY] FIGURE.

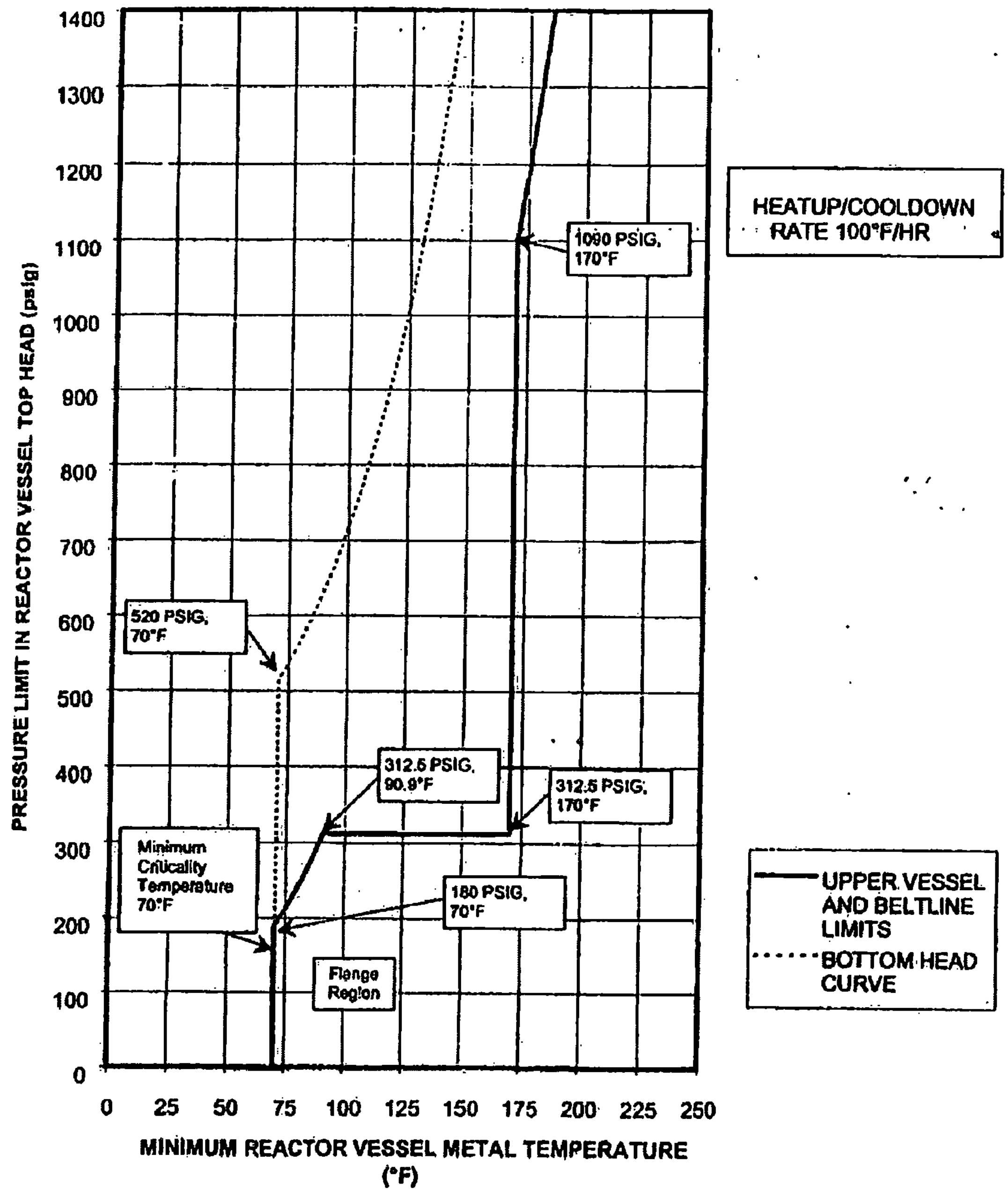


Figure 3.4.11-1(c): Core Critical Operation (Curve C) (Valid Up to 22 EFPY – Unit 1)

REMOVE and REPLACE this FIGURE with the RENAMED and UPDATED FIGURE.

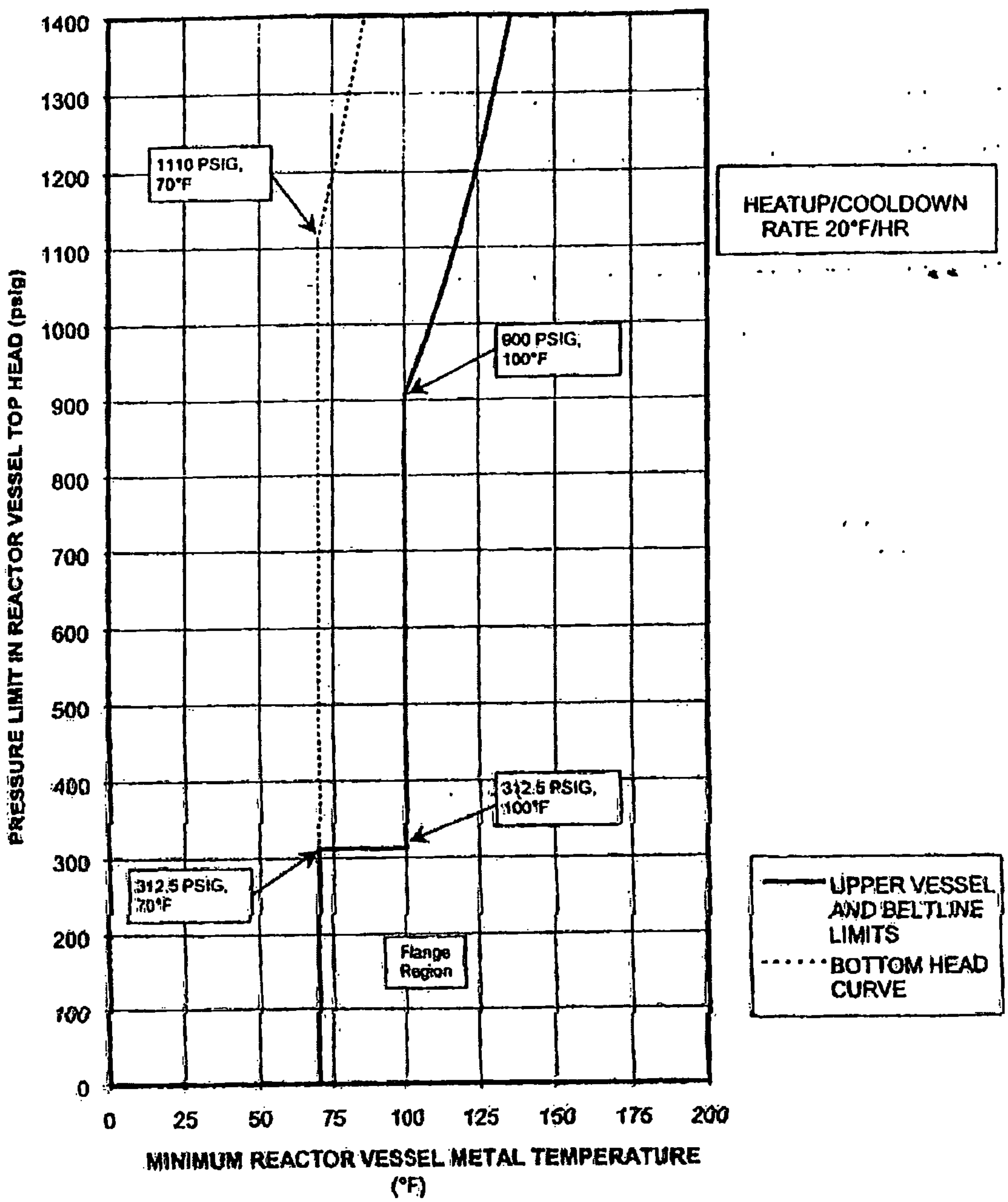


Figure 3.4.11-1(d): Pressure Test Curve (Curve A) (Valid Up to 32 EFPY – Unit 1)

REMOVE and REPLACE this FIGURE with the RENAMED and UPDATED FIGURE.

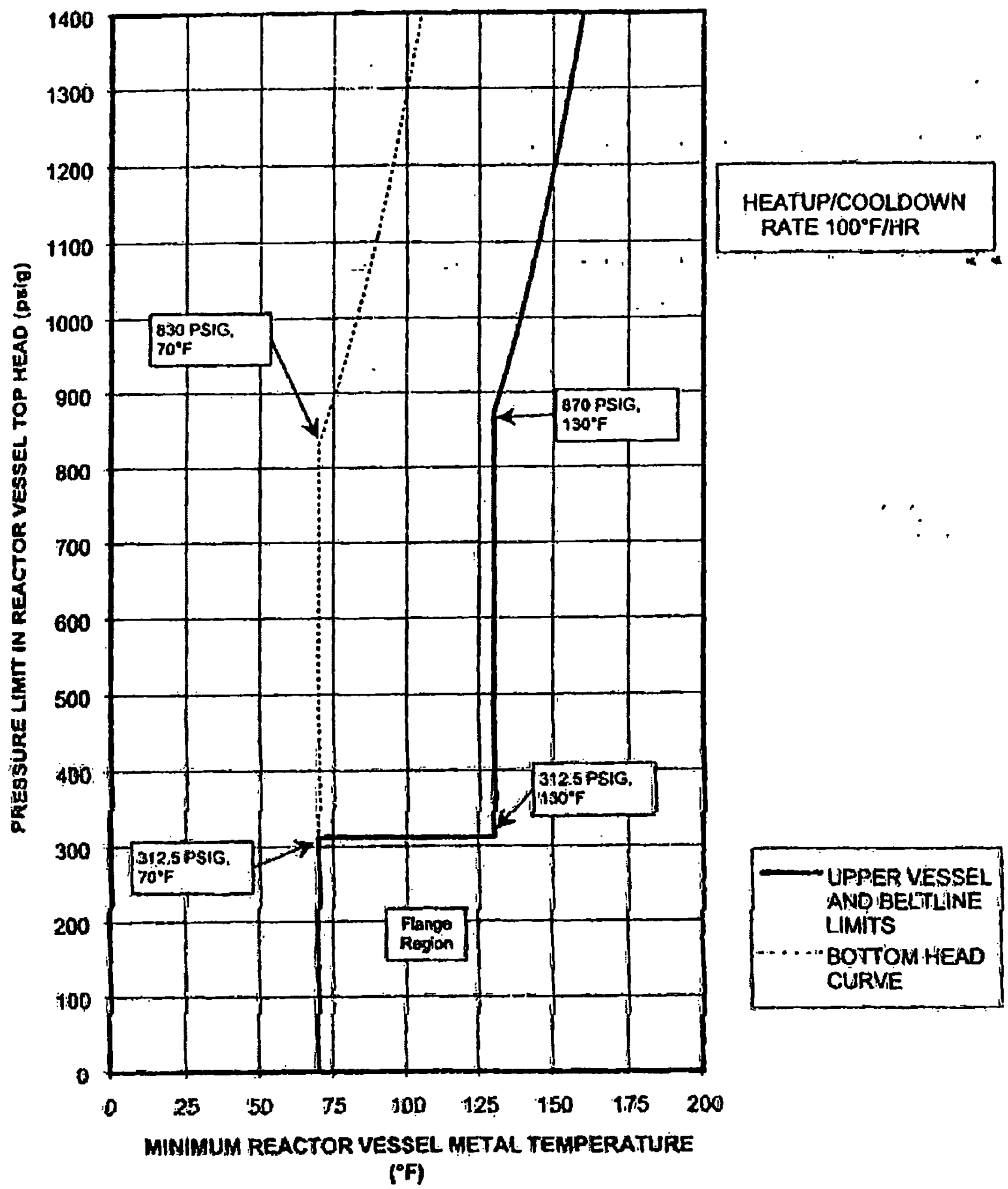


Figure 3.4.11-1(e): Non-Nuclear Heatup/Cooldown (Curve B) (Valid Up to 32 EFPY – Unit 1)

REMOVE and REPLACE this FIGURE with the RENAMED and UPDATED FIGURE.

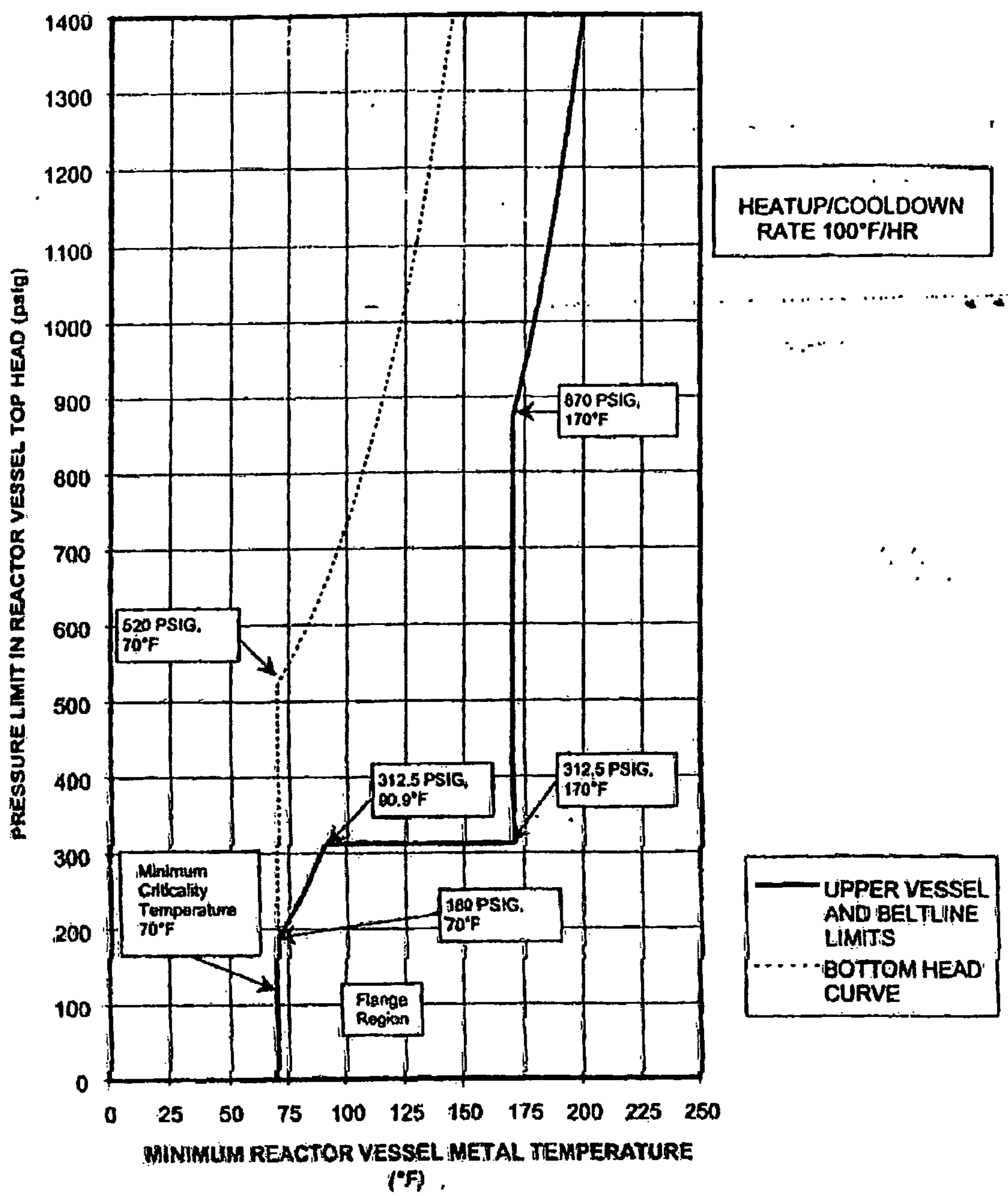


Figure 3.4.11-1(f): Core Critical Operation (Curve C) (Valid Up to 32 EFPY – Unit 1)

Attachment 2

Proposed Technical Specification Changes

(RETYPED)

(4 Pages Follow)

RETYPED, FOR INFORMATION ONLY

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. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify: a. RCS pressure and RCS temperature are within the limits of Figure 3.4.11-1; and b. RCS heatup and cooldown rates are within the limits of Figure 3.4.11-1.	30 minutes

(continued)

RETYPED, FOR INFORMATION ONLY

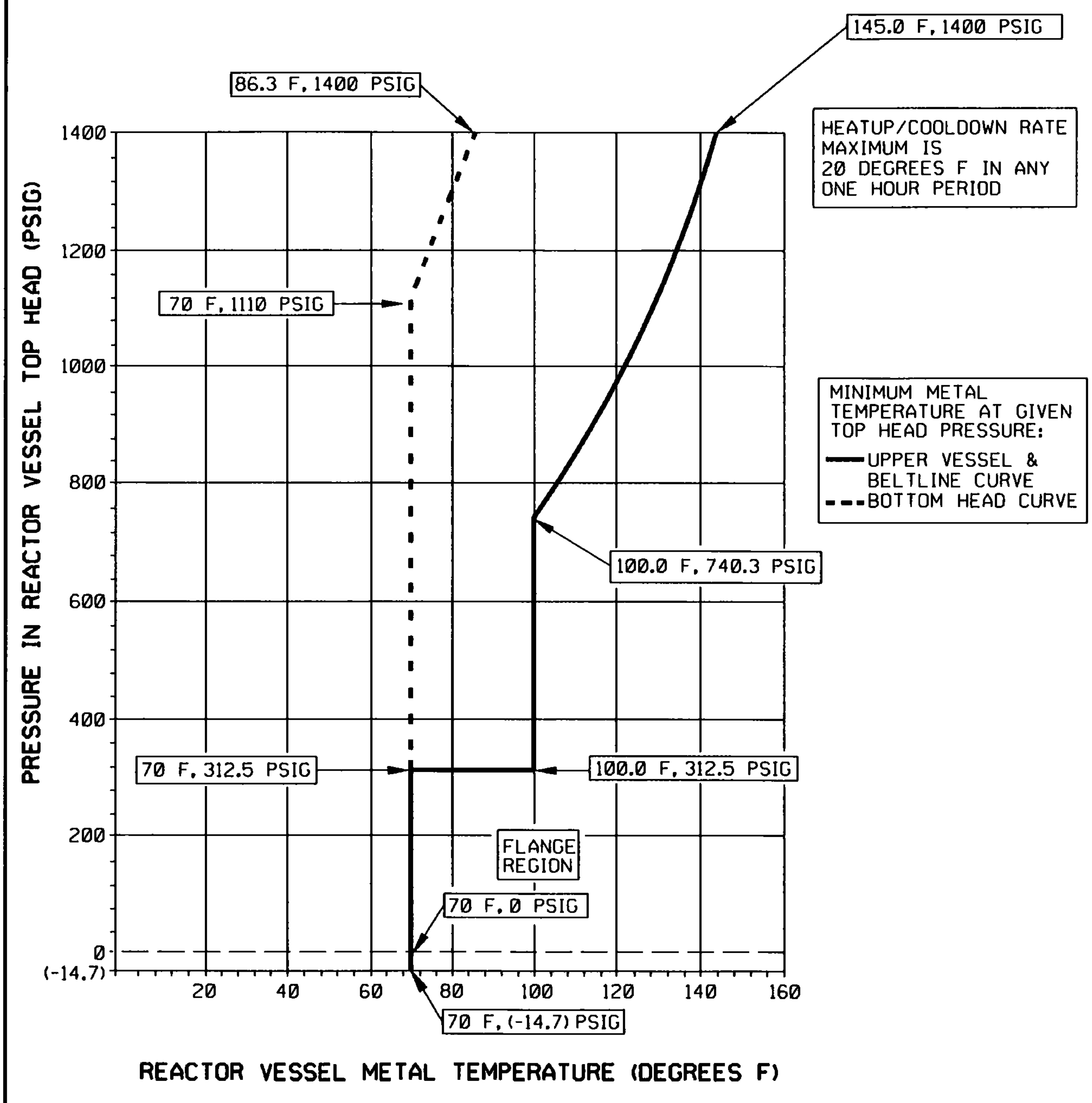


FIGURE 3.4.11-1(a): PRESSURE TEST CURVES (VALID UP TO 32 EFPY)

RETYPED, FOR INFORMATION ONLY

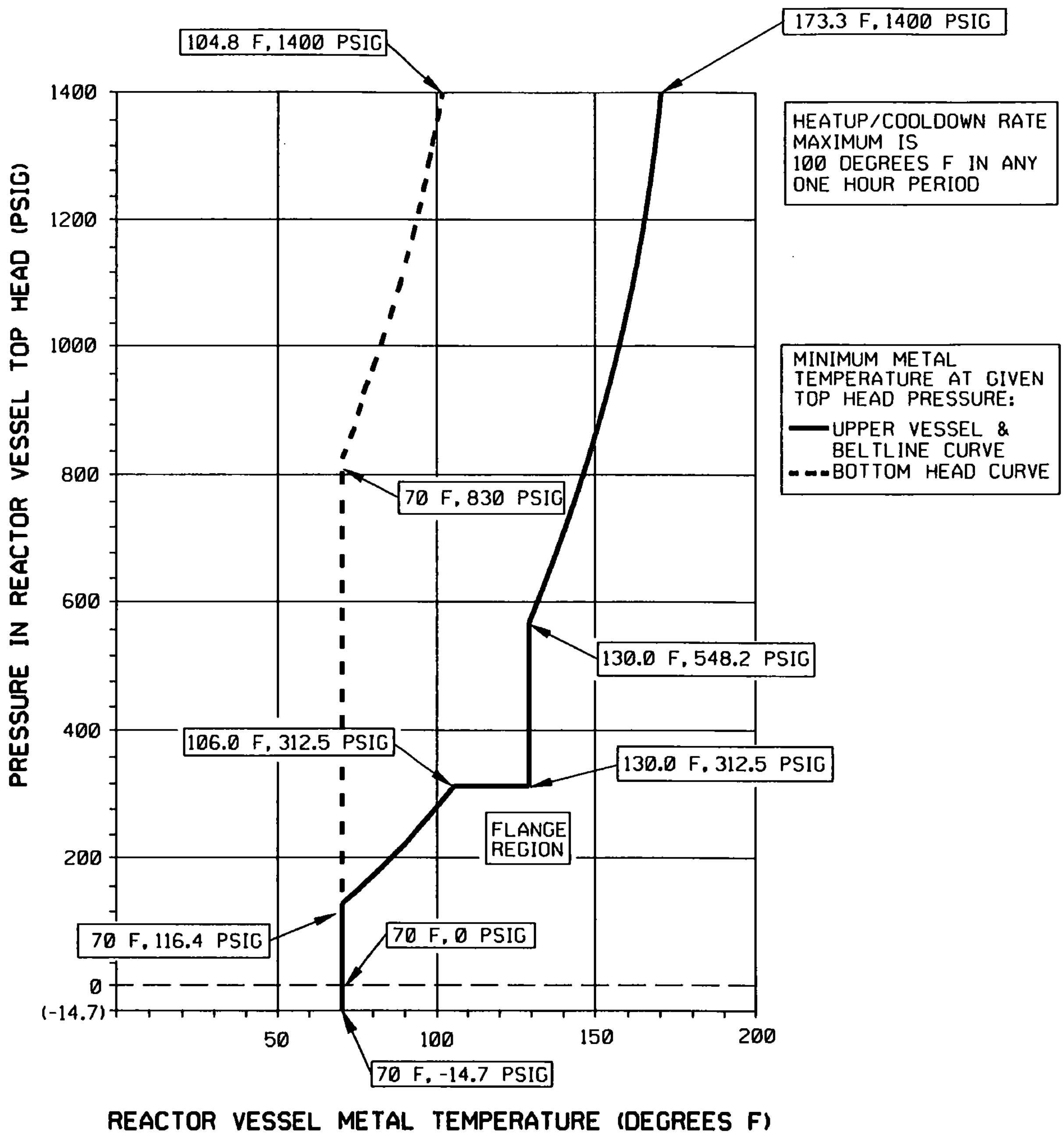


FIGURE 3.4.11-1(b): NON-NUCLEAR HEATUP/COOLDOWN CURVES (VALID UP TO 32 EFPY)

RETYPED, FOR INFORMATION ONLY

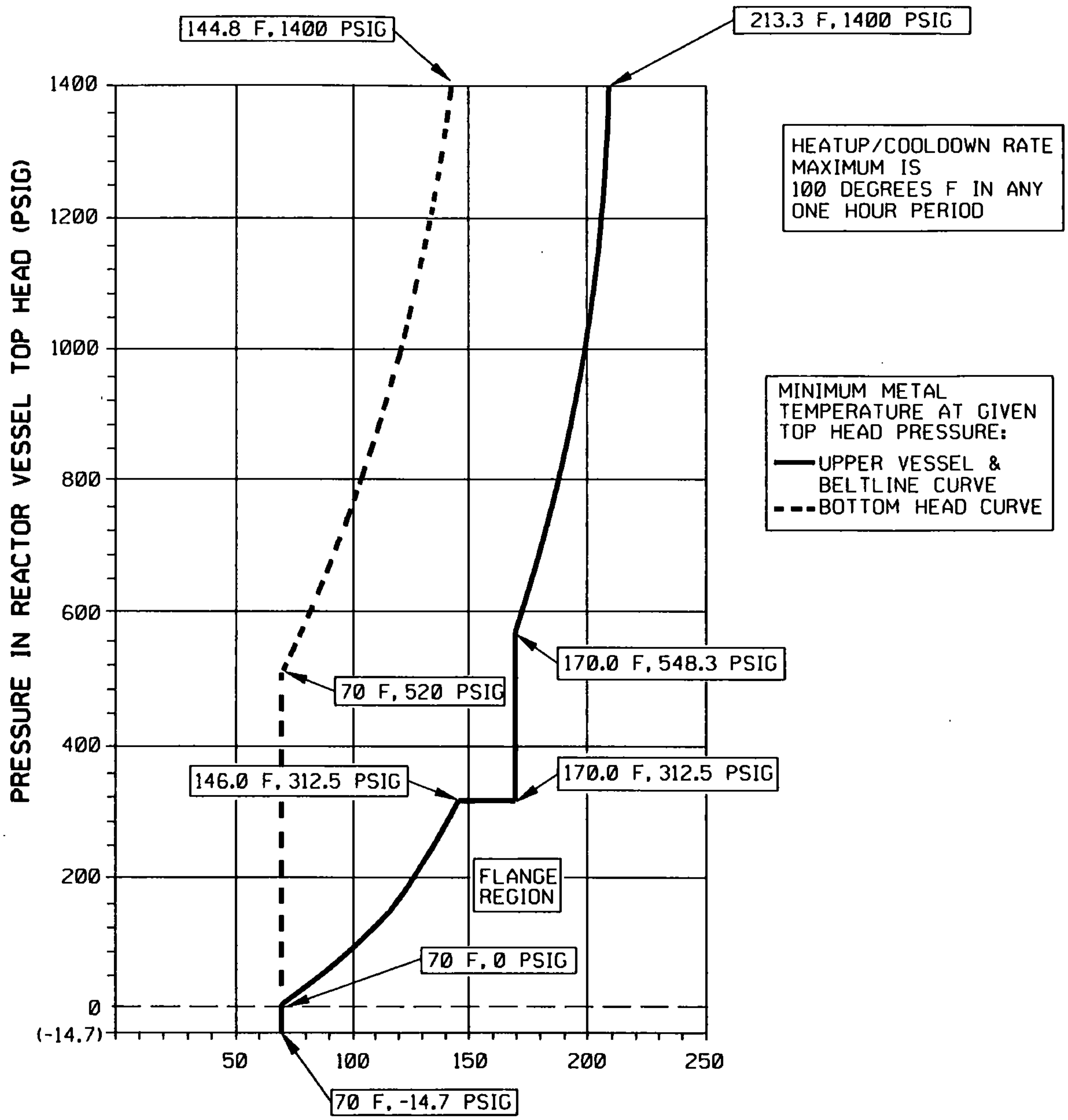


FIGURE 3.4.11-1(c): CORE CRITICAL OPERATION CURVES (VALID UP TO 32 EFPY)

Attachment 3

Proposed Technical Specification Bases Changes

(PROVIDED FOR INFORMATION)

(4 Pages Follow)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

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

DELETE reference to 22 EFPY

Figure 3.4.11-1 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. The heatup curve provides limits for both heatup and criticality. Curves are provided which are valid for up to ~~22 EFPY and~~ 32 EFPY.

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

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

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

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185-82 (Ref. 3) and 10 CFR 50, Appendix H

(continued)

BASES

BACKGROUND (continued)

(Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

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 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 the minimum permissible temperature for the inservice leak and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, 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. The 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 RCPB, a condition that is unanalyzed. ~~Reference 7 establishes~~ the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance

(continued)

INSERT:
References 7 and 11 establish

APPLICABLE SAFETY ANALYSES (continued)

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

RCS P/T limits satisfy Criterion 2 of the NRC Final Policy Statement on Technical Specification Improvements (58 FR 39132).

LCO

This LCO ensures that limits are met on RCS pressure, temperature, and heatup and cooldown rates. Elements of this LCO are:

a. RCS pressure, temperature, and heatup or cooldown rate are within limits during RCS heatup, cooldown, and inservice leak and hydrostatic testing.

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is within limits during each recirculation pump startup, and also during increases in THERMAL POWER or loop flow while in single loop operation at low THERMAL POWER or loop flow.

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel meets the limit during each recirculation pump startup, and also during increases in THERMAL POWER or loop flow while in single loop operation at low THERMAL POWER or loop flow.

d. RCS pressure and temperature are within the criticality limits prior to control rod withdrawal for the purpose of achieving criticality.

e. The reactor vessel flange and the head flange temperatures are within limits when tensioning the reactor vessel head bolting studs.

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

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak 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)

PROVIDED FOR INFORMATION ONLY

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.10 (continued)

material specimens, in accordance with ASTM E 185-82 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves in Figure 3.4.11-1 will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. GE-NE-0000-0000-8763-01, Revision 0, "Pressure-Temperature Curves For FirstEnergy Corporation, Using the K_{Ic} Methodology, Perry Unit 1," April 2002.
8. USAR, Section 15.4.4, "Abnormal Startup of Idle Recirculation Loop."
9. GE Services Information Letter, SIL No. 517 Supplement 1, "Analysis Basis for Idle Recirculation Loop Startup."
10. USAR, Sections 5.3.1.5, "Fracture Toughness," and 5.3.2, "Pressure-Temperature Limits."

INSERT:
11. Boiling Water Reactor Owners' Group Topical Report BWROG-TP-11-023-A, Revision 0, May 2013, "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations."

Attachment 4

Development of P/T Limit Curves for Technical Specification 3.4.11

(27 Pages Follow)

This attachment is considered a summary of the associated calculation; only selected pages of the approved document are provided.

Title/Subject: Development of P/T Limit Curves for Technical Specification 3.4.11

TABLE OF CONTENTS

[NOTE: Though 22 EFPY attachments or figures may be cited, only Attachments 2, 4 and 6 for 32 EFPY are provided. This aligns with the license amendment request to modify PNPP TS 3.4.11, which only submits updated 32 EFPY P/T Limit figures. Additional notes to the same have been added throughout this document.]

OBJECTIVE OR PURPOSE:

The objective of this calculation is to capture vendor documentation to provide new reactor coolant system pressure and temperature (P/T) curves that include the impact of the water level instrument nozzles (WLIN) on the reactor shell material. The results of this calculation will be used as input to a License Amendment to update Technical Specification (TS) 3.4.11 curves and remove the non-conservative TS.

Background

CR 09-64465 identified that General Electric assessment of the WLIN penetrations impact the reactor shell material which in turn affects the Pressure-Temperature (PT) curves in Technical Specification 3.4.11. Attachment 3 to EA-0246 calculation, GE Report, GENE-0000-0000-8763-01, P/T Curves for First Energy Corporation Using K_{Ic} Methodology Table A-2, "Geometric Discontinuities Not Requiring Fracture Toughness Evaluations" Perry's WLIN, N12, is forged from stainless steel and exempted from fracture toughness evaluation. GE letter (DIN 14) states that the NRC requested GE provide assessment to demonstrate the impact of the penetration on the PT curves non-beltline and beltline region of the vessel. Based on the input of the letter, a non-conservative Technical Specification was submitted at Perry and is currently in place. The technical basis supporting a permanent change to Technical Specifications was to be completed under corrective action 09-64465-001 which required initiation of this calculation revision. It is recognized that any change to a Technical Specification will require a License Amendment.

On January 31, 1996, the NRC staff issued Generic Letter 96-03 to inform licenses that they may request a license amendment to relocate the P/T curves from the Technical Specifications (TS) into a pressure temperature limits report (PTLR) or other licensee-controlled document that would be controlled through the TS. Thus the industry is working to remove the P/T curves from Technical Specifications and place them in a controlled procedure for future updates. In support of this effort, a Pressure Temperature Limits Report (PTLR) has been prepared by Structural Integrity and Associate (SIA) under contract to the BWR Owner's Group and another has been submitted to the NRC through General Electric Hitachi (GEH). It was during the NRC review of the GEH PTLR that the WLIN issue was identified. The SIA /BWROG submittal, SIR-05-044A, Rev. 0, had received final NRC review prior to the WLIN issue identification. Revision 1 to the SIR (DIN 16) has been prepared by SIA under contract to the BWROG and has been re-submitted to the NRC. The P/T modified curve is constructed following the guidance of this document. Support calculations for Revision 1 to the SIR included; BWROG –TP-11-023A Report 0900876.401 Rev 0, Linear Elastic Fracture Mechanics Evaluation of the General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure –Temperature Curve Evaluations (DIN 17); Report No. 0900876.303 Instrument Nozzle Stress Intensity Factor Calculation for Plant Specific 238-Inch BWR (DIN 15).

Revision 1 of this calculation prepares new formal modifications to the applicable P/T curves based on results from Report No. 0900876.303 Instrument Nozzle Stress Intensity Factor Calculation for Plant Specific 238-Inch BWR (DIN 15). This revision accounts for the WLIN impact to the vessel plate material. The most recent & limiting adjusted reference temperature NDT_{ART}, documented in this calculation will be considered for the affected plate.

Revision 2 adjusts the Upper Vessel and Beltline curves in the Attachments 2, 4 and 6 to include the static reactor head pressure. The bottom head curve remains unchanged and the source of the bottom head data is indicated in the attachments. Revision 2 was prompted by the PTLR author's comments on Revision1.

SCOPE OF CALCULATION/REVISION:

This calculation considers the impact of the WLIN penetration near the reactor core beltline on the Perry P/T curves. The nozzle was previously determined by GE (DIN14) to have no adverse effect on the P/T curves. New curves are presented based on the BWROG methods. A review of plant operating data within the past five years was conducted to determine the impact of the more conservative P/T curves. Lastly, structural justification is provided for a vacuum in the vessel that may exist prior, during, or after a Leak Test, Non-Core Critical, or Core Critical operation.

SUMMARY OF RESULTS/CONCLUSIONS:

The changes to the affected P/T curves have been included within this calculation as well as justification for a reactor vacuum under specific conditions. Follow up activities include:

1. Submittal of a License Amendment to include updated P/T curves based on the Attachment 2, 4 and 6. **ONLY the 32 EFPY curves (P/T Limits) are to be submitted.**

2. Apply more conservative Leak Test temperature (Leak test temperature at rated pressure must be increased to maintain compliance with the 32 EFPY Leak Test Curve (Attachment 2)). This is proposed for next refuel outage 15 in advance of approval because Perry is near the end of the 22 EFPY.

3. Add a Note to TS 3.4.11 to allow a reactor vacuum during Leak Test, Non-Core Critical Operation, and Core Critical Operation. **Alternatively, figures will depict a pressure region below 0 PSIG (vacuum region).**

LIMITATIONS OR RESTRICTIONS ON CALCULATION APPLICABILITY:

NRC acceptance of the BWROG/Structural Integrity PTLR has recently been obtained. The NRC Safety Evaluation may be found in DIN 17, BWROG-TP-11-023-A Rev. 0, "Linear Elastic Fracture Mechanics Evaluation of the General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure –Temperature Curve Evaluations".

This calculation assumes the current fluence and material adjusted fracture toughness properties remain applicable. Specimens removed from the reactor vessel in the Spring of 2013 are currently being analyzed under the guidelines of the BWRVIP-86-A, BWR Vessel and Internal Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan. The results of these analyses and the impact to the beltline materials are beyond the scope of this calculation and will be addressed at a later date.

IMPACT ON OUTPUT DOCUMENTS:

This calculation modifies the existing P/T curves as currently identified in Calculation EA-0246. Calculation EA-0246 remains active as it supports the P/T curves with the exception of the impact of the WLIN. A PIN cross reference to this calculation shall be added to calculation EA-0246 in the interim. Once calculation EA-0246 is revised for new fluence calculations and adjusted material reference temperatures, the curves in this calculation will be obsolete, however the references will remain applicable unless revised. The P/T curve information is contained in TS 3.4.11, SVI-B21T1176, ISI-B21 TI300, USAR, and the plant process computer. Typically, a license amendment request is submitted to the NRC for the revision to the Technical Specifications. Once NRC approval is received, the supporting documents may be revised (SVI's, procedures, USAR, plant process computer).

DOCUMENT INDEX

DIN No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
1	Condition Report 09-64465	N/A	x		
2	G1-06 Power Uprate Off-Rated Heat Balance and Performance Improvement Features, GE-NE-A2200084-06-01, Class 3	Rev. 0, Feb 1999		x	
3	ASME Section XI, Appendix G	2004 edition	x		
4	EPRI BWRVIP-135, (Technical Report 1011019), BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations	Rev 2	x		
5	08_-0622-0165-0 As built Vessel Weld Seams	Rev. 0		x	
6	08_-0195-0000 As-Built Vessel Nozzle Locations	Rev. 1		x	
7	08_-0622-00R13 Weld Seam and X-ray Locations	Rev. 0	x		
8	Perry I Contract 73-C108 CBI Nuclear Co. Appendix C9, "238 inch BWR-6 Vessel Fracture Toughness Calculations for the Code 1563 psig System Hydrostatic Test.	Rev 0 dated 4/75	x		
9	ASME Boiler and Pressure Vessel Code, 1971 Edition	1971 Edition	x		
10	ISI-B21-T1300-1 Reactor Coolant System Leakage Test	Rev. 17	x		
11	302-0608	Rev M		x	
12	302-0609	Rev. B		x	
13	CBI Design Report D1, Sheet 2 of 35	3/19/76		x	
14	GE Letter neFile 0000-0106-1616 Impact Assessment for Water Level Instrument Nozzle (Penetration) on Pressure – Temperature (PT) Curves Provided to BWROG Members. Andrew Meyers, GEH, to Paul Harden FE.	September 11, 2009	x		
15	Structural Integrity Calculation No. 0900876.303, "Instrument Nozzle Stress Intensity Factor Calculation for Plant Specific 238 –Inch BWR" PROPRIETARY	Revision 0 Dated 2/4/11	x		
16	Pressure –Temperature Limits Report Methodology for Boiling Water Reactors, Report No -05-044-A PROPRIETARY	Rev. 1 dated June 2011	x		

17	BWROG –TP-11-023A Report No. 0900876.401 Linear Elastic Fracture Mechanics Evaluation of the General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure –Temperature Curve Evaluations PROPRIETARY	Rev. 0 May 3, 2013	x		
18	Code of Federal Regulations, Appendix G to Part 50	July 19, 2011 (website date of last review)	x		
19	EA-0246 Reactor Vessel RCS Pressure Temperature Curves and Attachments	Rev 0		x	
20	22A6483 GE Design Specification Reactor Vessel Overprotection Report (Reactor design Pressure sheet 5 Reactor design Pressure = 1250 psig)	Rev. 3		x	
21	NEDC-33178P-A, Class III, "Licensing Topical Report GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves"	Rev 1, June 2008	x		
22	PDB-A008, Core Average exposure History	Rev 15	x		
23	305-006-00102 (ISI Seam Weld Layout)	Rev C			
24	Drawing 26_-0002-00004 High Pressure Condenser	Rev 5A		x	
25	Chicago Bridge and Iron ASME Code Data Report Vessel T49 (microfilm reel 13064 frame 106)	Dec. 19,1975		x	
26	2010 ASME Boiler and Pressure Vessel Code, Section III Division 1-Subsection NB Class 1 Components	2010 edition (no addenda)	x		
27	2010 ASME Boiler and Pressure Vessel Code Section II Part D Material Properties	2010 (no addenda)	x		
28	10CFR50.55a Codes and Standards	1/20/14 (Internet copy)	x		
29	IOI-0001 Cold Startup	Rev 38	x		
30	CR-2013-18689 (Corrective Action CA-2013-18689-009).	N/A	x		
31	CBI Nuclear Co., Surface Heat Transfer Coefficient, Book T1 Main Closure Flange Analysis, sheet 36.	Rev 0		x	
32	CBI Nuclear Co., Inside Surface Heat Transfer Coefficient, Book T11.3 sheet 24	Rev. 0		x	
33	CBI Nuclear Co., Water Level Instrument Nozzle Design Report Section D 11.3, sheet 1	Rev. 5		x	
34	Calculation B21-C11 Reactor Vessel Low Level Trip	Rev 3	x		

35	NEDC-32972P BWR Owner's Group Report Safety Analysis Evaluations Relative to Measurement Uncertainties for BWR/6 Improved Standard Technical Specifications.	Rev. 0 February 2001	x		
36	208-0010-00003 Vessel Temperature Monitoring Elementary	Rev T	x		
37	Chicago Bridge and Iron Design Report for Flange and Studs Sketch Sheet 2	Rev 4	x		

Analysis Methodology & Assumptions

P/T Curve and WLIN Impact

First, a review is conducted to establish the technical requirements for revising the affected P/T curves for the impact of the water level instrument nozzle (WLIN). Applicable Code of Federal Regulations and pertinent ASME Code requirements are identified.

Second, the limiting beltline material adjusted reference temperatures and supporting fluence calculations (DIN 19) are assumed applicable and are not changed from the previous P/T Curve submittal for 22 and 32 effective full power years (EFPY)[1].

Third, Structural Integrity Calculation No. 0900876.303, "Instrument Nozzle Stress Intensity Factor Calculation for Plant Specific 238 –Inch BWR" (DIN 15) is the primary input document for the Stress Intensity Factor for the WLIN. This calculation is a finite element analysis, prepared with Perry Inputs as Perry's nozzle and vessel were used as one of the bounding plants in BWROG-TP-11-023A Report No. 0900876.401 Linear Elastic Fracture Mechanics Evaluation of the General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure –Temperature Curve Evaluations (DIN 17) which has been reviewed by the Regulator.

BWROG-TP-11-023A Report No. 0900876.401, was submitted to the Regulator as a Licensing Topical Report (LTR). The LTR provides a linear elastic fracture mechanics (LEFM) evaluation of the WLIN used in P/T limit applications. The report provides a bounding partial penetration style water level instrument nozzle fracture mechanics solution which can be used to obtain plant specific stress intensity factors for an internal pressure load case and a 100 °F/hr thermal ramp load case for use in developing plant specific P/T curves, without having to develop and analyze a plant specific finite element model. Adequacy of this approach is demonstrated by comparing the boundary integral equation / influence function (BIE/IF) solution against finite element fracture mechanics results of comparable configurations and by comparison of the K_I obtained using the proposed equations against plant specific calculations for three benchmark cases. Section 2.2 of the LTR calculation indicates one of the benchmark cases was for a 238 inch diameter vessel as indicated in reference 11 (DIN15). This was actually Perry's WLIN configuration. Therefore, Perry is part of the envelope of plants included in the LTR. The LTR includes response to the requests for information in an attachment and also includes the Regulator's Safety Evaluation. This calculation uses the results of DIN 15 to prepare the new TS curves for the impact of the WLIN.

Reactor Vessel Vacuum

In Section V of the calculation, technical bases are established for the structural aspects of a reactor vacuum. An allowable external pressure is established and the vacuum is shown to be a small fraction of that value. Bases are also established for the minimum temperature allowed under vacuum conditions.

[1] EFPY is an indication of actual energy produced vs theoretical maximum which could have been produced over the same time period. Typically energy produced or fuel burn up is directly proportional to fast neutron flux or fluence in the vessel beltline materials. As used in the P/T curves, EFPY is a convenient term used to assess material damage or reduction in ductility (neutron embrittlement) over time. Embrittlement causes a shift in the affected materials' nil-ductility transition temperature or NDTT. The reference nil-ductility temperature, RT_{NDT}, is the higher of either the NDTT or the temperature, obtained from Charpy tests, at which the material exhibits at least 50 ft-lbs of impact energy and 35 mils of lateral expansion (normal to the working direction), minus 60 degrees. For the vessel, the ASME

Code Section III requires the establishment of RT_{NDT} when the vessel is constructed. Reactor specimen testing over the plant operating life provides information used to adjust the RT_{NDT} to the adjusted reference temperature or ART of affected materials.

I. Development of P/T Curves.

The impact of the WLIN was assessed using the BWROG reports. The approach used for calculating the pressure test (Curve A), core not critical (Curve B), and core critical (Curve C) P/T limit curves considering the WLIN region is taken directly from SIR -05-044, DIN 16, on page 3-1.

a. Evaluate surveillance data in accordance with Appendix A of the report.
b. Assume a coolant temperature, T. The temperature drop from the fluid to the metal temperature at the assumed flaw tip (i.e. T at 1/4t) is conservatively assumed to be zero and metal temperature is assumed equivalent to the coolant temperature.
c. Calculate the allowable stress intensity factor, K_{Ic}, using equation 2.4-2 for the assumed fluid temperature, T, and the limiting ART for the region being evaluated.
d. Calculate the thermal stress intensity factor, K_{It}, using one of the methods described in Sections 2.5.
e. Calculate the allowable pressure stress intensity, K_{Im}, using the methods described in Sections 2.5.2 and 2.5.3.
f. Calculate the allowable pressure, P_{allow}, using the methods described in Sections 2.5.2 or 2.5.3.
g. Repeat steps (b) – (f) for other temperatures to generate a series of P/T points. The resulting pressure and temperature series constitutes the P/T curve. The P/T curve relates the minimum required coolant temperature to the allowable measured reactor pressure.
h. For non-beltline P/T limits apply the additional minimum temperature requirements described in Sections 2.7 and 2.8.
i. Apply any applicable adjustments to the final temperatures and pressures, as described in Section 2.6.

The Perry Technical Specification contains P/T curves for both 22 and 32 EFPY; therefore both sets of curves are to be revised to show the WLIN impact. The applicable material adjusted reference temperature for shell ring 2, is considered for both 22 and 32 EFPY. The new curves are developed for each operating condition, Leak Test, Non-Core Critical, and Core Critical Operation. [NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]

A. Evaluate surveillance data in accordance with Appendix A of the report.

The Pressure –Temperature Limits Report Methodology for Boiling Water Reactors, Report No -05-044-A (DIN 16) discusses nozzles in the beltline region on page 2-21. Since the nozzle exists in the beltline region, the effects on the available fracture toughness must be addressed. The WLIN consists of an insert attached to the RPV with a partial penetration weld. Perry's nozzle material is stainless steel and therefore does not require consideration of fracture toughness, however the effect of the material on the adjacent shell must be considered.

As stated in the assumptions the current surveillance data is used and taken from DIN 19. Comparison of as-built nozzle and weld seam information DINs 5, 6, 7 and 23 indicates that 4 level instrument nozzles are located at the top of active fuel, designated N12, and lie in shell ring # 2. The irradiated material properties and adjusted reference temperatures for the plate material is taken from Attachment 3 to the base calculation, GE Report, GE-NE-0000-0000-8763-01, "P/T Curves for First Energy Corp. Using the K_{Ic} Methodology". This report is an Attachment to DIN 19.

Addendum 01 to the calculation (DIN 19) reviewed weld material heat number 5P6214B and changed the adjusted reference temperature based on industry testing at other plants (BWRVIP 135 Integrated Surveillance Program). However, it was noted that another seam weld material was controlling and therefore the P/T curves were not revised. From DIN 19, Attachment 3, there are 3 plates that make up the beltline ring #2 and one surveillance plate (best estimate chemistry). The greater of the beltline adjusted reference temperatures of the 4 heats is used and is identified in Table 1.

The layout of the four instrument N12 nozzles was checked against the vessel vertical and horizontal seam welds. Only one nozzle, at azimuth 164 degrees (DIN 5) appeared to be near a vertical seam weld BE at azimuth 160 degrees. There is approximately 8 inches on the outer surface between the centerlines of the nozzle and weld. A review of the initial RT_{NDT} values for the plate and weld material found in DIN 19's Attachment 3 Table 4-4b "Perry Beltline ART Values (32 EFPY)" was conducted. It was determined that the plate, C2557-1 initial values were greater, than those of the weld BE. Therefore the plate controls and not the weld from a material fracture toughness standpoint and therefore the plate ART values are used in this analysis.

Table 1

Plate and Weld Material Adjusted Reference Temperature

Plate or Weld	Heat No. or lot	22 EFPY ART ^{0}F	32 EFPY ART ^{0}F
MK 22-1-1 shell 2	C2557-1	52	59
MK 22-1-2 shell 2	B6270-1	12	19
MK 22-1-3 shell 2	A1155-1	32	39
Surveillance plate (best estimate chemistry)	C2557-1	47	53
Seam weld BE	5P6214B	-9	-4
Seam weld BE	626677	3	6
Seam weld BE	624063	-4	4
Seam weld BE	627069	-37	-34
Surveillance weld (best estimate chemistry)	5P6214B	-2	5
Integrated Surveillance BWRVIP [1]	5P6214B	-	12[1]

Notes:

1. Refer to Addendum 01 to Calculation EA-0246.

Review of the above data indicates the plate material, Heat No. C2557-1, controls ART for the WLIN.

B. Assume a coolant temperature, T.
The coolant temperatures considered for all the curves are in the range from 70 to the temperature corresponding to approximately 1400 psig which matches the pressure value of existing TS curves. The actual temperatures are identified in the attached spreadsheet calculation.

C. Calculate the allowable stress intensity factor, K_{Ic}, using equation 2.4-2 for the assumed fluid temperature, T, and the limiting ART for the region being evaluated.

K_{Ic} is the lower bound static fracture toughness (ksi-inch$^{1/2}$)

$$K_{Ic} = 33.20 + 20.734 * \exp[0.0200(T - ART)]$$ Equation 1

Applying the ART from Table 1 for the applicable 22 or 32 EFPY and the temperatures assumed in "B" above, corresponding values of static fracture toughness may be calculated for each temperature.

D. Calculate the thermal stress intensity factor, K_{It}, using one of the methods described in Sections 2.5.
For the WLIN, the BWROG Report BWROG-TP-11-023-A (DIN 17) was prepared. A finite element bench mark analysis is included in the report. Also, this report references the Perry specific finite element analysis, Reference 11 on page 10-1. Reference 11 is Structural Integrity Calculation No. 0900876.303, "Instrument Nozzle Stress Intensity Factor Calculation for Plant Specific 238 –Inch BWR" (DIN 15). The Perry specific analysis was used as a bench mark for the BWROG Report. As permitted in Section 2.5, finite element analysis is an acceptable approach in the necessary stress analysis for RPV regions. A description of the Perry specific analysis follows.

Calculation No. 0900876.303 finite element model heat transfer coefficients were leveraged from the values used by Chicago Bridge and Iron (CBI) in DIN 31 and DIN 32. Surface heat transfer coefficient of 0.2 BTU/hr-ft^2-^{0}F which is identical to that used in BWROG-TP-11-023-A and the inside surface value applied varied with temperature. Section 5 of Calculation No. 0900876.303 describes the model as a 3-D FEM constructed in ANSYS. Dimensions were taken from the original CBI analysis Design Report, DIN 33. Three dimensional SOLID45 elements were used for structural analyses, and 3-D SOLID70 elements are used for thermal analysis. The stainless steel RPV clad, Inconnel J-groove weld, Inconnel butter, and stainless steel instrument nozzle are modeled as separate materials. The air gap between the instrument nozzle and RPV shell is modeled for the thermal analysis only. A shutdown transient of 100 degrees F/ hr was analyzed as was done in BWROG-TP-11-023-A. Pipe reactions from the instrument line were not included as the resulting stress was determined to be negligible. The results of the finite analysis, Calculation No. 0900876.303, determined the unit pressure and maximum thermal stress intensity values of 68.4 ksi-in$^{0.5}$ and 38.6 ksi-in$^{0.5}$ respectively.

E. Calculate the allowable pressure stress intensity, K_{Im}, using the methods described in Sections 2.5.2 and 2.5.3.

The methods used in 2.5.2 and 2.5.3 are identical to that used in the ASME Code, Section XI, Non-mandatory Appendix G (DIN 3) K_{Im} is the allowable stress intensity factor resulting from membrane (pressure) stress (ksi-in$^{1/2}$)

$$K_{Im} = (K_{Ic} - K_{It}) / SF$$ Equation 2

Where SF = 2 for Level A & B Service Limits and SF = 1.5 for leak test condition. K_{Ic} and K_{It} were previously defined above. Note that K_{Im} will vary with temperature due to the influence of K_{Ic}.

F. Calculate the allowable pressure, P_{allow}, using the methods described in Sections 2.5.2 or 2.5.3.

From Section 2.5.3-6,

The allowable pressure, P_{allow}, for a 1/4t postulated limiting (axial) defect is defined as follows:

$P_{allow} = (K_{Im}\,P) / K_{Ip\text{-}applied}$ Equation 3

where: P_{allow} = the allowable internal pressure (psi)

K_{Im} = the allowable pressure stress intensity factor (ksi inch)

P = the operating pressure (psi)

$K_{Ip\text{-}applied}$ = the applied pressure stress intensity factor (ksi inch)

P_{unit} is the operating pressure and K_{Ip} is the stress intensity factor due to the applied unit pressure. DIN 15 Table 8 indicates a 1000 psig unit pressure was applied and resulted in a stress intensity factor 69.4 ksi-in$^{1/2}$. It is noted in Section 2.3.1 (DIN 15) that the results of the pressure load case are linear, the evaluated pressure is a "unit" loading, the results of which are scaled by the actual pressure.

Pressure adjustments to the ratio of P_{unit} to $K_{Ip\text{-}applied}$ are not necessary. It is noted that the existing ratio is 14.40922 (1000/69.4). Ratio of the unit results to the hydro test condition yields a K_{Ip} stress intensity of 108.4 ksi-in$^{1/2}$ (69.4* 1563/1000). Since these results are simple ratios of one another the ratio of P_{unit} to $K_{Ip\text{-}applied}$ remains a constant. Accounting for static head increases the pressure slightly. For the WLIN nozzle, the increase in pressure due to vessel static head at the nozzle is calculated as follows. Table 2 indicates the hydro-test pressure is 1563 psig and the static head of the fluid may also be taken into account. Hydrostatic pressure calculated assuming a full vessel:

$$P = (H-F)\ 0.0361\ \text{psi/inch} = P_{static}\ \text{psig}$$
$$= (853.13 - 363.5)\ 0.0361 = 17.67\ \text{psig}$$

The hydro-test pressure added to static pressure results in 1580.67 psig (1563 + 17.67). The new stress intensity is obtained by ratio of the previous results, 109.6985 ksi-in$^{1/2}$ (1580.67/1000 * 69.4). This results in a ratio of P_{unit} to $K_{Ip\text{-}applied}$ of 14.40922 (1580.67/109.6985).

Revision 2 to this calculation adds the total vessel static head to the upper vessel and beltline curve (see Section J below) which includes the effects of the WLIN curve.

Table 2

Reactor Vessel Physical Data

Normal Operating Dome Pressure 105% uprate	1025 psig [2]
Design Pressure	1250 psig [25]
Hydro-Pressure (top of vessel)	1563 psig [25]
Leak Test Dome Pressure	1025-1050 psig [10]
Vessel height above vessel zero (cold)	H = 853.13 inches (from vessel 0 invert") [12]
Top of Active Fuel and WLIN elevation	F = 363.5 inches (above vessel invert) [12]

Bottom of vessel invert	0 inches [12]
Vessel Inside Radius to Base Material	R=120.1875 [13]
Minimum Vessel Thickness (shell no. 2 ring)	t = 6 inches [25]

G. Determine (T-ART) based on a combination of the above equations

This may be necessary to facilitate graphing when the pressure and ART are known and the corresponding temperature is desired. Combining all 3 equations, and applying logarithm rules results in the following:

$$T - ART = Ln\left(\frac{K_{Ic} - 33.20}{20.734}\right) / 0.02$$

$$T - ART = Ln\left(\frac{(SF * K_{Im}) + K_{It} - 33.20}{20.734}\right) / 0.02$$

Further substituting $K_{Im} = P_{allow}/14.409$ and $K_{It} = 38.6$ and re-arranging the terms yields:

$$T = ART + Ln\left(\frac{SF * 0.069 * P_{allow} + 5.4}{20.734}\right) / 0.02 \qquad \text{Equation 4}$$

Note that this equation can be useful in determining the intersection of two curves when graphing a solution.

H. For the non-beltline P-T limits, apply the additional minimum temperature requirements described in Sections 2.7 and 2.8.

The minimum temperature requirements discussed in Sections 2.7 and 2.8 are taken from the 10CFR50 Appendix G. These are discussed extensively in Section II below.

I. Apply any applicable adjustments to the final temperatures and pressures, as described in Section 2.6.

Temperature and pressure measurement accuracies have not been considered in the past at Perry when using the plant computer data for cool down rates or leak tests. Instruments which control critical trips such as reactor level are accurate within a few inches. For example, DIN 34, Reactor Vessel Low Level Trip calculation indicates an analytical limit of 121.9 inches verses an allowable limit of 126.233 inches of H_2O. The increase of 4.333 inches accounts for loop accuracy, loop calibration accuracy, process measurement accuracy, primary element accuracy and insulation resistance. In item F above it was shown that approximately 490 inches of static head will not influence the pressure significantly to alter P_{allow}. The recommendation found in BWROG Report NEDC-32972 (DIN 35), Table 4, with respect to TS 3.4.11, is that there is substantial margin in the development of the curves which are based on regulatory and ASME Code requirements and this margin will accommodate any measurement uncertainties. It is concluded from this review that inclusion of uncertainty has been satisfactorily considered.

J. Inclusion of Vessel Static Head in P/T Curves.

This section is added per Revision 2 of this calculation. An independent review by the author of the PTLR identified that the vessel static head is included in the PTLR DIN 16 and 17 examples. Therefore the static pressure of a full vessel is to be included in the new curves. From Table 2, the top of the vessel to the bottom invert is 853.13 inches. The corresponding static head is 853.13 inches x 0.0361 psi/inch = 30.8 psig. This value is used to decrease the allowable pressure for the upper vessel and beltline curves. The bottom head curves were not changed in this update and the data source for these curves are referenced in the attachments.

For the leak test Curve A, the impact to both 22 and 32 EFPY curves occurs above 100 degrees and drops the curve uniformly by 30.8 psig. The difference between the 22 and 32 EFPY curves results from the change in the beltline material reference temperature. **ONLY the 32 EFPY curves (P/T Limits) are to be submitted.**

The non-critical Curve B is impacted by 10CFR50 Appendix G pressure limits. The points of intersection on the 312.5 psig flat part of the curve required temperature iteration to solve for a pressure of 343.3 psig (312.5 + 30.8). For the 22 EFPY Curve B, the upper vessel and beltline curve temperature of 94.773 is an iterative result for a given pressure of 343.3 psig and as indicated previously, the 343.3 psig reduced by vessel static head of 30.8 psig results in 312.5 psig. Similarly, for the 32 EFPY Curve B, the upper vessel beltline curve temperature of 105.974 degrees is an iterative result for a given pressure of 343.3 psig which when reduced by vessel static head results in 312.5 psig. The shift in temperature from 98.97 to 105.974 degrees is a result of the fluence on the beltline material adjusted reference temperatures for 22 and 32 EFPY respectively.

The core critical Curve C is also impacted by 10CFR50 Appendix G pressure limits. The same temperature iteration is performed to determine the temperature at which the 312.5 psig 10CFR50 Appendix G portion of the curve is intersected. For the 22 EFPY Curve C, the temperature is 138.97 degrees. This temperature includes the 40 degree shift from 10CFR50 Appendix G, therefore at 98.97 degrees the resultant pressure was approximately 343.3 psig or 312.5 psig when reduced by the reactor static head of 30.8 psig. Similarly, for the 32 EFPY Curve C, an iterative process found the temperature of 145.98 degrees would result in a pressure of 312.5 psig. The temperature reflects the increase of 40 degrees as required by 10CFR50 Appendix G and the pressure reflects reduction of 30.8 psig to account for reactor vessel hydrostatic head.

II. 10CFR50 Appendix G Technical Requirements for P/T Curves & Acceptance Criteria

The Code of Federal Regulations, Appendix G to Part 50 – Fracture Toughness Requirements establishes the requirements for ferritic materials of the pressure - retaining components of the reactor coolant pressure boundary. These requirements provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. The ASME Code forms the basis for the requirements, specifically, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components". The Sections, Editions, and Addenda of the ASME Code specified in 10CFR50.55a have been approved for incorporation by reference in the

Federal Register. 10CFR 50, Appendix G, Section 1 Reactor Vessel Upper – Shelf Energy requirements are as follows:

Item a., states the vessel beltline materials must have a Charpy Upper Shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lbs initially. A Charpy upper – shelf of 50 ft-lb must be maintained throughout the life of the vessel, unless demonstrated in a manor approved by the regulatory process that lower values of Charpy upper – shelf energy will provide margins of safety against fracture. These margins of safety required against fracture must be equivalent to those required by Appendix G of Section XI of the ASME Code. The latest edition of the ASME Code incorporated by reference into 10CFR50.55a(b)(2) is to be used at the time the analysis is submitted. From an internet copy of 10CFR50.55a from the NRC web page, the following is stated:

> 2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, and include the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition (Division 1) through the 2004 Edition (Division 1),...

Review of the supporting calculation, DIN 15, indicates the 2004 Edition of Appendix G to Section XI is referenced which correctly lines up with the above Federal requirements.

Additional 10CFR Appendix G Pressure-Temperature Limits and Minimum Temperature Requirements are as follows:

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In Table 1, the vessel pressure is defined as a percentage of the pre-service system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions. From DIN 20, the Perry ASME Code vessel hydro-test pressure was 1563 psig. The hydrostatic test pressure is not less than 1.25 times the design pressure. The Perry specific 20% of design hydro-test value discussed in Table 3 below is 312.5 psig (1250 x 1.25 x 0.20).

b. The pressure-temperature limits identified as "ASME Appendix G limits" in Table 3 below require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

c. The minimum temperature requirements given in Table 3 below pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 1, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical.

The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in Table 3.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

Table 3

Pressure and Temperature Requirements for the Reactor Pressure Vessel

Operating condition	Vessel pressure[1]	Requirements for pressure-temperature limits	Minimum temperature requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	([2])
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	([2]) +90 ° F([6])
1.c No fuel in the vessel (Preservice Hydrotest Only)	ALL	(Not Applicable)	([3]) +60 ° F
2.a Core not critical	≤20%	ASME Appendix G Limits	([2])
2.b Core not critical	>20%	ASME Appendix G Limits	([2]) + 120 ° F.
2.c Core critical	≤20%	ASME Appendix G Limits + 40 ° F.	Larger of [([4])] or [([2]) + 40° F.]
2.d Core critical	>20%	ASME Appendix G Limits + 40 ° F.	Larger of [([4])] or [([2])+160°F]

2.e Core critical for BWR (5)	≤20%	ASME Appendix G Limits + 40 ° F.	(2)+60°F

1 Percent of the preservice system hydrostatic test pressure.
2 The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
3 The highest reference temperature of the vessel.
4 The minimum permissible temperature for the inservice system hydrostatic pressure test.
5 For boiling water reactors (BWR) with water level within the normal range for power operation.
6 Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

Perry Minimum Temperature Requirements of 10CFR50 Appendix G
These requirements are unchanged as a result of the WLIN impact but are presented here to allow for a standalone calculation.

For the hydrostatic leak test core not critical, Note 2, requires the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. DIN 19's Attachment 3 explains that the Charpy data for the Perry closure studs did not meet all the 45 ft-lb and 25 MLE ASME Code Section III, Subsection NB-2300 requirements at 10 degrees F. Therefore the lowest service temperature (LST) for the bolting material is at the test temperature + 60 degrees F (70 degrees F), DIN 19, Attachment 3, Table 4-3. It is concluded that for pressure less than or equal to 312.5 psig (20% of vessel hydro test pressure) the minimum temperature is limited to 70 degrees F. For pressure greater than 312.5 psig, typically the larger of the LST or RT_{NDT} + 90 degrees F or 100 degrees F has been used. RT_{NDT} values for the closure flange materials are found in DIN 19, Attachment 3,Table 4-1 and is equal to 10 degrees F.

For the core not critical operation, at pressures less than or equal to 312.5 psig, the same minimum temperature limit applies as the leak test, which is 70 degrees F which is tied to stud LST. For core not critical and pressures greater than 312.5 degrees F, the greatest reference temperature of the material in the closure flange region is 10 degrees F, DIN 19, Attachment 3,Table 4-1. Therefore the RT_{NDT} + 120 is 130 degrees F.

Lastly, for core critical operation, item 2e for BWRs from Table 3, the minimum temperature at pressure less than or equal to 312.5 psig is the same as the previous minimum temperature limit associated with the leak test and core not critical limit associated with the closure flange region which is 70 degrees F. For core critical, with pressure greater than 312.5 psig, the greatest reference temperature of the material in the closure flange region is 10 degrees F, DIN 19, Attachment 3,Table 4-1. Therefore the minimum temperature at pressure greater than 312.5 degrees is 170 degrees F (10+160).

III Computations
The overall goal of this section is to develop a composite curve of ASME Appendix G and 10CFR50 Appendix G limiting requirements. DIN 19 Section 5 indicates there are four vessel regions that are monitored against P/T curve operating limits. These are the Closure Flange Region, Core Beltline Region, Upper Vessel, and Lower Vessel. This calculation will simply superimpose the WLIN curve on top of the existing curves developed in DIN 19 and develop a composite or limiting curve for the applicable region. The bottom head curve is unaffected by the WLIN and remains unchanged.

The computations for the curve construction are organized in the order and sequenced numbering as they exist in the current Technical Specification. This is done to minimize verification effort between this calculation and the Technical Specification submittal. The pressure as defined in all the curves is that of the top head. Section I A of this calculation identified the plate material, Heat No. C2557-1, controls ART for the WLIN. The ART used in the computations are 52 and 59 degrees for 22 and 32 respective EFPY. Pressures recorded less than 0 psig are acceptable and are addressed in Section V of this calculation.

A. Curve A Leak Test Curves **[ONLY the 32 EFPY curve (ref. Attachment 2) will be submitted.]**
The leak test assumes a limiting heat up and cool down rate of 20 degrees F per hour. Current industry practice as indicated in the PTLR (DIN 16 Page 2-16 definition of K_{It}) is thermal transient results do not apply. The thermal transient results at the low 20 degree per hour rate are near isothermal conditions. If the plant exceeds the 20 degree per hour limit then Curve B or Curve C may be used. Therefore the K_{It} of 100 degrees per hour is reduced to 0. The factor of safety is 1.5. The ART used in the computations are 52 and 59 degrees for 22 and 32 respective EFPY. The bottom head curve remains unchanged.

For the 22 EFPY curve the data points are plotted in Attachment 1. Figure 1of Attachment 1 is the update of Technical Specification (TS) Figure 3.4.11-1(a) Pressure Test Curve (Curve A) (Valid to 22 EFPY-Unit 1). The Upper Vessel and Belt line limits are a composite of the 10CFR50 Appendix G minimum temperature requirements and the WLIN impact on the beltline. Figure 2 of Attachment 1 provides a comparison of the existing and new WLIN beltline curves. The WLIN curve controls pressure over the existing curve at temperatures above approximately 100 degrees F and 808 psig.

For the 32 EFPY curve the data points are plotted in Attachment 2. Figure 1 of Attachment 2 is the update of TS Figure 3.4.11-1(d) and includes the WLIN results. Figure 2 of Attachment 2 compares the current TS beltline curve to the WLIN beltline curve. The WLIN curve is more conservative (requires higher temperature for same pressure) than the current beltline curve. The WLIN Curve controls pressure above 100 degrees F and 740 psig.

B. Curve B Non-Nuclear Heatup/Cooldown **[ONLY the 32 EFPY curve (ref. Attachment 4) will be submitted.]**
The Non-Nuclear Heatup and Cooldown curve considers a thermal transient of 100 degrees per hour. For these Curves K_{It} = 38.6 ksi-inch$^{1/2}$. The factor of safety for the pressure stress intensity, K_{Im}, is 2.0. Lastly, The ART used in the computations is 52 and 59 degrees for 22 and 32 respective EFPY. The bottom head curve remains unchanged.

For the 22 EFPY curve the data points are plotted in Attachment 3. Figure 1 of Attachment 3 is the update of TS Figure 3.4.11-1(b) Non-Nuclear Heat up / Cool down Curve (Curve B) (Valid to 22 EFPY-Unit 1). The Upper Vessel and Belt line limits are a composite of the 10CFR50 Appendix G minimum temperature requirements and the WLIN impact on the beltline. Figure 2 of Attachment 3 provides a comparison of the current beltline curve and new more limiting WLIN beltline curve. The WLIN curve controls pressure over the existing curve at temperatures between 70 and approximately 99 degrees and at temperatures above 130 degrees F.

For the 32 EFPY curve the data points are plotted in Attachment 4. Figure 1 of Attachment 4 is the update to TS Figure 3.4.11-1(e) Non-Nuclear Heatup / Cool down (Curve B) (Valid up to 32 EFPY-Unit 1). Figure 2 illustrates the impact of the WLIN curve to control pressure between 70 and approximately 106 degrees and greater than 130 degrees.

C. Curve C Core Critical Curves **[ONLY the 32 EFPY curve (ref. Attachment 6) will be submitted.]**
The core critical curves assume a thermal transient of 100 degrees per hour. For these Curves K_{It} = 38.6 ksi-inch$^{1/2}$ and the factor of safety for the pressure fracture stress intensity, K_{Im}, is 2.0. The curves have an additional temperature shift at allowable pressure of T + 40 degrees F over the non-core critical curves. This shift is required by 10CFR50 Appendix G.

For the 22 EFPY curve the data points are plotted in Attachment 5. Figure 1 of Attachment 5 is the update of TS Figure 3.4.11-1(c) Core Critical Operation (Curve C) (Valid to 22 EFPY-Unit 1). The Upper Vessel and Belt line limits are a composite of the 10CFR50 Appendix G minimum temperature requirements and the WLIN impact on the beltline. Figure 2 of Attachment 5 provides a comparison of the current beltline curve and new more limiting WLIN beltline curve. The WLIN curve controls pressure over the existing curve at temperatures between 70 and approximately 139 degrees F and at temperatures above 170 degrees.

For the 32 EFPY curve the data points are plotted in Attachment 6. Figure 1 of Attachment 6 is the update of TS Figure 3.4.11-1(f) Core Critical Operation (Curve C) (Valid Up to 32 EFPY- Unit 1). The Upper Vessel and Belt line limits are a composite of the 10CFR50 Appendix G minimum temperature requirements and the WLIN impact on the beltline. Figure 2 of

Attachment 6 provides a comparison of the current Upper Vessel Beltline curve to the WLIN Beltline curve. The WLIN is more conservative (lower allowable pressure) than the previous curve between 70 and approximately 146 degrees and at temperatures above 170 degrees.

IV. Evaluation of Plant operation and the impact of the calculated WLIN 22 and 32 EFPY Curves
The 22 EFPY curves are reviewed because the Core Average Exposure Summary, PDB-A008 (DIN 22), indicates 21.79 effective full power years equivalent exposure is predicted at the end of Cycle 15. This exposure includes a best guess predicted exposure for the current cycle of 1.74 EFPY. Therefore use of the 22 EFPY Curve to assess the plant leak tests and start-up and shutdowns is appropriate. The 32 EFPY curves are reviewed since they will be required very close to the beginning of the next cycle. **[NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]**

Leak Test Data
ISI-B21-T1300-1 Rev 17, Reactor Coolant System Leakage Test, is the procedure that documents the results of reactor pressure tests. Reactor vessel temperature is recorded on the surface of the reactor at locations identified in TABLE 4.

TABLE 4
Leak Test Recorded Instrument Points

ISI-B21-T1300-1 Designation	Asset Labels	Description[1]	Reference Azimuth Degrees
Point 1	1B21N029A	Vessel Head Flange	135
Point 2	1B21N030A	Bottom Head	339
Point 3	1B21N050B	Shell Flange	270

Notes:
1. 208-0010-00003 Rev T (DIN 36)

The reactor pressure is recorded at 1C34-R609-red (1H13-P680-3B). Temperature monitoring points, 1 and 2, identified in Table 4 are for the upper flange region and the lower of the two temperatures has been used in the plots. Reactor narrow range pressure (850-1050 psig) data was used over 1B21R23A-red (1H13-P601-20B) Wide Range Pressure (0-1500 psig). The leak test data from the last three refuels was plotted against the revised WLIN curve for 22 EFPY. Test pressure slightly exceeded the new 22 EFPY Curve. A review of the 32 EFPY WLIN curve indicated that test temperatures will have to increase above 125 degrees F to achieve similar leak test pressures.

Exceeding the new 22 EFPY Curve is explained by the differences in the methodologies used to calculate the WLIN impact on the new curves. The Non-conservative Technical Specification supporting analyses were performed by GEH. In DIN 14 and 21 Appendix J, GEH indicated they performed a finite element analysis (FEA) of the WLIN. From this model, P/T curve changes were submitted to the regulator as well as the utilities. It is not uncommon for the finite element crack model to result in lower stress intensity values than the boundary integral evaluation (BIE) results used in this calculation. The FEA and BIE methods are both technically acceptable and both have been approved by the regulator. Therefore, the small violations to the new curve are judged to be acceptable without additional review or analysis. **[NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]**

Heat Up and Cooldown Data
The following heat up and cool down data has been reviewed and compared to the 22 EFPY Core Critical revised WLIN curve. The Core Critical curve is more limiting than Non-core Critical thus satisfying its P/T limits assures envelope of the Non-critical curves.

Sept. 8, 2013; Order 200575505; heat up performed under SVI –B21-T1176 Rev. 12. Minimum reactor flange temperature recorded at the beginning of the heat up was 223.0 degrees F (0700 hrs) with a recorded pressure of 20 psig. This point is

well to the right of the 22 EFPY and 32 EFPY WLIN P/T curve (Attachment 5 & 6). No transients exceeded 100 degrees F/hr. [NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]

Sept. 6, 2013; Order 200575435; Cool down under SVI-B21-T1176 Rev. 12. Temperature recorded at bottom head drain and bottom flange. At the minimum coolant temperature of 189 degrees, pressure was recorded at or near 0 psig. No transients exceeded 100 degrees F/hr.

June 20, 2013; Order 200567745; heat up under SVI-B21-T1176 Rev 12. The recorded P/T values were compared to the 32 EFPY Core Critical Curve (Attachment 6). Recorded values were well below the curve and therefore acceptable. No transients exceeded 100 degrees F/hr.

June 16, 2013; Order 200567165; cool down under SVI-B21-T1176 Rev. 12. The recorded P/T values were compared to the 32 EFPY Core Critical Curve (Attachment 6). Recorded values were well below the curve and therefore acceptable. No transients exceeded 100 degrees F/hr.

May 11, 2013; Order 200563581; heat up under SVI-B21-T1176 Rev 12. The recorded P/T values were compared to the 32 EFPY Core Critical Curve (Attachment 6). Recorded values were well below the curve and therefore acceptable. No transients exceeded 100 degrees F/hr.

May 3, 2013; Order 200562976; RPV Leak Test under SVI-B21-T1176 Rev 12. The recorded P/T values were compared to the 32 EFPY Leak Test Curve (Attachment 2). **Recorded values exceeded curve** near 120 degree F 993.55 psig limit at 0600hrs. The new 22 EFPY Leak Test Curve was slightly exceeded but is judged acceptable based on discussion above under leak test data. The 32 EFPY Leak Test Curve was not met. No transients exceeded 100 degrees F/hr.

March 18, 2013; Order 200556257; Cool down under SVI-B21-T1176 Rev. 12. Temperature recorded at bottom head drain and bottom flange. At the minimum coolant temperature of 200 degrees, pressure was recorded at or near 0 psig. The data is enveloped by the most limiting core Critical Curve C at 32 EFPY. No transients exceeded 100 degrees F/hr.

January 26, 2013; Order 200546116; Heat up under SVI-B21-T1176 Rev 12. The recorded P/T values were compared to the 32 EFPY Core Critical Curve (Attachment 6). Recorded values were well below the curve and therefore acceptable. No transients exceeded 100 degrees F/hr.

January 22, 2013; Order 200546240; The data from SVI-B21-T1176 indicates that a non-nuclear cool down was in process. The vessel bottom flange temperature monitoring indicated the bottom head curve, Curve B, non-nuclear heat up and Cooldown was not exceeded. Vessel bulk temperature as recorded by 1B33R604 never exceeded the WLIN 22 or 32 EFPY Upper Vessel and Beltline Curve (Attachment 3 and 4 respectively). No transients exceeded 100 degrees F/hr.

June 17, 2012; Order 200509654; Data review indicated a heat up condition. The data satisfied the curve C, Core critical curve P/T requirements for the WLIN 32 EFPY curve (Attachment 6). No transients exceeded 100 degrees F/hr.

June 14, 2012; Order 200509302; The data indicated a non-nuclear Cooldown was in process at 01:00 hours. Minimum temperature of the bottom head drain indicated 207 degrees F at 0 psig. This value is to the right of the limiting WLIN 32 EFPY Curve B, Attachment 4. No transients exceeded 100 degrees F/hr.

March 3, 2012; Order 200495466; Data indicates a heat up condition beginning with a reactor flange temperature of 178 degrees F. Data comparison to the limiting WLIN 32 EFPY Curve C (Attachment 6) was satisfactory. No transients exceeded 100 degrees F/hr.

March 1, 2012; Order 200495135; Work order review indicated a Non-Nuclear Cooldown, Curve B applies. Reactor bottom head drain minimum temperature was approximately 204 degrees at 0 psig. This is well to the right of the limiting WLIN 32 EFPY Curve C (Attachment 4). No transients exceeded 100 degrees F/hr.

October 18, 2011; Order 200479294; Data review a heat up condition. Reactor flange temperature recorded as 183 degrees at 0 psig. Applying the core critical WLIN 32 EFPY curve (Attachment 6) indicated all values to the right of the curve and satisfactory. No transients exceeded 100 degrees F/hr.

October 2, 2011; Order 200477615; Data review indicated a Cooldown event. At a reactor bottom head drain coolant temperature of 213 degrees reactor pressure was recorded at 0 psig. The most limiting Curve, Core critical at 32 EFPY (Attachment 6) is satisfied. No transients exceeded 100 degrees F/hr.

June 4, 2011; Order 200463010; Data review indicated a heat up event. Reactor flange temperature was recorded at 132 degrees with 4 psig in the vessel. Gradual pressurization was applied. The most limiting Core critical Curve C, WLIN Curve C at 32 EFPY (Attachment 6) was satisfied. No transients exceeded 100 degrees F/hr.

May 31, 2011; Order 200462243; SVI-B21-T1176 Rev 12 data indicated an inservice leak test was performed. Review of the flange temperature data and pressure indicated that the most restrictive curve for pressure test, WLIN at 32 EFPY was slightly exceeded at 122.2 degrees and 1033 psig. The new 22 EFPY Leak Test Curve was slightly exceeded but is judged acceptable based on discussion above under leak test data. Observed heat up rate was less than 20 degrees per hour. Cooldown from the leak test found both 22 and 32 EFPY WLIN curves (Attachments 1 and 2) satisfied as well as the 20 degree per hour cooling rate.

April 17, 2011; Order 200455955; SVI-B21-T1176 Rev 12 data indicated a Cooldown event. Bottom head drain temperature was 191 degrees at 0 psig satisfying the most critical curve WLIN at 32 EFPY Curve C (Attachment 6). No transients exceeded 100 degrees F/hr.

May 17, 2010; Order 200416463; SVI-B21-T1176 Rev 12 data indicated a Heatup event. At 12:06 control rod drive withdraw commenced with reactor flange temperature of 130 degrees and 0 psig RPV pressure. Data satisfied the most limiting WLIN 32 EFPY Core Critical P/T limits (attachment 6). The heat up rate was within 100 degree per hour limit.

May 13, 2010; Order 200416155; SVI-B21-T1176 Rev 12 data indicated heat up was terminated. No data to review.

May 11, 2010; Order 200416154; SVI-B21-T1176 Rev 12 data indicated Cooldown event. At 3 psig, reactor bottom drain temperature was 219 degrees at 0130 hours. This point is well to the right of the most limiting Curve C, WLIN 32 EFPY (Attachment 6) core critical Cooldown. No transients exceeded the 100 degree per hour limit.

November 1, 2009; Order 200392883; SVI-B21-T1176 Rev 12 data indicated a nuclear Heatup. Reactor flange temperature was approximately 141 degrees at 2 psig in the reactor vessel. The most limiting curve C WLIN 32 EFPY (Attachment 6) was satisfied. No transients exceeded the 100 degree per hour limit.

October 15, 2009; Order 200391448; SVI-B21-T1176 Rev 11 data indicated a Cooldown condition. Reactor pressure of 2 psig was reached with reactor bottom head drain temperature of 199 degrees F. The most limiting Curve C WLIN 32EFPY (Attachment 6) was satisfied. No transients were observed that exceeded the 100 degree per hour limit.

Conclusion
Historical review of heat up and cool down data has been conducted over approximately 5 years. The historical data indicates no violations of the 22 EFPY revised curves were observed. The 32 EFPY curves were satisfied with the exception of the leak test curves. The minimum temperature of the leak test needs to be increased to satisfy the impact of the WLIN Leak Test curve at 32 EFPY, Attachment 2. [NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]

V. Pressure less than 0 PSIG in the Reactor
Perry operation has included operating the vessel at a vacuum (less than atmospheric pressure). CR 2009-66988 documented a condition where emergency service water was not available and a vacuum was maintained in the main condenser while

MSIVs remained open in order to provide immediate access to an alternate decay heat removal method (described in ONI-E12-2 attachment 11, "Cold Shutdown Heat Removal by Steaming'). Another instance of reactor vessel vacuum occurs when the plant starts up or shuts down. CR 2011-03864 identifies an NRC resident concern that RCS Pressure and Temperature Limits for Tech Spec 3.4.11 curves for Non-Nuclear heatup has a lower pressure number on the curve that stops at "O" PSIG. Potential may exist that drawing a vacuum and heating up the reactor would put us outside the curve. It was requested that Engineering evaluate the concern and to determine if any potential exists of not being within the curve boundaries.

The discussion that follows supersedes that discussed in the CR 2011-03864. This is a result of the new curves in the attachments that incorporate the stress concentration effect of the WLIN in the vessel plate. The overall conclusion that the vacuum has no adverse effects on the reactor however remains unchanged.

Allowable External Pressure
The Reactor Vessel is designed following the rules of ASME Section III Subsection NB for Class 1 components. From the Chicago Bridge and Iron ASME Code Data Report for the reactor vessel, DIN 25 identifies, the vessel head is 4 19/32 inch thick (no clad) with an inner radius of 119 inches. DIN 37 identifies the vessel wall below the flange is 6 inches thick (with nominal cladding thickness of 3/16 inch) with an inner radius of 120 inches. This results in a diameter of approximately 252.4 inches. In both the abnormal case of cold shutdown heat removal by steaming and the start-up case either critical or non-critical, the condenser is the source of the reactor vacuum. The condenser is basically a square box constructed of commercial grade materials. DIN 24 identifies a maximum water box pressure of 100 psig. Applying the ASME Code rules for external pressure design to the reactor vessel provides a sense of margin that the vessel has relative to the condenser.

To simplify the analysis it is assumed that a vacuum in the vessel is equivalent to an external pressure of 14.7 psia with zero psia in the vessel. For determining the maximum permitted external pressure, DIN 26 rules are applied following the procedure of NB-3133.2 For the shell of the vessel, the ratio of the outer diameter (D) to thickness (T) is 252.4/6 = 42.1. The length of the vessel (including both heads) is approximately 853 inches. Including the reactor heads in the length provides a conservative (more flexible shell with respect to collapse). The length to diameter ratio becomes L/D = 853/252.4 = 3.38. Solving for the factor, A, from Figure G of Section II Part D Subpart 3, the value of "A" is approximately .0014. Per DIN 25, the shell is constructed of SA-533 Gr-B Class 1 material. The factor "B" is determined from Figure CS-5 (page 752) "Chart for Determining shell Thickness of Components under external Pressure developed for "... SA-533 Class 1 Gr A,B,C ...." The "B" factor at 500 degrees is greater than 15,000. From NB 3122.2, the maximum external pressure is:

$P = \dfrac{4B}{3D/T}$ Solving this equation with D/T = 42.1 and B equal to 15,000 (conservatively envelopes 550 degrees F) results in an allowable external pressure of 475 psi.

For the spherical head, the procedure from NB-3133.4 is followed. The material is the same as the shell per DIN 25. The flange would add stiffness and is ignored in this calculation.

$A = \dfrac{0.125}{R/T}$ R is the inner radius of the head 119 inches and T = 4 19/32. Solving for A results in a value of 4.825E-03.
From Figure CS-5 for the value of A = 4.825 E-03 is B = 18,000. The external pressure is calculated from:

$P = \dfrac{B}{R/T}$ The result is a pressure of 694 psi.

The results of this conservative review indicate the shell is limiting at 475 psi external pressure. For this assessment the flange stiffness has been ignored for both shell and head external pressure calculations and the shell was assumed the full length of the vessel. It is concluded that the vessel is in no danger of collapse from an external pressure or vacuum from the condenser. A substantial margin exists between the reactor and the condenser.

In the assessment applied above, ASME Code rules were applied from the 2010 ASME Boiler Code. It is understood that 10CFR50.55a, DIN 28, recognizes only ASME Editions through the 2008 Addenda (Division 1). The 2014 Code was readily available and the same formulae exist in earlier Code Editions. Conclusions of the calculation with respect to the structural adequacy of the vessel under minor external pressure load are not likely to change due to the robust design. Therefore, use of the later Code is judged acceptable.

Further review of earlier ASME Codes found that the 1983 Edition through Summer 84 Addenda included the same formulation as well as the same curves. A spot check was conducted between the ASME 2010 applicable Sections and the ASME 1983 edition through the Summer 84 Addenda. The formulation was found to be the same and the charts for determining formula variables were the same for the vessel. Except for the administrative move of the external pressure / material curves from Appendices in the 1983 Code to Section II Part D Properties (Customary) Materials in the 2010 Code, no other changes were noted that would change the results as presented.

Fracture Toughness under Vacuum

Procedure IOI-0001, Cold Startup Rev 38, DIN 29, identifies in Section 4.1.5 that the condenser vacuum is approximately 5 In- Hg equivalent to 2.46 psi (5 In-Hg x .491 psi/In-Hg @32 degrees F) or approximately 12.24 psia. In Section 4.2.12, the "Note" identifies the reactor may produce steam at 160 degrees F with the reactor at negative pressure. A quick check of an ASME Steam Table confirms that at saturation temperature of 160 degrees F, the corresponding pressure is approximately 4.7 psia. Section 4.3.3 requires monitoring of reactor vessel temperature within 15 minutes of pulling any control rods per SVI-B21-T1176 to assure reactor P/T limits are met. In addition to minimum temperatures requirements, the heat up and cool down rates are monitored for compliance with the 100 degrees per hour limit. As identified above, the 100 degrees per hour is an input for the development of the fracture toughness and ultimately the WLIN P/T curves.

Appendix G to 10CFR50 concerns the fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary of light water nuclear power plants. Appendix G requires adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The rules set forth in this guide provide protection from catastrophic failure of the reactor coolant pressure boundary due to uncontrolled crack growth of an undetected flaw. In general, a tensile stress is needed for crack propagation. Pressure within the pressure boundary is one source of tensile load for fracture consideration. For the case under review, a vacuum would tend to have the reverse effect, creating a compressive stress field. Another source of tensile load includes those stresses from reactor main closure flange tensioning. The studs are pre-loaded prior to operation including leak testing to maintain a tight seal. An external pressure would tend to close the flanges and thus release stud preload. The main closure flange assembly and studs play a significant role in the development of PT curves as discussed below.

The materials from which the vessel is constructed require fracture toughness properties to assure that a catastrophic failure of the pressure vessel does not occur. Examination of the P/T curve indicates that the controlling component at the bottom of the curve is the vessel flange region. From GE Report, "Pressure -Temperature Curves for First Energy Corporation, Using the K_{Ic} Methodology", Section 4.1.2 (Reference 1), values of initial test temperature (RT_{NDT}) and Lowest Service Temperature (LST) provide identification of the fracture toughness results for the various vessel materials. The LST for the main closure flange studs is the test temperature + 60 degrees or 70 degrees F. The greatest material RT_{NDT} in the closure flange region is 10 degrees F for the top head and upper shell materials. Thus, the higher of the LST and the RT_{NDT} + 60 degrees F is 70 degrees F and is the bolt-up limit in the closure flange region. A key point here is the limiting temperature of 70 degrees is arrived at from the material properties without consideration of operating pressure.

Review of Attachments 3, 4, 5, and 6 for 22 and 32 EFPY for non-critical and core critical operation indicate that the impact of the WLIN drives the Appendix G curve downward and to the right. The operating pressure applied in the WLIN results in tensile stresses that will not be present during a vacuum. The vessel margin with respect to allowable external pressure is

[NOTE: Only Attachments 4 and 6 for 32 EFPY are provided herein.]

judged adequate to offset any minor redistribution of stresses to achieve static equilibrium within the vessel wall during the vacuum.

The reactor vacuum will not create an unstable structural condition in the reactor. Substantial margin exists between the allowed maximum external pressure and that produced from an external pressure of 14.7 psia with a vessel internal pressure of 0 psia. The fracture toughness properties of the materials and the assumptions with respect to the closure flange remain unchanged as long as the minimum temperature requirement of 70 degrees F is maintained. It is therefore concluded that a reactor vacuum during any of the Technical Specification conditions, Leak Test, Non-Critical, and Core Critical operation is acceptable.

Conclusion

A clarifying note is proposed to be added to the Technical Specification to indicate a reactor vessel vacuum will not adversely affect reactor vessel structural integrity. It is not recommended that the curves be changed to reflect 0 psia as this may cause confusion in the recording of events since plant data is in gauge pressure.

VI. Results, Conclusions and Recommendations

This calculation was prepared from input calculations to a BWROG supported topical report supporting removal of P/T curves from the Technical Specification. The BWROG approach uses the boundary integral formulae to develop the pressure and thermal stress intensity values resulting from the stress concentration effect of the WLIN on the applicable reactor shell material. The GEH methodology from which the non-conservative Technical Specification information was created was based on a finite element analysis of a representative WLIN (DIN 21, Appendix J). In the GEH analysis, K_I values are calculated for each time step in the load case for each node along the crack front. The maximum value of K_I along the crack front is chosen for the T-RT_{NDT} calculations and equations as described above. The BWROG method is slightly more conservative than the GEH submitted information used as input for the non-conservative Technical Specification in 2009. Either method is technically acceptable and has received Regulatory approval. The BWROG method, has recently received Regulatory approval and Perry was one of the bounding geometries (DIN 15) used in the development of the BWROG PTLR. The WLIN curves cause a shift in the current Perry P/T curves. The first recommendation is that upon completion of this calculation, a License Amendment is required to update the current P/T curves to account for the WLIN (Attachments 1, 2, 3, 4, 5, and 6.) [NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]

As a result of the shift to the right of the new curves relative to the non-conservative Technical Specification, a sample of Perry "Leak Test", "Heat up" and "Cooldown" historical results within the last 5 years were reviewed. The results of the review indicated that with the exception of the 32 EFPY Leak Test Curve A (Attachment 2), the new curves are acceptable. Two instances of slight violation of the 22 EFPY leak test Curve A (Attachment 1) were noted. This was attributed to the conservatism of the new analyses supporting the new curves. It is therefore concluded that Perry has operated within acceptable leak test limits. A second recommendation is that Perry is approaching the 32 EFPY threshold, therefore leak test temperature should be increased to satisfy curve requirements in advance of the NRC acceptance of the License Amendment.

Lastly, it was noted above that a vacuum is sometimes applied to the reactor. A vacuum violates literal compliance with the existing Technical Specification Section 3.4.11 Curves as the curves refer to psig for reactor dome pressure. It was discussed above that the vessel has substantial structural margin to the limiting external pressure. It was also discussed that maintaining the minimum temperature of 70 degrees preserves the margin to the fracture toughness requirement of components affected by reactor head tensioning. The third recommendation is that a note be added to the Technical Specifications to allow a vacuum in the vessel during any of the monitored conditions of Leak Testing, Non Core Critical, and Core Critical Operation.

[NOTE: Only Attachments 2, 4 and 6 for 32 EFPY are provided herein.]

Figure 1

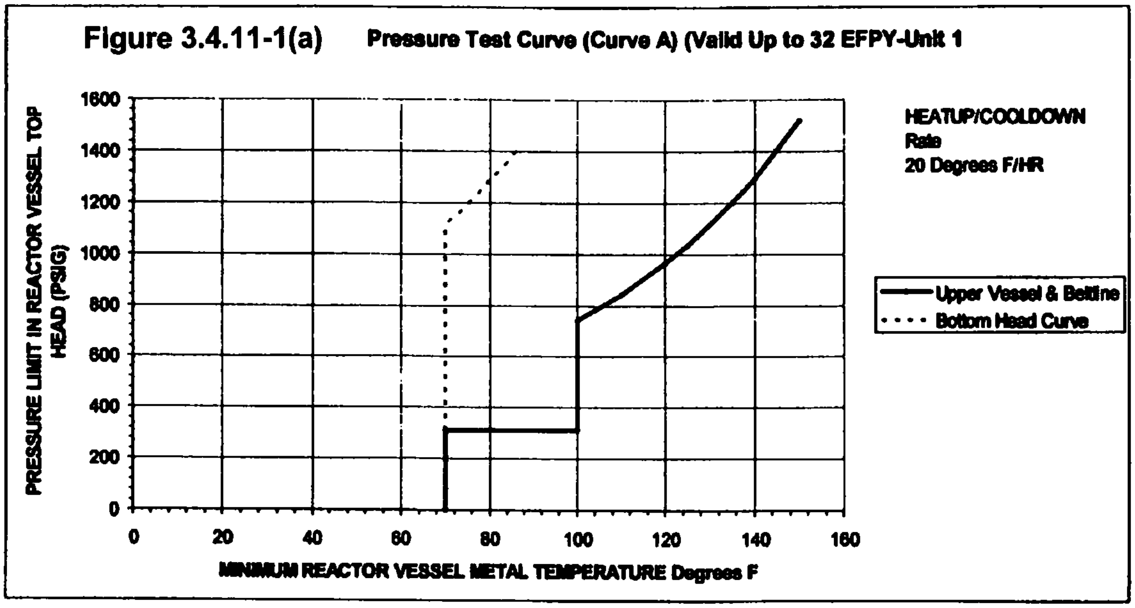


Figure 3.4.11-1(a) Pressure Test Curve (Curve A) (Valid Up to 32 EFPY-Unit 1

Upper Vessel & Beltline

Deg F	PSIG
70	0
70	312.5
100	312.5
100	740.3
110	840.5
120	962.7
125	1033.7
130	1112.1
135	1198.8
140	1294.5
150	1517.4

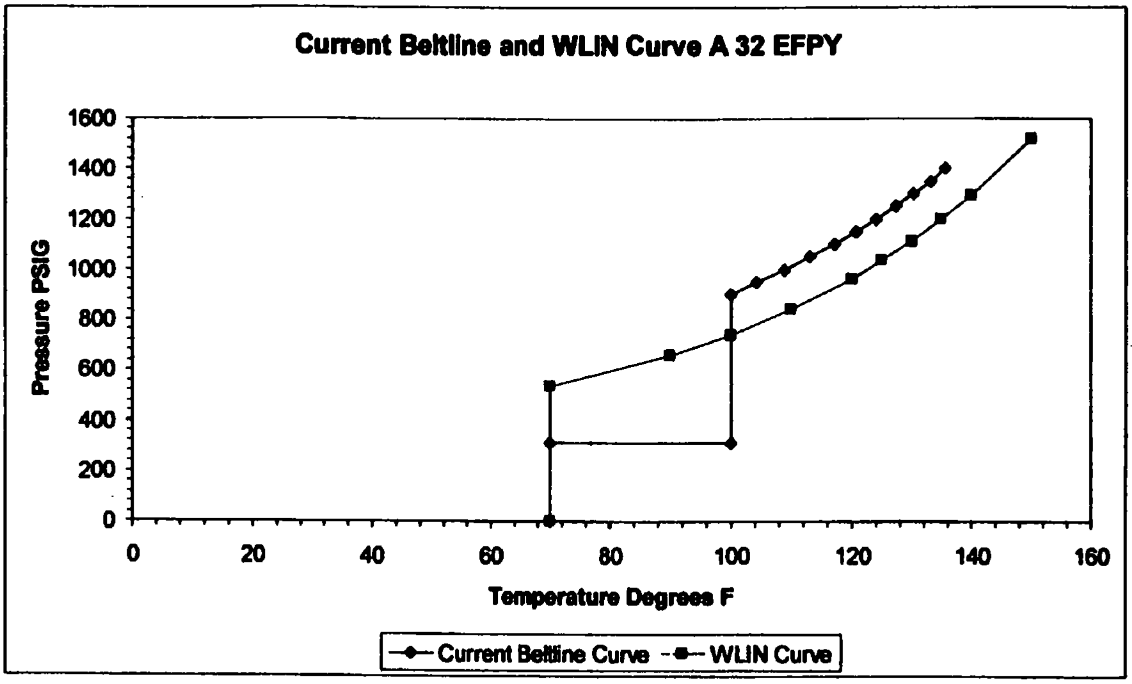
Bottom Head Curve

Deg F	PSIG
70	0
70	1110
72.5	1150
75.6	1200
78.5	1250
81.2	1300
83.8	1350
86.3	1400

Data Source GENE
0000-0000-8763-01

Leak Test Curve A
32 EFPY

Figure 2



Current Beltline Curve

Deg F	PSIG
70	0
70	312.5
100	312.5
100	900
104.4	950
109	1000
113.2	1050
117.1	1100
120.7	1150
124.1	1200
127.3	1250
130.2	1300
133.1	1350
135.7	1400

WLIN Curve

Deg F	PSIG
70	0
70	536.3
90	658.4
100	740.3
110	840.5
120	962.7
125	1033.7
130	1112.1
135	1198.8
140	1294.5
150	1517.4

GENE 0000-0000-8763-01

Figure 1

Figure 3.4.11-1(b) **Non-Nuclear Heatup/Cooldown (Curve B) (Valid Up to 32 EFPY-Unit 1)**

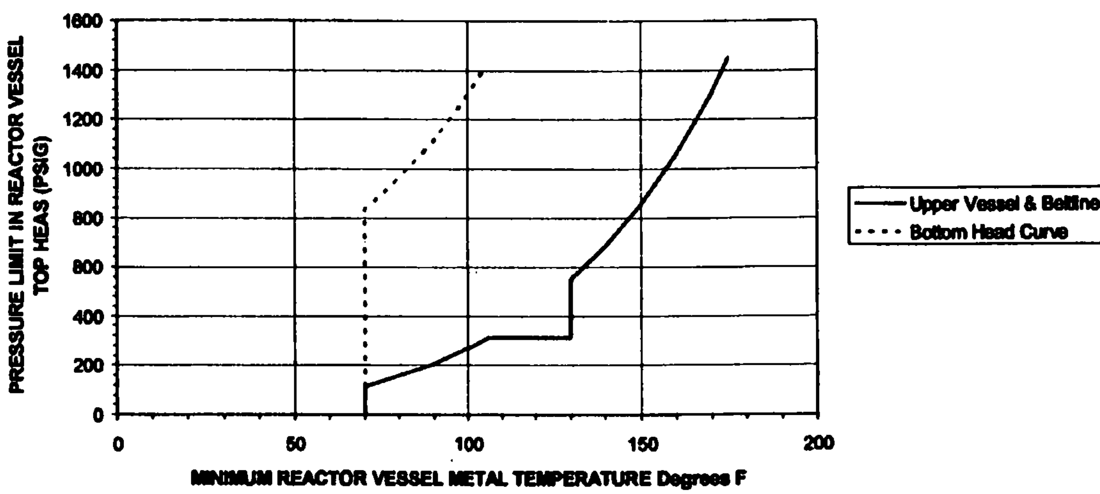


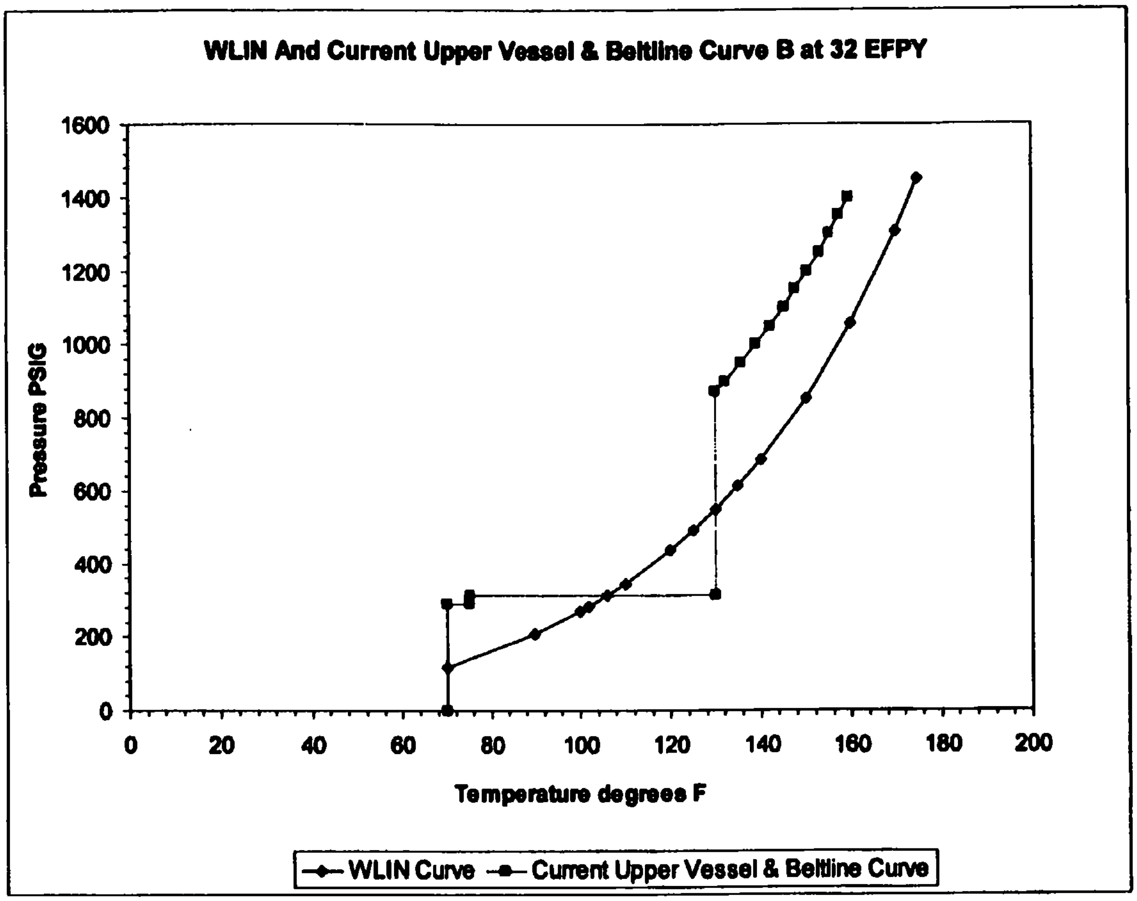
Figure 3.4.11-1(b) Non-Nuclear Heatup/Cooldown (Curve B) (Valid Up to 32 EFPY-Unit 1)

Upper Vessel & Beltline		Bottom Head Curve	
Deg F	PSIG	Deg F	PSIG
70	0		
70	116.43	70	0
90	207.98	70	312.5
100	269.46	70	830
101.773	281.70	71.4	850
105.974	312.50	75.6	900
110	312.50	79.4	950
120	312.50	83	1000
125	312.50	86.3	1050
130	312.50	89.4	1100
130	548.29	92.3	1150
135	613.29	95.1	1200
140	685.12	97.7	1250
150	852.24	100.2	1300
160	1056.36	102.6	1350
170	1305.67	104.8	1400
175	1450.32		

Data GENE 0000-0000-8763-01
32 EFPY Bottom Head Curve B

Non-Nuclear Heat up/ Cool down Curve
32 EFPY

Figure 2



WLIN Curve

Deg F	PSIG
70	0.00
70	116.43
90	207.98
100	269.46
101.773	281.70
105.974	312.50
110	344.55
120	436.27
125	489.48
130	548.29
135	613.29
140	685.12
150	852.24
160	1056.36
170	1305.67
175	1450.32

Upper Vessel & Beltline

Deg F	PSIG
70	0
70	290
75.2	290
75.2	312.5
130	312.5
130	870
132.1	900
135.6	950
139	1000
142.1	1050
145.1	1100
147.8	1150
150.5	1200
153	1250
155.4	1300
157.6	1350
159.8	1400

Data GENE 0000-0000-8763-01
and non conservative Tech Spec

Figure 1

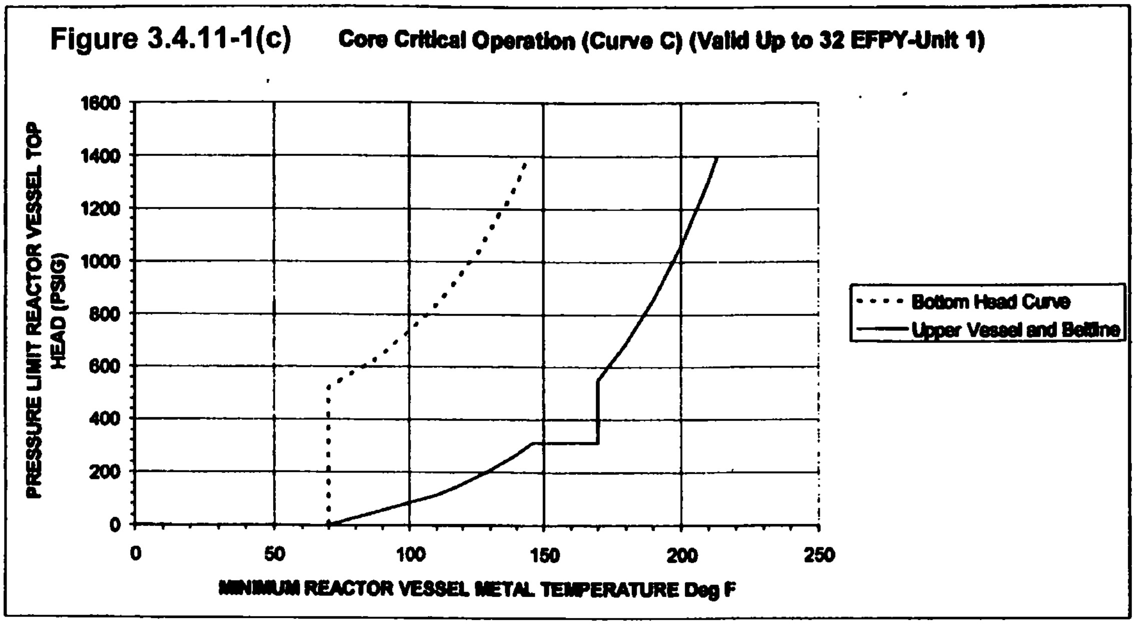


Figure 3.4.11-1(c) Core Critical Operation (Curve C) (Valid Up to 32 EFPY-Unit 1)

Bottom Head Curve

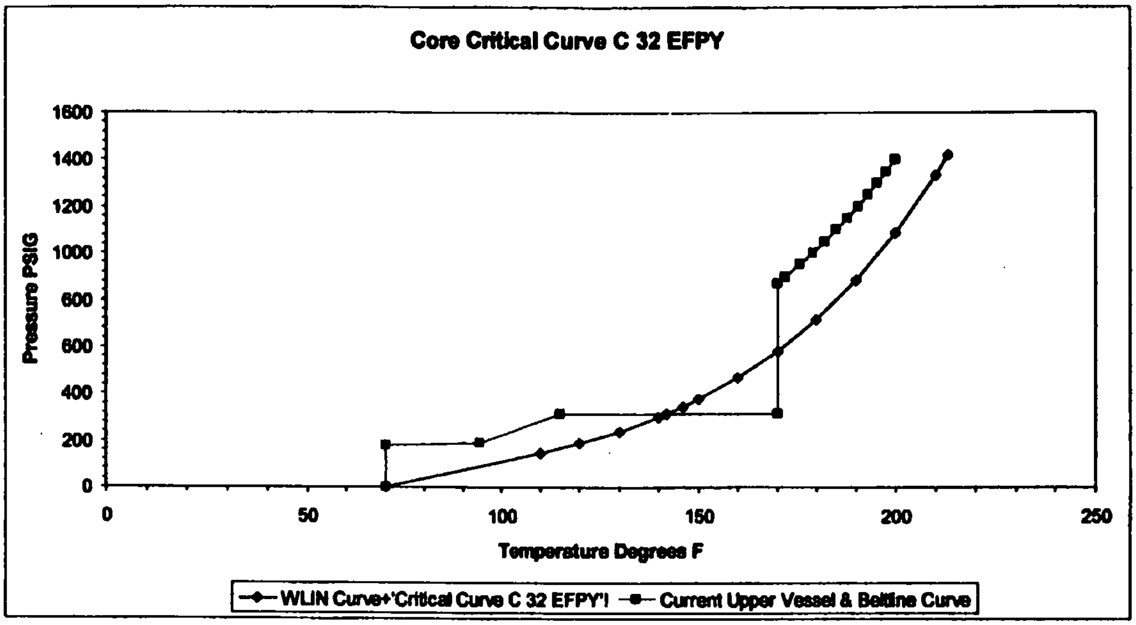
Deg F	PSIG
70	0
70	520
74.9	550
83.2	600
90.2	650
96.4	700
102	750
106.9	800
111.4	850
115.6	900
119.4	950
123	1000
126.3	1050
129.4	1100
132.3	1150
135.1	1200
137.7	1250
140.2	1300
142.6	1350
144.8	1400

Data GENE
0000-0000-8763-01
32 EFPY Curve C

Upper Vessel & Beltline

Deg F	PSIG
70	0.00
110	116.43
120	157.64
130	207.98
140	269.46
145.975	312.50
170	312.50
170	548.29
180	685.12
190	852.24
200	1056.36
210	1305.67
213	1390.72

Figure 2



Existing Curve
Upper Vessel & Beltline

Deg F	PSIG
70	0
70	180
94.3	190
115.2	312.5
170	312.5
170	870
172.1	900
175.6	950
179	1000
182.1	1050
185.1	1100
187.8	1150
190.5	1200
193	1250
195.4	1300
197.6	1350
199.8	1400

GENE 0000-0000-8763-01
and Non-Conservative Technical
Specification

WLIN Curve

Deg F	PSIG
70	0.00
110	116.43
120	157.64
130	207.98
140	269.46
141.7729	281.70
145.975	312.50
150	344.55
160	436.27
170	548.29
180	685.12
190	852.24
200	1056.36
210	1305.67
213	1390.72