PECO NUCLEAR
A Unit of PECO Energy

Nuclear Group Headquarters
200 Exelon Way
Kennett Square, PA 19348

10 CFR 50.90
10 CFR 50.12

November 20, 2000

Docket No. 50-353

License No. NPF-85

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

Subject: Limerick Generating Station, Unit 2
Technical Specifications Change Request No. 00-02-2
Changes to Reactor Pressure Vessel Pressure-Temperature Limits and Request for Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a)

References:

1. Letter from J. A. Hutton (PECO Nuclear) to U.S. Nuclear Regulatory Commission, Limerick Generating Station, Unit 1, Technical Specifications Change Request No. 00-02-1, "Changes to Reactor Pressure Vessel Pressure-Temperature Limits," and Request for Exemption from 10CFR50.60, dated May 15, 2000, as supplemented August 10, 2000.

2. Letter from B. C. Buckley (NRC) to J. A. Hutton (PECO Nuclear), "Limerick Generating Station, Unit 1 - Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G (TAC No. MA8954)," dated September 7, 2000.

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

Dear Sir/Madam:

PECO Energy Company (PECO Energy) is submitting Technical Specifications Change Request No. 00-02-2, in accordance with 10 CFR 50.90, requesting a change to Appendix A of Facility Operating License No. NPF-85 for Limerick Generating Station

APO1

(LGS), Unit 2. The proposed change is to LGS Unit 2 Technical Specifications (TS) Figure 3.4.6.1-1, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure," and associated changes to TS Bases Section 3/4.4.6. In support of this change, PECO Energy is also requesting exemption from 10CFR50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," in accordance with the requirements of 10CFR50.12, "Specific Exemptions."

The proposed change revises the pressure-temperature (P-T) limits by revising the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel (RPV) of Unit 2 to a maximum of 32 Effective Full Power Years (EFPY). The use of 32 EFPY conservatively bounds Unit 2 which is currently at approximately 9.7 EFPY. The proposed change provides a reduction of burden on operators by eliminating the requirement to maintain reactor coolant system within a narrow temperature band less than 212°F during pressure testing and provides potential outage schedule savings.

The proposed changes rely on recently approved American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code methodology for determining allowable P-T limits. This methodology includes the incorporation ASME B&PV Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." ASME Code Case N-640 provides an alternate method for determining the fracture toughness of reactor pressure vessel materials for use in determining P-T Limits. The code case was approved for use by the ASME on February 26, 1999. The use of this Code Case results in a reduction in allowable temperatures, for a given pressure, than would have been required without the use of the Code Case. This results in an increase in operating margin during hydrostatic test performance between the lower end temperature (from the P-T curve) and the upper end temperature (from the TS 3/4.10.8 Special Test Exception limitation of 212° F). Although this Code Case has not yet received USNRC approval for generic usage, Technical Specification Amendment 145 using this Code Case was recently approved by the NRC for LGS, Unit 1 (Reference 3).

This TS change request also includes a request for an exemption in accordance with 10 CFR 50.12 from the requirement of 10 CFR 50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," to comply with 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The requested exemption from 10 CFR 50.60(a) is to allow use of ASME Code Case N-640. A similar exemption was granted for LGS, Unit 1 (Reference 2).

The following attachments are provided in support of this TS change request and request for exemption.

1. Attachment 1 to this letter describes the proposed changes and provides justification for the changes, including the basis for PECO Energy's determination that the proposed changes do not involve a significant hazards consideration and information supporting an Environmental Assessment.

2. Attachment 2 to this letter provides the “marked-up” Technical Specifications pages.

3. Attachment 3 to this letter provides the "camera-ready" Technical Specifications pages.

4. Attachment 4 to this letter provides information supporting the request for exemption from the requirements of 10CFR50.60(a) to allow the use of ASME B&PV Code Case N-640.

5. Attachment 5 to this letter provides the detailed technical basis for the revised P-T limit curve methodology developed by Messers. Warren Bamford and Bruce Bishop of Westinghouse Electric Company which justified the change in ASME B&PV Code, Section XI, Appendix G methodology permitted by ASME Code Case N-640. This information was previously provided to the NRC by PECO Nuclear (Reference 1).

6. Attachment 6 to this letter provides General Electric (GE) Report GE-NE-B11-00836-00-02, "*Pressure-Temperature Curves for PECO Energy Company, Limerick Generating Station, Unit 2*," Rev. 0, dated July, 2000, which GE considers to contain proprietary information. The proprietary information is identified by a vertical bar in the right margin. GE requests that the proprietary information in Attachment 6 be withheld from public disclosure, in accordance with the requirements of 10 CFR 9.17(a)(4), 10 CFR 2.790(a)(4) and 2.790(d)(1). An affidavit supporting this request is provided in the preface to the report.

7. Attachment 7 to this letter provides a *non-proprietary* version of the GE Report (GE-NE-B11-00836-00-02a NP).

8. Attachment 8 to this letter provides additional non-proprietary information that supports information provided by GE in Attachments 6 and 7. This information includes: data to support the fluence value for Limerick Unit 2 specified in the GE report; the location of all of the surveillance capsules represented by this data; the identification of obstructions, if any, between the reactor core and the surveillance capsules identified in the data; and the location of certain welds identified in the GE report.

The attached information is being submitted under affirmation, and the required affidavit is enclosed.

The P-T curves that are currently provided in the LGS, Unit 2, TS are valid until 10 EFPY which currently corresponds to a date of March 27, 2001. If the P-T curves for LGS, Unit 2, are not revised by this date, the P-T limitations will no longer be valid and LGS, Unit 2, will be required to shut down. Therefore, we request approval of this amendment prior to March 27, 2001, to ensure the validity of the P-T limitations and to support continued operation of LGS, Unit 2. If approved, we request that the changes become effective within 30 days of issuance, but no later than March 27, 2001.

If you have any questions, please do not hesitate to contact us.

Sincerely,

J. A. Hutton /FOR

James A. Hutton
Director - Licensing

Attachments

cc:	H. J. Miller, Administrator, Region I, USNRC	w/ Attachments
	A. L. Burritt, USNRC Senior Resident Inspector, LGS	"
	R. R. Janati, PA Bureau of Radiological Protection	w/o Attachments

COMMONWEALTH OF PENNSYLVANIA :

: ss

COUNTY OF CHESTER :

J. J. Hagan, being first duly sworn, deposes and says:

That he is Senior Vice President of PECO Energy Company, the Applicant herein; that he has read the enclosed Technical Specifications Change Request No. 00-02-2, "Changes to Reactor Pressure Vessel Pressure-Temperature Limits," for Limerick Generating Station, Unit 2, Facility Operating License No. NPF-85, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

JJ Hagan

Senior Vice President

Subscribed and sworn to

before me this 20th day

of November, 2000.

Vivia V. Gallimore

Notary Public

Notarial Seal
Vivia V. Gallimore, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires Oct. 6, 2003
Member, Pennsylvania Association of Notaries

ATTACHMENT 1

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

Information Supporting Changes - 7 Pages

Introduction

PECO Energy Company (PECO Energy) is requesting Technical Specifications (TS) changes which will revise the heatup, cooldown, and inservice test Pressure-Temperature (P-T) limitations (TS Figure 3.4.6.1-1) of Limerick Generating Station (LGS), Unit 2, Reactor Pressure Vessel (RPV) to a maximum of 32 Effective Full Power Years (EFPY). Marked-up TS pages indicating the proposed changes are provided in Attachment 2. This Technical Specifications Change Request (TSCR) attachment provides a discussion and description of the proposed changes, a safety assessment, information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

Discussion and Description of the Proposed Changes

Summary of Proposed Changes

PECO Energy is requesting TS changes which will revise the heatup, cooldown, and inservice test P-T limitations (curves) specified in TS Figure 3.4.6.1-1 for the LGS, Unit 2 RPV to a maximum of 32 EFPY. In addition, text changes are proposed to both Limiting Condition for Operation (LCO) 3.4.6.1 and Surveillance Requirement (SR) 4.4.6.1.1 to delete the reference to the A' curve on TS Figure 3.4.6.1-1 since this curve will not be included in the proposed Figure 3.4.6.1-1. An intermediate hydrotest curve (Curve A_{22}) was also added to TS Figure 3.4.6.1-1, which is valid to 22 EFPY.

In addition, PECO Energy is requesting changes to TS Bases Section B 3/4.4.6 to update several RPV material chemistry parameters. The need for these revisions was identified during a Certified Material Test Report (CMTR) data search performed by General Electric Company during development of the proposed P-T curves. The proposed changes result in no impact to either the existing or proposed P-T curves since they are minor changes to non-limiting RPV materials. Throughout this attachment these changes will be referred to as "RPV material property changes." Details of these changes are described below.

TS Bases Table B 3/4.4.6-1:

- Change Heat B 3416-1 ΔRT_{NDT} from +82°F to +48°F
- Change Weld AB ΔRT_{NDT} from +114°F to +58°F

The current values (+82°F and +114°F) are actually not the calculated "ΔRT_{NDT}" but are the values of the "shift" for these materials ("shift" = ΔRT_{NDT} + a margin term). The removal of this margin term leaves the "ΔRT_{NDT}" remaining.

- Change the bottom head dome heat number from C9306-2 to C9245-2
- Change the bottom head dome RT_{NDT} from +26°F to +22°F
- Change the vessel flange RT_{NDT} from -20°F to +10°F

These new RPV material property values are documented in General Electric Report GE-NE-B11-00836-00-02.

Description of the Current P-T Curves

Limiting Conditions for Operation and Surveillance Requirements provide for the reactor pressure vessel metal temperature and pressure to be limited and monitored within the acceptable regions as shown on TS Figure 3.4.6.1-1, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure." The operating limit curves of Figure 3.4.6.1-1 were derived from the fracture toughness requirements of 10 CFR 50, Appendix G, and ASME Code, Section XI, Appendix G.

Bases for the Current P-T Curves

All components of the reactor coolant system are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These cyclic loads are introduced by heatup and cooldown operations, power transients, and reactor trips. The various categories of load cycles used for design purposes are provided in Section 3.9 of the UFSAR. In accordance with Appendix G to 10 CFR 50, the Technical Specifications limit the pressure and temperature changes during heatup and cooldown to be within the design assumptions and the stress limits for cyclic operation. These limits are defined by the P-T curves for heatup, cooldown, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. These curves are used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The P-T curves in the LGS TS were established in accordance with the requirements of 10 CFR 50, Appendix G, and ASME Code, Section XI Appendix G, to assure that brittle fracture of the RPV is prevented. Part of the analysis involved in developing the curves was to account for neutron irradiation embrittlement effects in the core region, or beltline. Regulatory Guide 1.99, Revision 2, was used to predict the shift in RT_{NDT} as a function of neutron fluence in the beltline region and to develop the P-T curves which are in the LGS TS. Regulatory Guide 1.99, Revision 2, provides the general procedures which are acceptable to the NRC staff to be used to calculate the effects of neutron radiation embrittlement.

Pressure testing required by Section XI of the ASME B&PV Code is performed prior to startup after each refueling outage. The minimum temperatures at the required pressures allowed for these tests are determined from the RPV pressure and temperature limits required by TS Figure 3.4.6.1-1, Curve A. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account the anticipated vessel neutron fluence. With increasing RPV neutron fluence, the minimum allowable RPV temperature increases for a given pressure. With this increase in minimum allowable temperature over time, pressure testing will eventually be required with a minimum reactor coolant temperature that exceeds 212° F. Performance of pressure testing at temperatures greater than 212°F is currently prohibited by TS Special Test Exception 3/4.10.8, "Inservice Leak and Hydrostatic Testing," which permits hydrotest performance up to 212°F, while the plant remains in Operational Condition (OPCON) 4. In addition to this TS limitation in performing pressure tests greater than 212°F, performance of pressure testing at such elevated temperatures is not desirable due to:

- decreased personnel safety when conducting inspections at increased coolant temperatures due to steam leak potential as well as increased ambient temperatures,
- decreased leakage detection capability caused by required observation of steam leaks versus water leaks, and
- increased potential to spread contamination in containment.

Need for Revision of the P-T Curves

The P-T curves that are currently available in the LGS TS are valid until 10 EFPY for LGS, Unit 2. These curves require revision prior to the plant reaching this limit to ensure that continuity is maintained regarding the availability of Pressure-Temperature limitations. The revised curves will reflect P-T limitations valid until 32 EFPY. The use of 32 EFPY conservatively bounds Unit 2 which is currently at approximately 9.7 EFPY.

The existing P-T curves also provide a challenge to operations personnel in the performance of hydrostatic tests, due to the small permissible temperature operating window provided between the current A curve, and the 212°F temperature limitation. This window, at 1060 psia is currently approximately 25°F for LGS, Unit 2 (at 10 EFPY). With increased plant fluence, and therefore increased shift of the A curve, the ability to achieve rated hydrostatic test pressure will become increasingly more difficult.

Bases for Revision of the P-T Curves

The proposed changes to the P-T limits have been developed in accordance with the technical requirements of the ASME B&PV Code, Section XI Appendix G, in conjunction with ASME Code Case N-640.

The proposed changes rely on recently approved American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code methodology for determining allowable P-T limits. Several improvements were made to this methodology, including the incorporation of ASME B&PV Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." ASME Code Case N-640 provides an alternate method for determining the fracture toughness of reactor pressure vessel materials for use in determining P-T Limits. The code case was approved for use by the ASME on February 26, 1999. The use of this Code Case results in a reduction in allowable temperatures, for a given pressure, than would have been required without the use of the Code Case. This results in an increase in operating margin during hydrostatic test performance between the lower end temperature (from the P-T curve) and the upper end temperature (from the TS 3/4.10.8 Special Test Exception limitation of 212° F). Although this Code Case has not yet received USNRC approval for generic usage, Technical Specification Amendment 145 using this Code Case was recently approved by the NRC for LGS, Unit 1 (Reference 3).

This TS change request also includes a request for an exemption in accordance with 10 CFR 50.12 from the requirement of 10 CFR 50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," to comply with 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The requested exemption from 10 CFR 50.60(a) is to allow use of ASME Code Case N-640 and is provided in Attachment 4. A similar exemption was granted for LGS, Unit 1 (Reference 2).

Details of the evaluations performed by General Electric Company to calculate the revised P-T limits using this methodology are described in GE Report GE-NE-B11-00836-00-02 which is provided in Attachment 6.

The resultant benefits of this proposed P-T curve revision include the following:

- extension of the valid usage of the TS P-T curves to a maximum of 32 EFPY, which ensures the continuity of valid Pressure-Temperature limitations,
- reduction in the challenges to operations personnel in conducting pressure testing within narrow temperature bands at the required pressure test pressure (between the lower temperature limits provided by the P-T "A" curve and the upper limit by the 212°F Special Test Exception) via expansion of this operating window,
- expansion of operating margin to all P-T curve limitations,
- facilitate the performance of hydrostatic tests at rated pressure, and
- minimize operation of the recirculation pumps at low reactor pressure.

In addition, several minor changes are required to be made to TS Bases Section B 3/4.4.6 to update RPV chemistry parameters. The need for these revisions were identified during a Certified Material Test Report (CMTR) data search performed by General Electric Company during development of the revised P-T curves. The identified changes result in no impact to either the existing P-T curves or the proposed P-T curves since they are minor changes to non-limiting RPV materials.

Safety Assessment

The proposed changes to the P-T limits have been developed in accordance with the technical requirements of the ASME B&PV Code, Section XI, Appendix G as modified by ASME Code Case N-640.

ASME Code Case N-640

The proposed P-T Limits have been developed using the K_{Ic} fracture toughness curve shown on ASME B&PV Code, Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME B&PV Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound of fracture toughness. The other margins inherent with the ASME B&PV Code, Section XI, Appendix G process to determine P-T limit curves remain unchanged.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits is technically more correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the static initiation fracture toughness because the controlled heatup and cooldown process limits the rate at which stress is developed in the RPV wall to rates that are more appropriate for static initiation fracture toughness.

When the K_{Ia} curve was codified in 1974, the initial conservatism of the K_{Ia} curve was necessary due to limited experience and knowledge of RPV material fracture toughness. The conservatism also provided margin thought to be necessary to cover other uncertainties and the postulated material effects of operating loads.

Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that has reduced many of these uncertainties and resolved the postulated material effects from operating loads. Since the original formulation of the K_{Ia} and K_{Ic} curves in 1972, the fracture toughness database has been increased by orders of magnitude, and both K_{Ia} and K_{Ic} ASME B&PV Code, Section XI curves remain lower bound curves. The additional data has significantly reduced the uncertainties associated with material fracture toughness. The added data ensures the ASME B&PV Code, Section XI K_{Ic} curve statistically bounds the data, as presented in Figure 1 of Attachment 5. The new information indicates the lower bound on fracture toughness provided by the K_{Ic} curve is extremely conservative. This lower bound on fracture toughness provides a greater margin of safety beyond that which is required to protect public health and safety from a potential reactor pressure vessel failure.

Details of the evaluations performed to calculate the P-T limits using this methodology are provided in Attachment 6.

<u>Information Supporting a Finding of No Significant Hazards Consideration</u>

We have concluded that the proposed changes to the Limerick Generating Station (LGS), Unit 2 Technical Specifications (TS), which will revise the heatup, cooldown and inservice test Pressure-Temperature (P-T) limitations specified in TS Figure 3.4.6.1-1 for the LGS, Unit 2 Reactor Pressure Vessel (RPV) to a maximum of 32 Effective Full Power Years (EFPY), do not involve a Significant Hazards Consideration. In support of this determination an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. <u>The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   There are no physical changes to the plant being introduced by the proposed changes to the P-T curves. The proposed changes do not modify the reactor coolant pressure boundary, i.e., there are no changes in operating pressure, materials or seismic loading. The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. The proposed P-T curves were generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, in conjunction with ASME Code Case N-640. The proposed P-T curves were established in compliance with the methodology used to calculate the predicted irradiation effects on vessel beltline materials. Usage of these procedures provides compliance with the intent of 10 CFR 50, Appendix G, and provides margins of safety that ensure that failure of the reactor vessel will not occur. The proposed P-T curves prohibit operational conditions in which brittle fracture of reactor vessel materials is possible. Consequently, the primary coolant pressure boundary integrity will be maintained. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes to the P-T curves were generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, and ASME B&PV Code, Section XI, Appendix G, in conjunction with ASME Code Case N-640. Compliance with the proposed P-T curves will ensure that conditions in which brittle fracture of primary coolant pressure boundary materials are possible will be avoided. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Since the integrity of the reactor pressure vessel is ensured, use of the revised P-T curves will continue to provide the same level of protection as was previously reviewed and approved. Further, the proposed changes to the P-T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate nor mitigate any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

   The proposed changes reflect an update of the P-T curves to extend the reactor pressure vessel operating limit to 32 Effective Full Power Years (EFPY). The revised curves are based on the latest ASME guidance. These proposed changes are acceptable because the ASME guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME B&PV Code, Section XI, Appendix G, was approved in 1974. The revised pressure-temperature limits, although less restrictive than the current limits, were established in accordance with current regulations and the latest ASME Code information. Because operation will be within these limits, the reactor vessel materials will continue to behave in a non-brittle manner, thus preserving the original safety design bases. No plant safety limits, set points, or design parameters are adversely affected by the proposed TS changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the TS changes proposed by this TS Change Request because the requested changes to the Limerick Generating Station, Unit 2 TS conform to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The proposed changes will have no impact on the environment. The proposed TS changes do not involve a Significant Hazards Consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite. In addition, the proposed TS changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the Limerick Generating Station, Unit 2 Technical Specifications, and have concluded that they do not involve an unreviewed safety question and they will not endanger the health and safety of the public.

ATTACHMENT 2

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

AFFECTED PAGES
(Mark-ups)

<u>UNIT 2</u>

3/4 4-18
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B 3/4 4-5
B 3/4 4-7

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curves A and ~~A'~~ for hydrostatic or leak testing; (2) curves B ~~and B'~~ for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C ~~and C'~~ for operations with a critical core other than low power PHYSICS TESTS, with: DELETE DELETE DELETE DELETE

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A ~~and A'~~, B ~~and B'~~, or C ~~and C'~~ as applicable, at least once per 30 minutes.
DELETE DELETE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C ~~and curve C'~~ within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup. DELETE

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens located within the surveillance capsules shall be removed and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

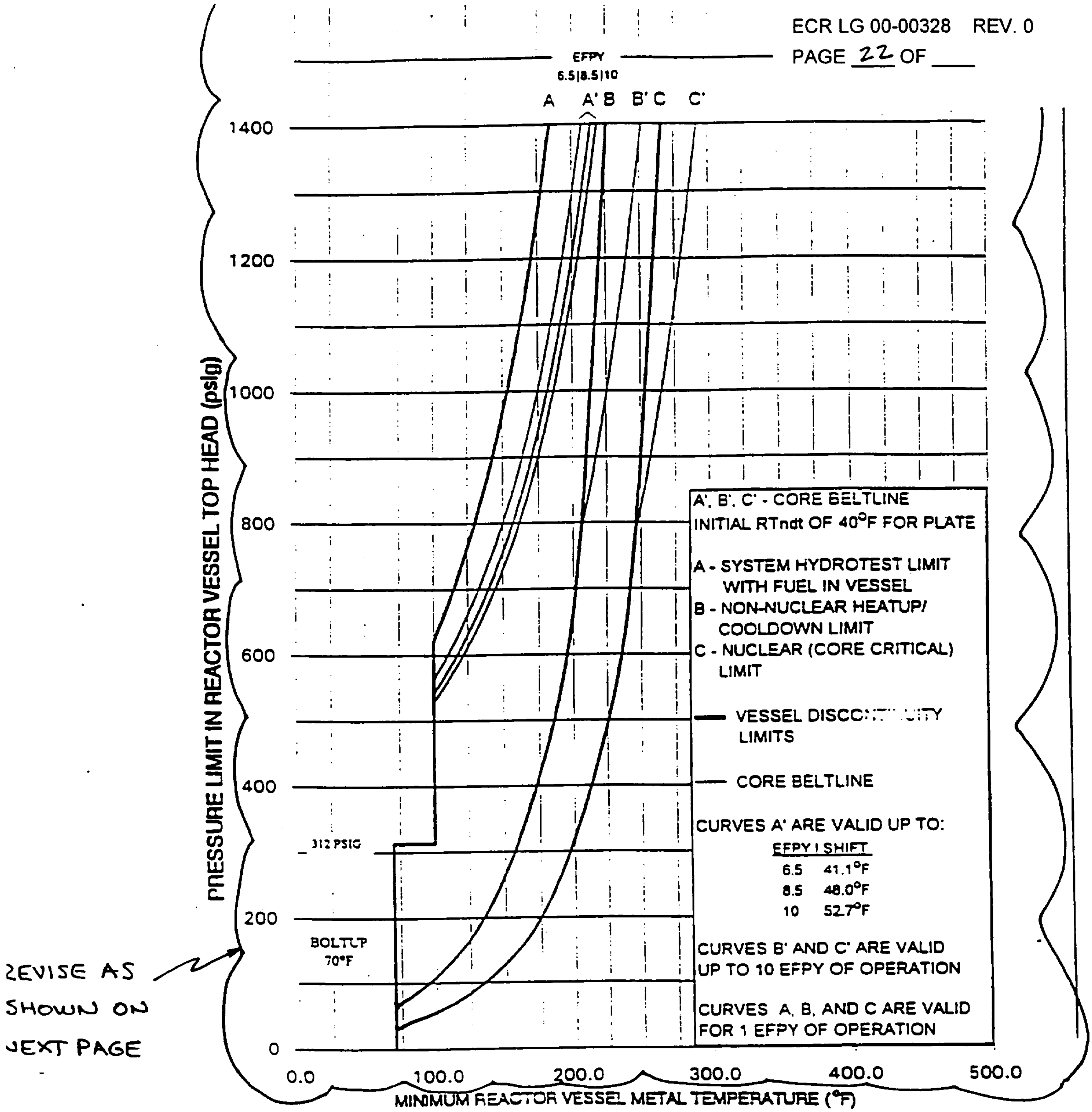
a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

Refer to PORC
Position # 12



MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE
FIGURE 3.4.6.1-1

Refer to PORC
Position # 12

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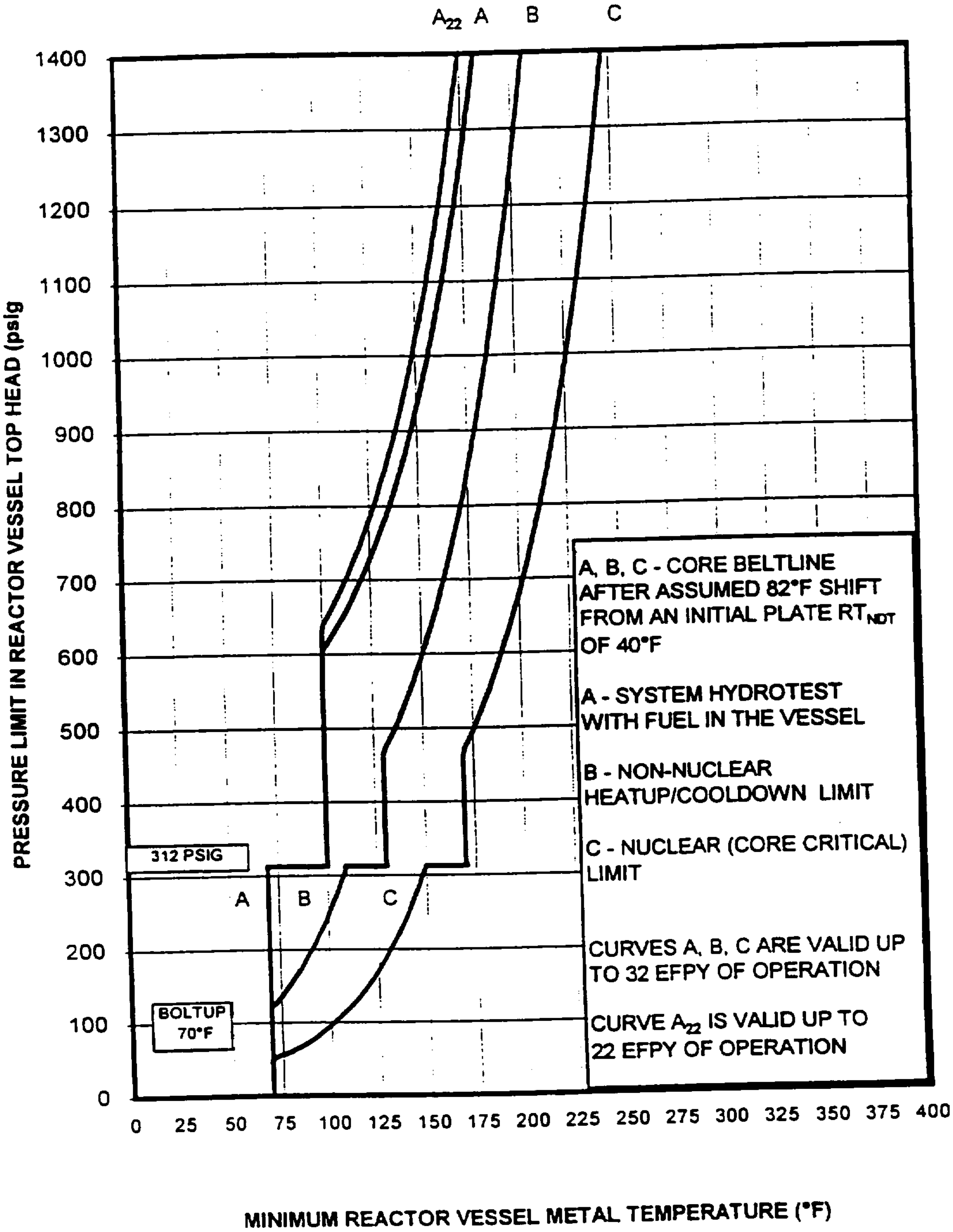

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE
FIGURE 3.4.6.1-1

NEW CURVES AS PER THIS ECR

Refer to PORC Position # 12

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section ~~III~~, XI REPLACE Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A(1), B(1) and C(1), includes an assumed shift in RT_{NDT} for the conditions ~~at~~ 22 REPLACE 32 ~~10~~ EFPY. In addition, an intermediate A(1) DELETE ~~curves~~ curve ~~have~~ has been provided for ~~6.5 and 8.5~~ EFPY. PLACE The A, B and C limit curves are predicted to be bounding for all areas of the RPV until ~~2~~ 32 EFPY. ~~when the beltline materials RT_{NDT} will shift, due to neutron fluence, and the beltline curves will intersect the non-beltline discontinuity curves~~. DELETE

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the charpy specimens and vessel inside radius are essentially identical, the irradiated charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

DELETE The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, ~~and C'~~, and A ~~and A'~~, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

BASES TABLE B 3/4.4.6-1
REACTOR VESSEL TOUGHNESS*

LIMITING BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU (%)	Ni (%)	STARTING RT_{NDT} (°F)	ΔRT_{NDT} **(°F)	MIN. UPPER SHELF (LFT-LBS)	ART (°F)
Plate	SA-533 Gr. B,CL. 1	B 3416-1	.14	.65	+40	~~+82~~ +48	NA	+122
Weld	AB (Field Weld)	640892/ J424B27AE	.09	1.0	-60	~~+114~~ +58	NA	+54

NOTES: * Based on 110% of original power.
** These values are given only for the benefit of calculating the end-of-life (EOL/32 EFPY) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Top Shell Ring	SA 533, Gr. B, CL. 1	C9800-2	-16
Bottom Head Dome	"	~~C9306-2~~ C9245-2	~~+26~~ +22
Bottom Head Torus	"	C9362-2	+20
Top Head Torus	"	C9646-2	-20
Top Head Flange	SA-508, CL. 2	123B300	+10
Vessel Flange	"	2L2058	~~-20~~ +10
Feedwater Nozzle	"	Q2Q29W	0
Weld	Non-Beltline	All	-12
LPCI Nozzle***	SA-508, CL. 2	Q2Q33W	-4
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. Exp. at +10°F

*** The design of the LPCI nozzles results in their experiencing an EOL fluence in excess of 10^{17} N/Cm^2 which predicts an EOL (32 EFPY) RT_{NDT} of +35°F.

ATTACHMENT 3

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

AFFECTED PAGES
(Camera-ready)

<u>UNIT 2</u>

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B 3/4 4-5
B 3/4 4-7

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A, B or C as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens located within the surveillance capsules shall be removed and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

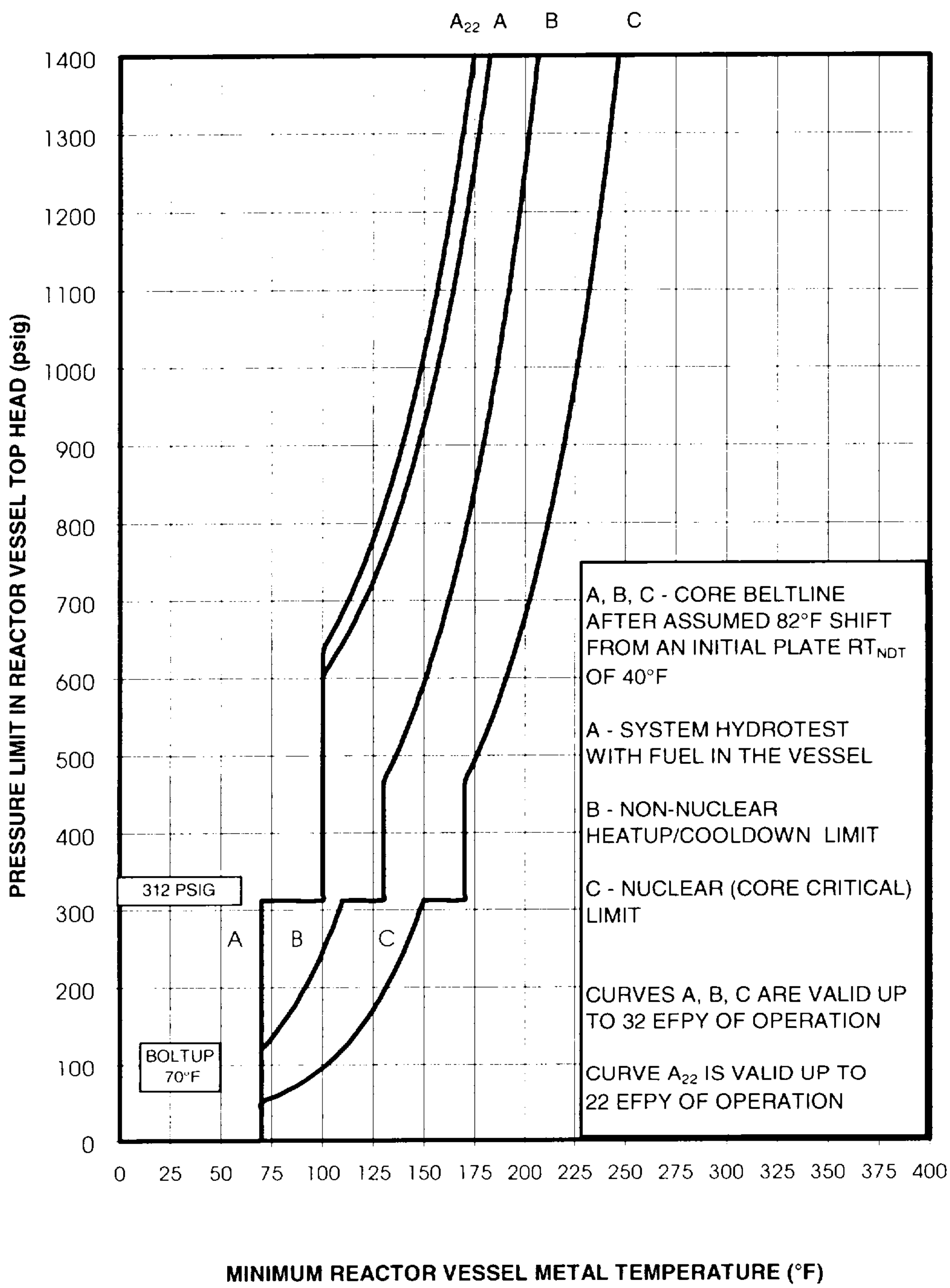
4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE
FIGURE 3.4.6.1-1

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B and C, includes an assumed shift in RT_{NDT} for the conditions at 32 EFPY. In addition, an intermediate A curve has been provided for 22 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the charpy specimens and vessel inside radius are essentially identical, the irradiated charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

BASES TABLE B 3/4.4.6-1
REACTOR VESSEL TOUGHNESS*

LIMITING BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU (%)	Ni (%)	STARTING RT_{NDT} (°F)	ΔRT_{NDT} **(°F)	MIN.UPPER SHELF (LFT-LBS)	ART (°F)
Plate	SA-533 Gr. B,CL. 1	B 3416-1	.14	.65	+40	+48	NA	+122
Weld	AB (Field Weld)	640892/ J424B27AE	.09	1.0	-60	+58	NA	+54

NOTES: * Based on 110% of original power.
** These values are given only for the benefit of calculating the end-of-life (EOL/32 EFPY) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Top Shell Ring	SA 533, Gr. B, CL. 1	C9800-2	-16
Bottom Head Dome	"	C9245-2	+22
Bottom Head Torus	"	C9362-2	+28
Top Head Torus	"	C9646-2	-20
Top Head Flange	SA-508, CL. 2	123B300	+10
Vessel Flange	"	2L2058	+10
Feedwater Nozzle	"	Q2Q29W	0
Weld	Non-Beltline	All	-12
LPCI Nozzle***	SA-508, CL. 2	Q2Q33W	-4
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. Exp. at +10°F

*** The design of the LPCI nozzles results in their experiencing an EOL fluence in excess of 10^{17} N/Cm^2 which predicts an EOL (32 EFPY) RT_{NDT} of +35°F.

ATTACHMENT 4

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

Information Supporting a Request for Exemption
from the Requirements of 10 CFR 50.60(a) - 3 Pages

Request for Exemption from 10CFR50.60(a)

In accordance with 10 CFR 50.12, "Specific exemptions," PECO Energy Company is requesting an exemption from the requirements of 10 CFR 50.60(a) "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b.

Justification for Use of Code Case N-640

<u>10 CFR 50.12(a) Requirements</u>

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME B&PV XI, Appendix G to determine the pressure-temperature limits for the reactor pressure vessel meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the commission may grant an exemption from requirements contained in 10 CFR 50 provided that the following is met:

1. <u>The requested exemption is authorized by law.</u> No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. <u>The requested exemption does not present an undue risk to the public health and safety.</u> The revised pressure-temperature (P-T) limits being proposed for Limerick Generating Station, Unit 2, rely in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining P-T limit curves remain unchanged.

   Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heat-up and cooldown process of a reactor pressure vessel.

   Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of reactor pressure vessel materials and their fracture response to applied loads. As described in Attachment 5, the additional knowledge demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential reactor pressure vessel failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential reactor pressure vessel failure and does not present an undue risk to public health and safety.

P-T curves based on the K_{Ic} fracture toughness limits will enhance overall plant safety by opening the pressure-temperature operating window. The two primary safety benefits that would be realized during the pressure test are a reduction in the challenges to operators in maintaining a high temperature in a limited operating window and personnel safety while conducting inspections in primary containment at elevated temperatures with no decrease to the margin of safety.

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

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

   (a)(2) (ii) – demonstrates the underlying purpose of the regulation will continue to be achieved;

   (a)(2) (iii) – would result in undue hardship or other costs that are significant if the regulation is enforced and;

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

   10CFR 50.12(a)(2)(ii): ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure vessel and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfy the underlying requirement that:

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

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

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

10CFR50.12(a)(2)(iii): The Reactor Pressure Vessel pressure-temperature operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of Limerick Generating Station, Unit 2, with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the pressure-temperature operating window. This restriction challenges the operations staff during pressure tests to maintain a high temperature within a limited operating window.

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

10CFR50.12(a)(2)(v): The requested exemption provides only temporary relief from the applicable regulation and Limerick Generating Station, Unit 2, has made a good faith effort to comply with the regulation. We request the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry.

Code Case N-640, Conclusion for Exemption Acceptability: Compliance with the specified requirement of 10 CFR 50.60(a) would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the lower bound fracture toughness used in ASME B&PV Code, Section XI, Appendix G, in the determination of reactor coolant system pressure-temperature limits. This proposed alternative is acceptable because the ASME Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for Limerick Generating Station, Unit 2, will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor pressure vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, reactor coolant system pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in TS Section 3.4.6.1. Therefore, this exemption request does not present an undue risk to the public health and safety.

ATTACHMENT 5

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

Technical Basis for Revised Pressure-Temperature Limit Curve Methodology
Developed by Messers. Warren Bamford and Bruce Bishop
Previously submitted by PECO Nuclear (Reference 1) - 22 Pages

TECHNICAL BASIS FOR REVISED P-T LIMIT CURVE METHODOLOGY

by Messers. Warren Bamford and Bruce Bishop

Abstract

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature limits, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate safety margins for nine different parameters; one of which is a lower bound fracture toughness curve.

There are two lower bound fracture toughness curves available in Section XI, K_{Ia}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{Ic}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{Ia} to K_{Ic}. The other margins involved with the process remain unchanged.

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By opening up the operating window relative to the pump seal requirements, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin which exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease.

Introduction

The startup and shutdown process, as well as pressure testing, for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate four specific safety margins:

1. Large flaw, ¼ thickness
2. Safety factor = 2 on pressure stress for startup and shutdown
3. Lower bound fracture toughness
4. Upper bound adjusted reference temperature (RT_{NDT})

Although the above four safety margins were originally included in the methodology used to develop P-T Limit Curves and hydrotest temperatures, it is important to mention that several sources of stress were not considered in the original methodology. The two key factors here are the weld residual stresses, and stresses which result from the clad-base metal differential thermal expansion. Furthermore, the method as originally proposed assumed that the maximum value of the stress intensity factor occurred at the deepest point of the flaw. These elements were all considered in the sample problems which were carried out, so their effects on the margins could be assessed.

There are two lower bound fracture toughness curves available in Section XI, K_{Ia}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{Ic}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{Ia} to K_{Ic}. The other margins involved with the process remain unchanged. There are a number of reasons why the limiting toughness in the Appendix G pressure-temperature limits should be changed from K_{Ia} to K_{Ic}.

Use of K_{Ic} is More Technically Correct

The heatup and cooldown process is a very slow one, with the fastest rate allowed being 100° per hour. The rate of change of pressure and temperature is often constant, so the rate of change in stress is essentially constant. Both the slow heatup and cooldown and the pressure testing are essentially static processes. In fact, all operating transients (levels A, B, C and D) correspond to static loadings, with regard to fracture toughness.

The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{Ia} should be used for the reactor pressure vessel is when a crack is running. This might happen during a PTS transient event, but not during heatup or cooldown. Therefore, use of the static toughness K_{Ic} lower bound toughness would be more technically correct for development of P-T limit curves.

Use of Historically Large Margin No Longer Necessary

In 1974, when the Appendix G methodology was first codified, the use of K_{Ia} (K_{Ir} in the terminology of the time) to provide additional margin was thought to be necessary to cover uncertainties and a number of postulated but unquantified effects. Almost 25 years later, significantly more is known about these uncertainties and effects.

Flaw Size

With regard to flaw indications in reactor vessels, there have been no indications found at the inside surface of any operating reactor in the core region which exceed the acceptance standards of Section XI, in the entire 28 year history of Section XI. This is a particularly impressive conclusion when considering that core region inspections have been required to concentrate on the inner surface and near inner surface region since the implementation of Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations". Flaws have been found, but all have been qualified as buried, or embedded.

There are a number of reasons why no surface flaws exist, and these are related to the fabrication and inspection practices for vessels. For the base metal and full penetration welds, a full volumetric examination and surface exam is required before cladding is applied, and these exams are repeated after cladding.

Further confirmation of the lack of any surface indications has recently been obtained by the destructive examination of portions of several commercial reactor vessels, for example the Midland vessel and the PVRUF vessel.

Fracture Toughness

Since the original formulation of the K_{Ia} and K_{Ic} curves, in 1972, the fracture toughness database has increased by more than an order of magnitude, and both K_{Ia} and K_{Ic} remain lower bound curves, as shown for example in Figure 1 for K_{Ic}[1] compared to Figure 2, which is the original database[2]. In addition, the temperature range over which the data have been obtained has been extended, to both higher and lower temperatures than the original data base.

In can be seen from Figure 1 that there are a few data points which fall just below the curve. Consideration of these points, as well as the (over 1500) points above the curve, leads to the conclusion that the K_{Ic} curve is a lower bound for a large percentage of the data. An example set of carefully screened data in the extreme range of lower temperatures is shown in Figure 3, from Reference [3].

Local Brittle Zones

A third argument for the use of K_{Ia} in the original version of Appendix G was based upon the concern that there could be a small, local brittle zone in the weld or heat-affected-zone of the base material that could pop-in and produce a dynamically moving cleavage crack. Therefore, the toughness property used to assess the moving crack should be related to dynamic or crack arrest conditions, especially for a ferritic pressure vessel steel showing distinct temperature and loading-rate (strain-rate) dependence. The dynamic crack should arrest at a ¼-T size, and any re-initiation should consider the effects of a minimum toughness associated with dynamic loading. This argument provided a rationale for assuming a ¼-T postulated flaw size and a lower bound fracture toughness curve considering dynamic and crack arrest loading. The K_{Ir} curve in Appendix G of Section III, and the equivalent K_{Ia} curve in Appendix A and Appendix G of Section XI provide this lower bound curve for high-rate loading (above any realistic rates in reactor pressure vessels during any accident condition) and crack arrest conditions. This argument, of course, relies upon the existence of a local brittle zone.

After over 30 years of research on reactor pressure vessel steels fabricated under tight controls, micro-cleavage pop-in has not been found to be significant. This means that researchers have not produced catastrophic failure of a vessel, component, or even a fracture toughness test specimen in the transition temperature regime. The quality of quenched, tempered, and stress-relieved nuclear reactor pressure vessel steels, that typically have a lower bainitic microstructure, is such that there may not be any local brittle zones that can be identified. Testing of some test specimens at ORNL [4] has shown some evidence of early pop-ins for some simulated production weld metals, but the level of fracture toughness for these possible early initiations is within the data scatter for other ASTM-defined fracture toughness values (K_{Ia} and/or K_{Ic}). Therefore, it is time to remove the conservatism associated with this postulated condition and use the ASME Code lower bound K_{IC} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument can be made that the proposed change will increase plant safety and reduce personnel exposure for both PWRs and BWRs.

Impact on PWRs:

By opening up the operating window relative to the pump seal requirements, as shown schematically in Figure 4, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

The proposed change also reduces the need for lock-out of the HPSI systems, which improves personnel and plant safety and reduces the potential for a radioactive release. Finally, challenges to the plant low temperature overpressure protection system (LTOP) and potential problems with reseating the valves would also be reduced.

Impact on BWRs:

The primary impact on the BWR will be a reduction in the pressure test temperature. BWRs use pump heat to reach the required pressure test temperatures. Several BWR plants are required to perform the pressure test at temperatures over 212°F under the current Appendix G criteria. The high test temperature poses several concerns: (i) pump cavitation and seal degradation, (ii) primary containment isolation is required and ECCS/safety systems have to be operational at temperatures in excess of 212°F, (iii) leak detection is difficult and more dangerous since the resulting leakage is steam and poses safety hazards of burns and exposure to personnel. The reduced test temperature eliminates these safety issues without reducing overall fracture margin.

Reactor Vessel Fracture Margins

It has long been known that the P-T limit curve methodology is very conservative[5,6]. Changing the reference toughness to K_{Ic} will maintain a very high margin, as illustrated in Figure 5, for a pressurized water reactor. Similar results are shown for a BWR hydrotest in Figure 6. These figures show a series of P-T curves developed for the same plant (either a BWR or a PWR), but with different assumptions concerning flaw size, safety margin and fracture toughness.

Results were obtained for a sample problem which was solved by several members of the Section XI working group on Operating Plant Criteria, for both PWR and BWR plants. The problem statement details are provided in Appendix A (separate problems for the PWR and BWR). The sample problem requires development of an operating P-T cooldown curve or the pressure test for an irradiated vessel. Two P-T curves were required, one using K_{Ia} and the second using K_{Ic}. In both cases the quarter thickness flaw was used, along with the appropriate safety factor on pressure.

To determine the margins (pressure ratios) that are included in these curves, a reference P-T curve was developed, using a best estimate (mean) K_{Ic} curve, and no safety factor on stress, along with a flaw depth of one inch. These analyses all considered the K_I/K_{Ic} ratio at all points on the crack front located in the ferritic steel. Typical results are shown in Table 1 for a PWR. Comparing the reference or best estimate curve with the two P-T curves calculated using code requirements, we see that there is a large margin on the allowable pressure, whether one uses K_{Ia} or K_{Ic} limits in Appendix G.

For PWRs, another important contribution to the margin, which cannot be quantified, is the low temperature overpressure protection system (LTOP) which is operational in the low temperature range. The margins increase significantly for higher temperatures, as seen in Figure 5.

<u>Impact of the Change on P-T Curves</u>

To show the effect that the proposed change would produce, a series of P-T limit curves were produced for a typical plant. These curves were produced using identical input information, with one curve using K_{Ia} and the other using the proposed new approach, with K_{Ic}. Since the limiting conditions for the PWR (cooldown) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs.

The results are shown in Figure 7 for a typical PWR cool-down transient.

<u>Summary and Conclusions</u>

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin that exists with the revised methodology is still very large.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures, need for HPSI systems lock-out, LTOP system challenges and/or fuel margin problems, and personnel hazards and exposure will all decrease.

<u>References</u>

1. VanderSluys, W.A. and Yoon, K.K., "Transition Temperature Range Fracture Toughness in Ferritic Steels and Reference Temperature of ASTM", prepared for PVRC and BWOG, BAW 2318, Framatome Technologies, April 1998.

2. Marston, T.U., "Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A", EPRI Special Report NP-719-SR, August 1978.

3. Nanstad, R.K. and Keeney, J.A., and McCabe, D.E., "Preliminary Review of the Bases for the K_{Ic} Curve in the ASME Code", Oak Ridge National Laboratory Report ORNL/NRC/LTR-93/15, July 12, 1993.

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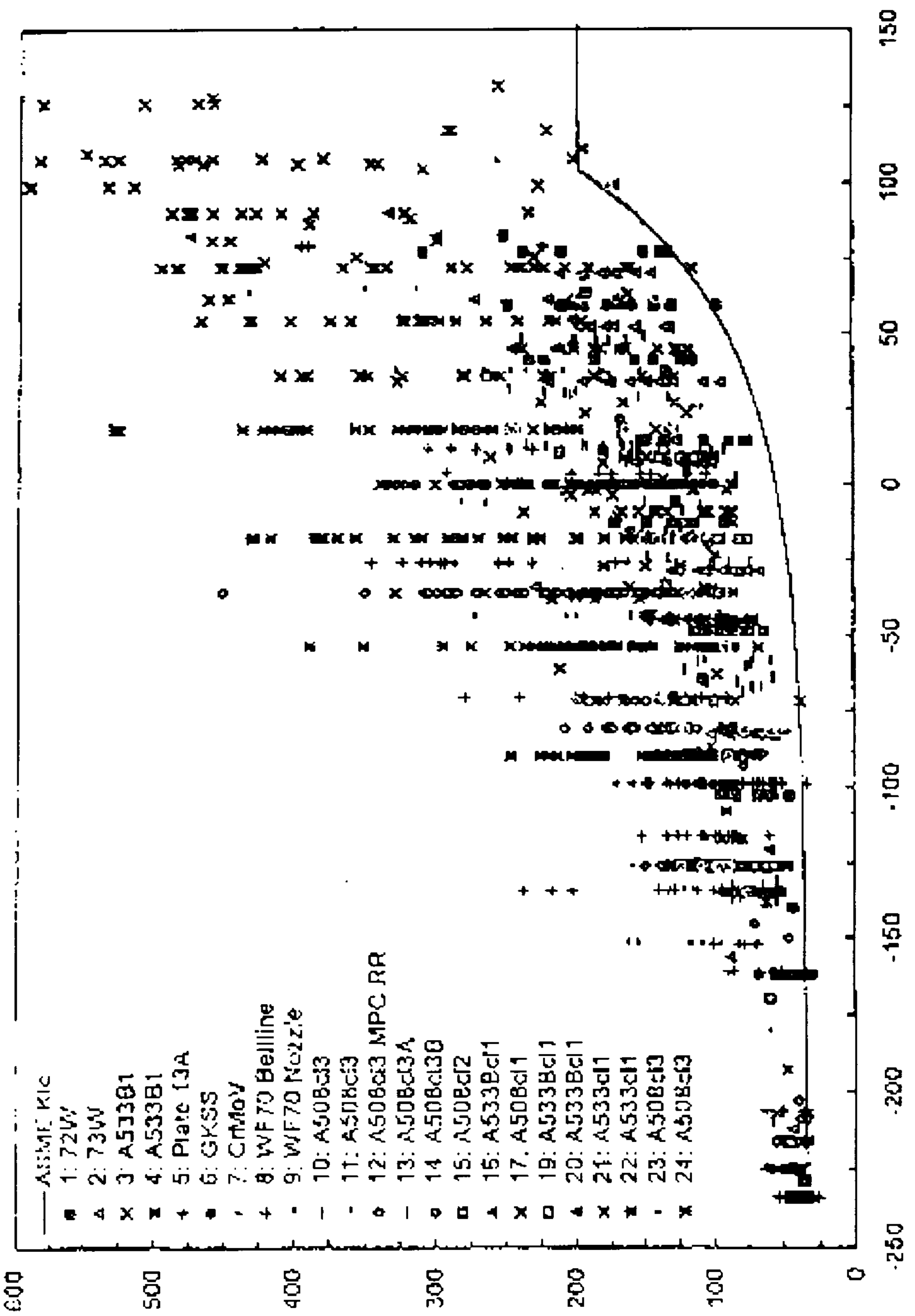
5. Chirigos, J.N. and Meyer, T.A., "Influence of Material Property Variations on the Assessment of Structural Integrity of Nuclear Components", ASTM Journal of Testing and Evaluation, Vol. 6, No. 5, Sept. 1978, pp 289-295.

6. White Paper on Reactor Vessel Integrity Requirements for Level A and B conditions, prepared by Section XI Task Group on R.V. Integrity Requirements, EPRI TR-100251, January 1993.

Table 1
Summary of Allowable Pressures for
20 Degree/hour Cooldown of Axial Flaw
at 70 Degrees F and RT_{PTS} of 270 F
(Typical PWR Plant)

Type of Evaluation	Allowable Pressure* (psi)	Pressure Ratio
Appendix G with t/4 flaw and K_{Ia} Limit	420	1.00
Appendix G with t/4 flaw and K_{Ic} Limit	530	1.26
Reference Case: 1 inch flaw For pressure, thermal, Residual and cladding loads	1520	3.61
Reference Case: 1 inch flaw for pressure, thermal and residual loads	1845	4.38
Reference Case: 1 inch flaw for pressure and thermal loading only	2305	5.48

* Note: Comparable values of allowable pressure were calculated by the ASME Section XI Operating Plant Working Group Members from Westinghouse, Framatome Technologies and Oak Ridge National Laboratory



Toughness, ksi √in

Figure 1. Static Fracture Toughness Data (K_{JC}) Now Available, Compared to K_{IC} [1]

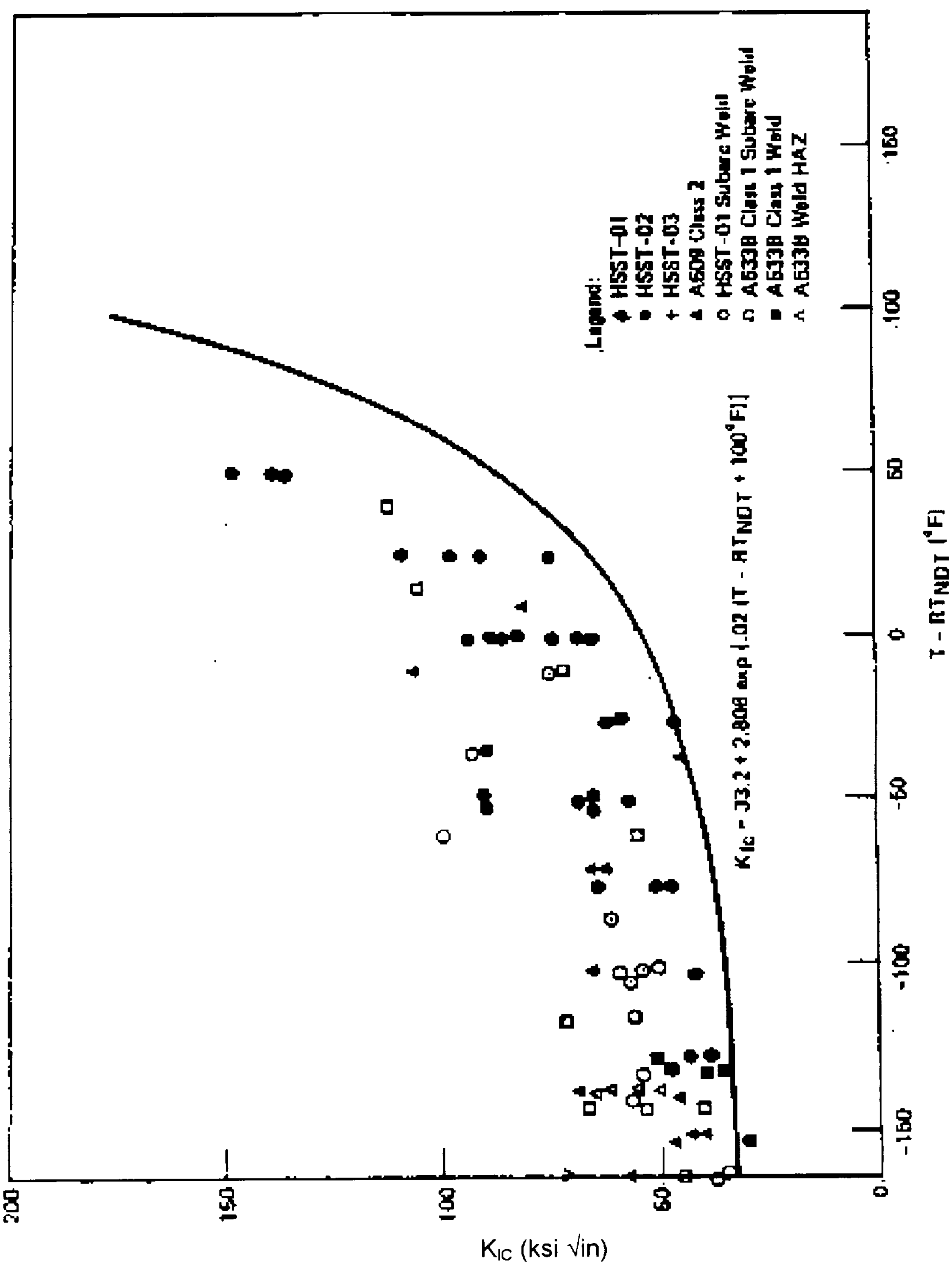
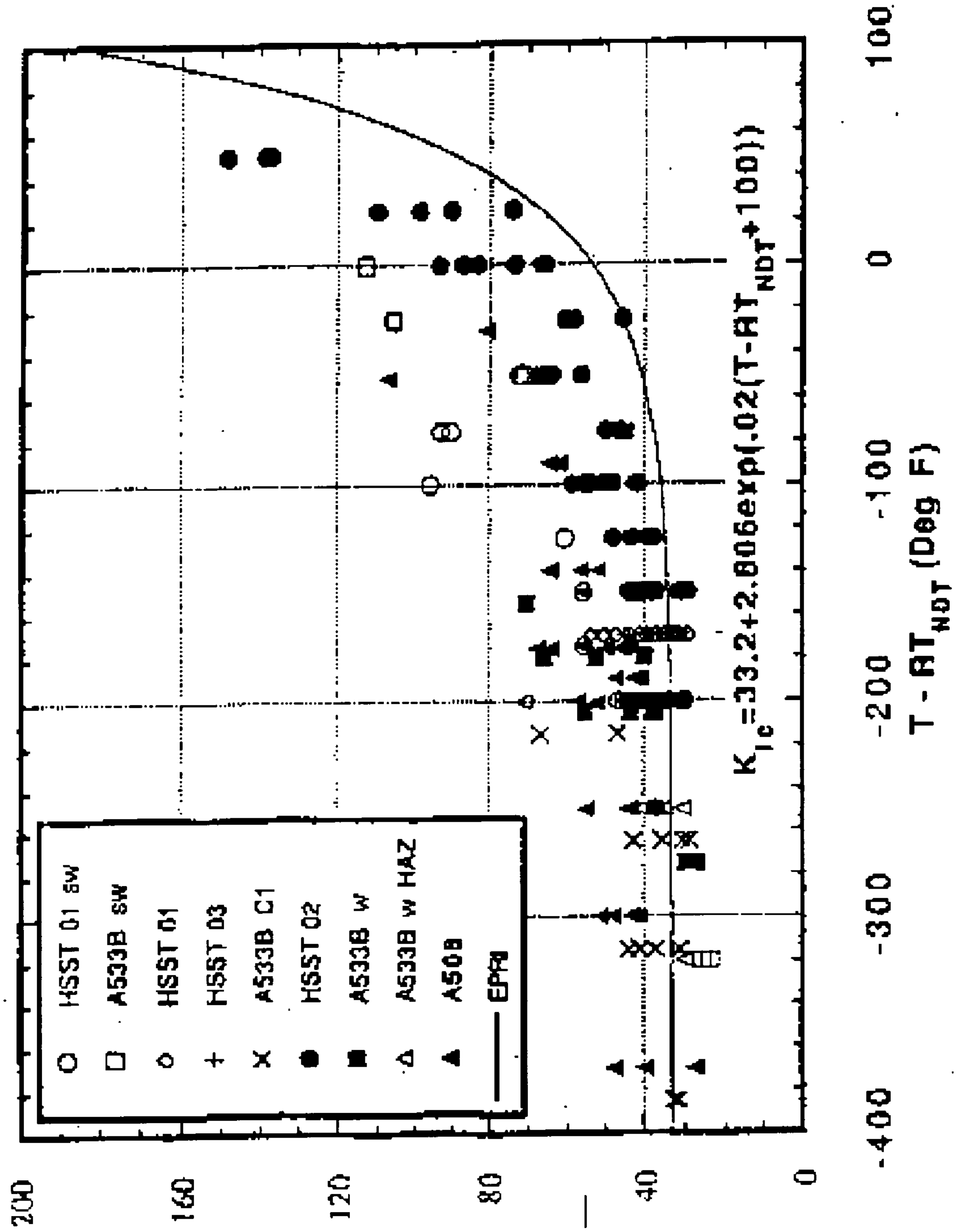


Figure 2. Original K_{IC} Reference Toughness Curve, with Supporting Data [2]



K_{IC} (ksi**.5)

Figure 3. K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

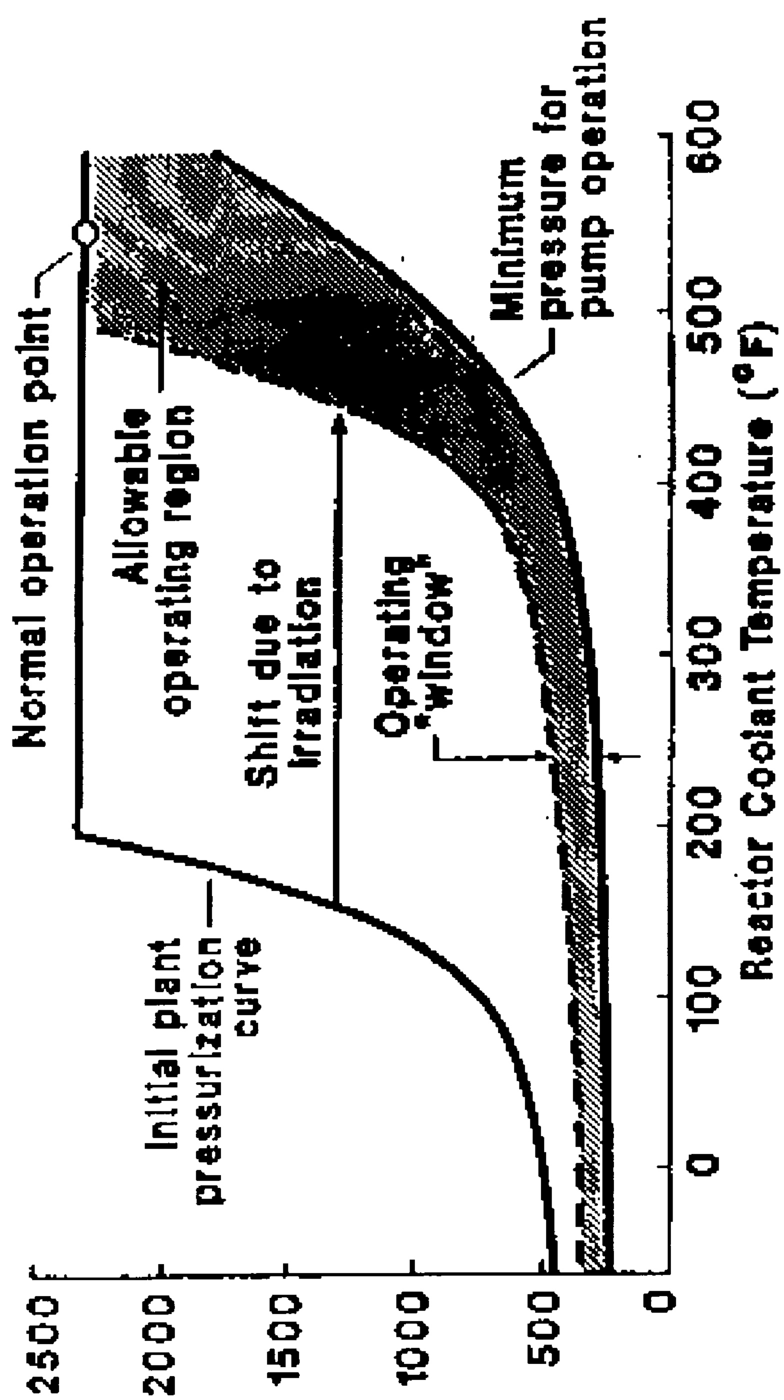


Figure 4. Operating Window From P-T Limit Curves [4]

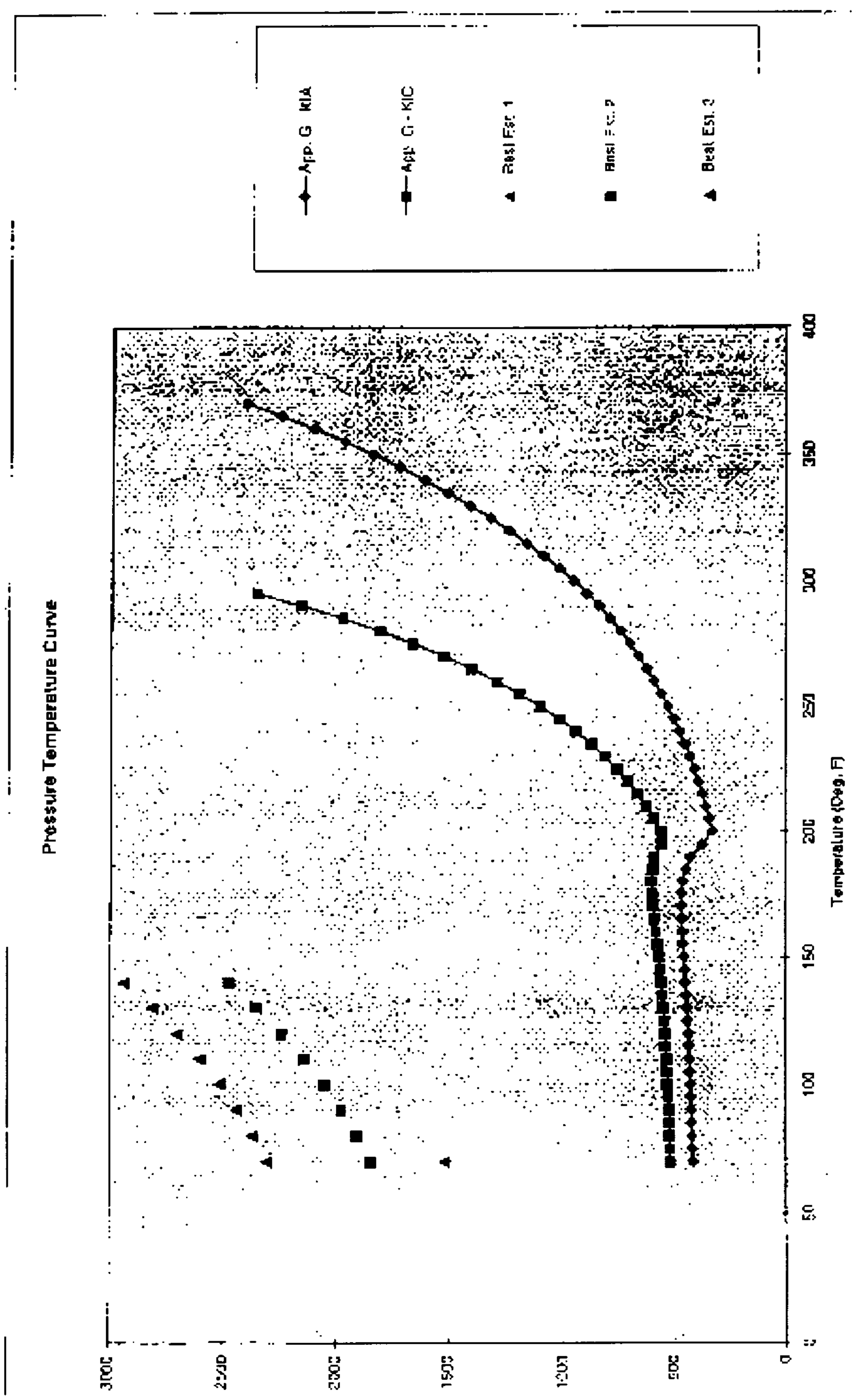


Figure 5. P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

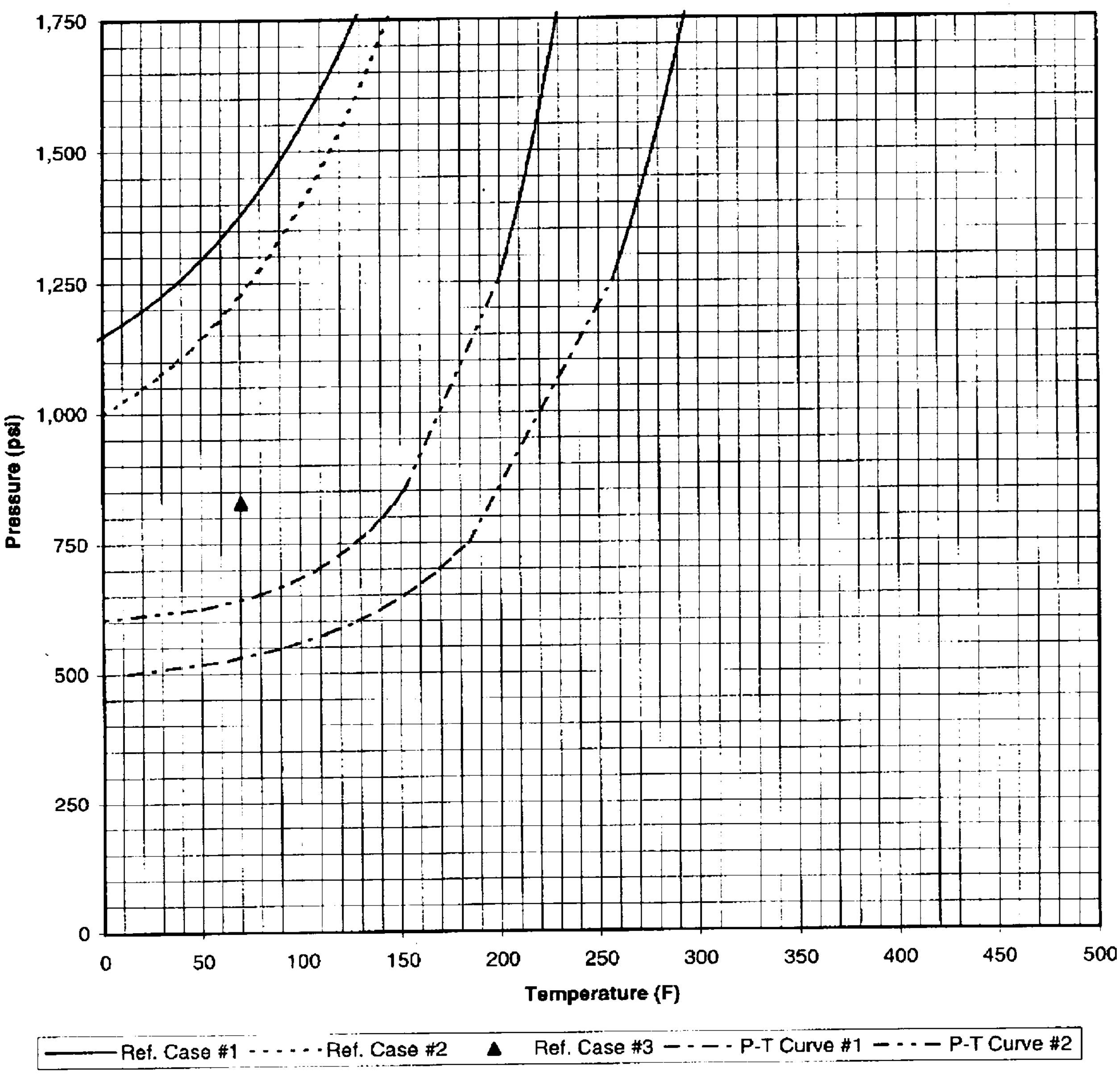


Figure 6. P-T Limit Curves Illustrating Deterministic Safety Factors
for a BWR Reactor Vessel

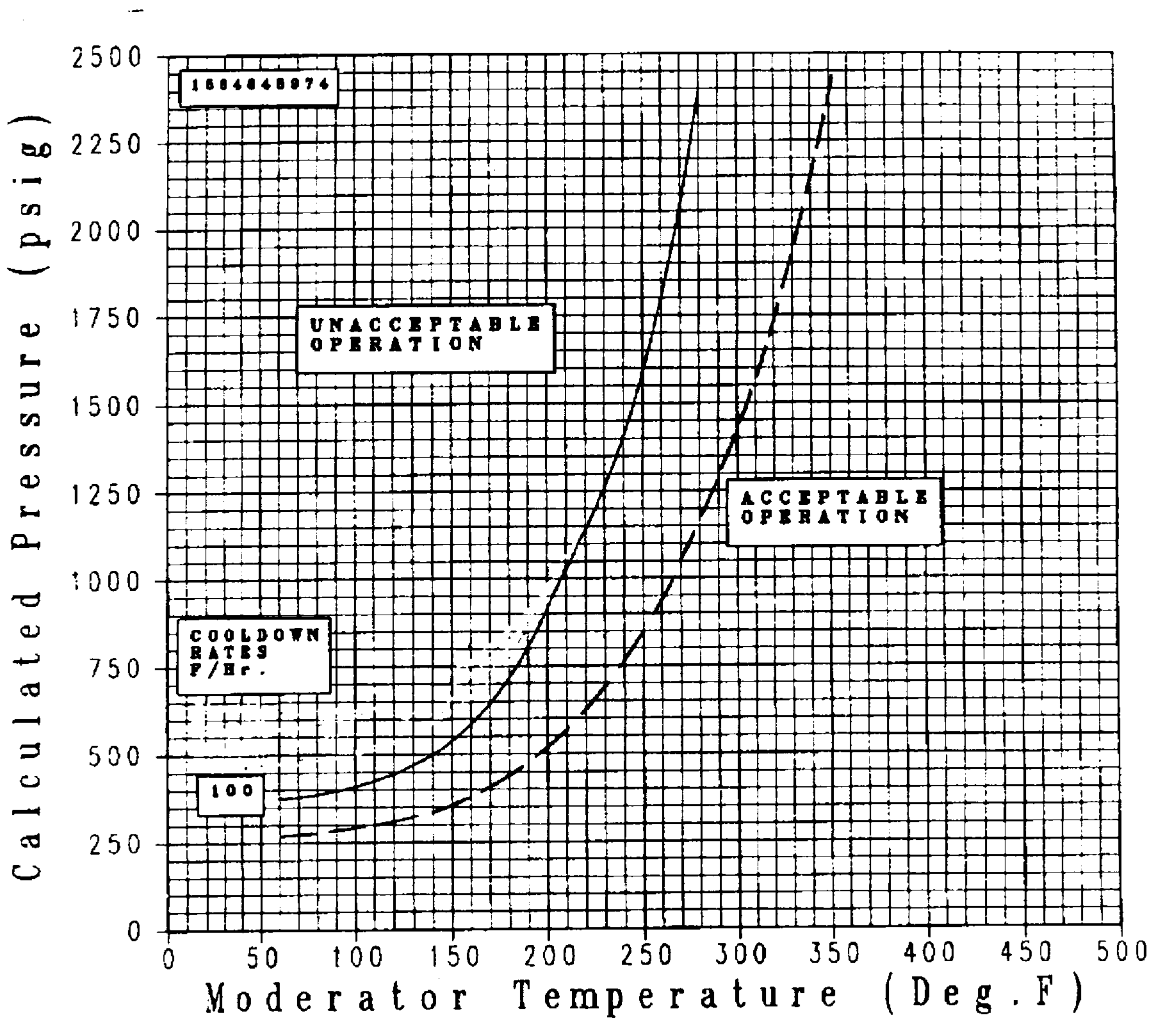


Figure 7. Comparison of Cool-Down Curves for the Existing and Proposed Methods - PWR [Dashed Curve = Existing (K_{Ia}) and Solid Curve = Proposed (K_{Ic})]

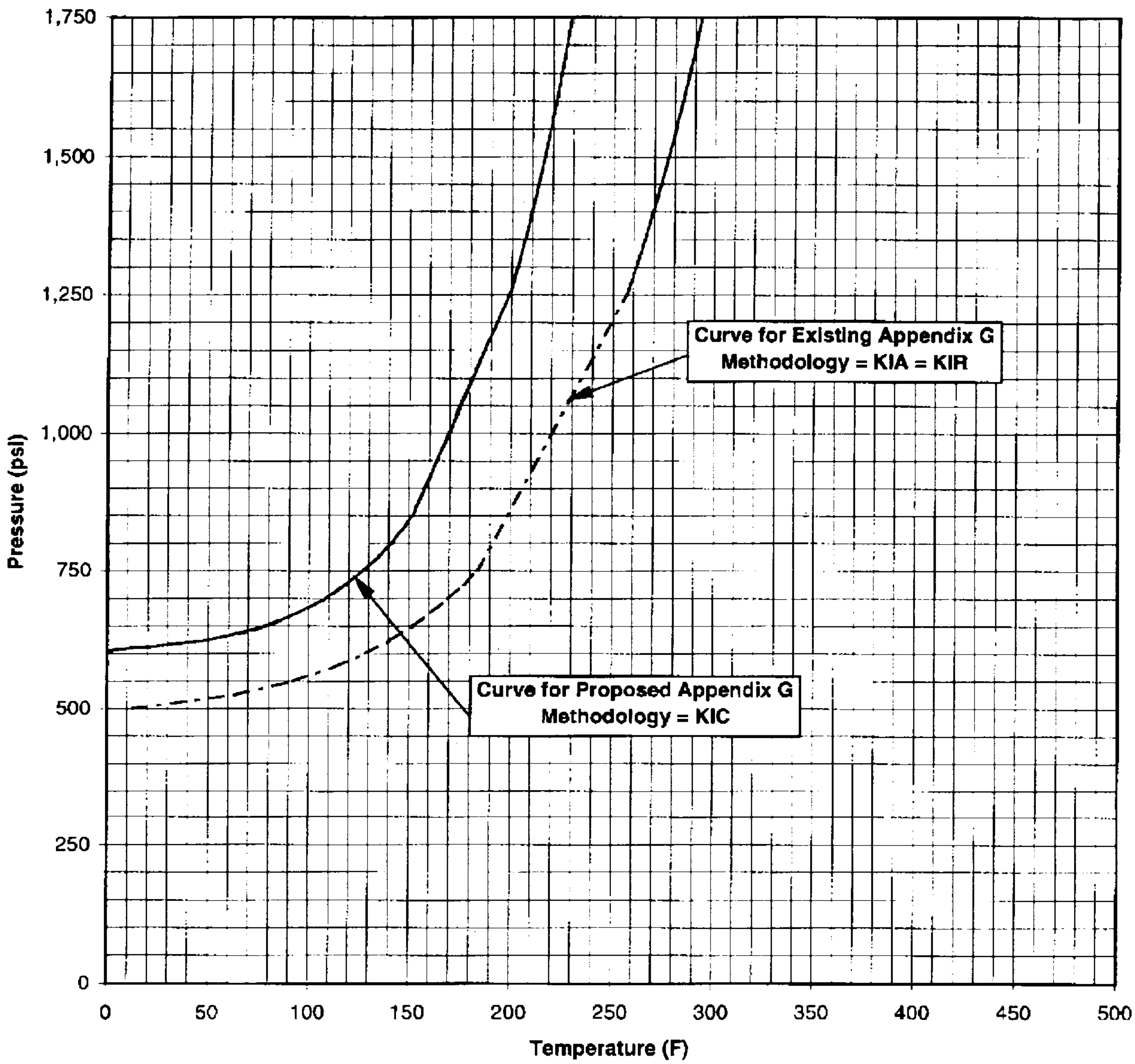


Figure 8. Comparison of Hydrotest P-T Curves for the Existing and Proposed Methods - BWR [Dashed Curve=Existing (K_{Ia}) and Solid Curve=Proposed (K_{Ic})]

Appendix A

Section XI P-T Limit Curve Sample Problems

Introduction

This series of sample problems was developed to allow comparison calculations to be carried out to support the proposed change from K-IA to K-IC in Appendix G of Section XI. These problems were developed in a meeting held on July 7, 1998, between the NRC staff, Westinghouse, ORNL, and Framatome Technologies. Later, a variation on the sample problems was developed for application to BWRs.

The sample problems involve a tightly specified reference case, with two variations, and then two P-T Limit curve calculations whose input is also tightly specified, one using K-IA and the second using K-IC. The goal of the problems is to determine the margin on pressure which exists using the K-IA approach, and the margin which exists with the proposed K-IC approach.

The problem input variables are contained in the attached tables. The problem statement is given below. As will be seen there are two problem types, the first being a best-estimate, or reference problem, and the second being standard P-T limit curves determined using code-type assumptions, with safety factors.

Reference Cases (Best Estimate)

Determine a best estimate P-T Cooldown Curve for a typical reactor vessel, over the entire temperature range of operation, starting at 70F. For BWR plants, also calculate a hydrotest pressure versus temperature curve. The problem input is defined in Table 1. This problem is meant to be a best estimate curve with no specific safety factors, and best estimate values for each of the variables. Only pressure and thermal stresses are used for case R1. Although these stresses are the only ones presently considered in P-T limit curve calculations, other stresses can exist in the vessel, and two other cases were constructed to obtain additional information on these issues. These other two cases treat stresses which are at issue regardless of which toughness is used for the calculations, but are provided for information.

Reference case R2. This case is similar to case R1, but the weld residual stresses are added for a longitudinal weld in the reactor vessel.

Reference case R3. This case is similar to case R2, but now the clad residual stresses are added. Since the clad residual stresses are negligible at higher temperatures, this calculation is only performed at room temperature, or 70F.

The stress intensity factor results for the reference cases may not always result in the maximum value at the deepest point of the flaw, so care should be taken to check this. If the maximum value is not at the deepest point, the calculated ratio of K / K-IC should be calculated around the periphery, and reported. The resulting allowable pressure would then be determined from the governing result at each time step. The calculation method could use either Section XI Appendix A, or the ORNL method, as documented in Table A-1.

P-T Curve Cases

Case 1 is a classic P-T Curve calculation done according to the existing rules in Section XI Appendix G, using the K-IA curve and the code specified safety factors. The input values are provided in Table A-2, for both PWR and BWR plants.

Case 2 is the same as case 1, except that the fracture toughness curve K-IC is used. This is the proposed Code change.

In each case a full P-T limit curve should be calculated, but there is no need to calculate leak test temperature, bolt-up temperature, or any other parameters. For BWR plants, a hydrotest pressure versus temperature curve is also required.

TABLE A-1: REFERENCE CASE VARIABLES

Reference Case 1

Vessel Geometry:	Thickness = 9.0 inch (PWR) or 6.0 inches (BWR) Inside Radius = 90 inch (PWR) or 125 inches (BWR) Clad Thickness = 0.25 inch
Flaw:	Semi-elliptic Surface Flaw, Longitudinal Orientation Depth = 1.0 inch Length = 6 x Depth
Toughness:	Mean K_{IC} , from report ORNL/NRC/LTR/93-15, July 12, 1993 $K_{IC} = 36.36 + 51.59 \exp [0.0115 (T-RT_{NDT})]$
Loading:	100F/Hr cooldown from 550F to 200F 20F/Hr cooldown from 200F to 70F
Film Coefficient:	h = 1000B/hr-ft-F
Stress Intensity Factor Expression:	Section XI, Appendix A, or ORNL Influence Coefficients, from ORNL/NRC/LTR-93-33 Rev. 1, Sept. 30, 1995
Irradiation Effects:	RT_{NDT} = 236°F(PWR) or 168°F (BWR) @ inside surface = 220°F(PWR) @ depth = 1.0 in. = 200°F(PWR) @ depth = T/4 = 133°F(PWR) @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of coolant temperature and for BWR plants, calculate hydrotest pressure as a function of coolant temperature.

Reference Case 2

Same as Reference Case 2, but for the loadings, add a weld residual stress distribution.

	Location (a/t)	Stress(ksi)	Location (a/t)	Stress(ksi)
Inner Surface	0.000	6.50	0.045	5.47
	0.067	4.87	0.101	3.95
	0.134	2.88	0.168	1.64
	0.226	-0.79	0.285	-3.06
	0.343	-4.35	0.402	-4.31
	0.460	-3.51	0.510	-2.57
	0.572	-1.70	0.619	-1.05
	0.667	-0.46	0.739	0.35
	0.786	0.87	0.834	1.41
	0.881	1.96	0.929	2.55
	0.976	3.20	1.000	3.54

Reference Case 3

Same as Reference Case 2, but add clad residual stress distribution, and calculate allowable pressure only at 70°F.

For the clad residual stress distribution, choose either distribution 1 or distribution 2, from the attached figures. Figure A-1 was calculated from the ORNL Favor Code, and Figure A-2 was taken from a technical paper which presents results of residual stresses measured on nozzle drop-out materials.

TABLE A-2: P-T Calculation Cases

Calculation Case 1

Vessel Geometry: Thickness - 9.0 inch (PWR), 6.0 inches (BWR)
Inside Radius = 90 inch (PWR), 125 inches (BWR)
Clad Thickness = 0.25 inch

Flaw: Semi-elleptic Surface Flaw, Longitudinal Orientation
Depth = 1.0 inch
Length = 6 x Depth

Toughness: K_{Ia}

Loading: 100F/hr cooldown, 550 to 200 F
20F/hr cooldown, 200 to 70F

Stress Intensity Factor Expression: Latest Section XI App G expression (from ORNL/NRC/LTR-93-33, Rev. 1)

Irradiation Effects: ART = 236F(PWR) or 168°F(BWR) @ inside surface
= 220F(PWR) @ depth = 1.0 inch
= 200F(PWR) @ depth = T/4
= 133F(PWR) @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of temperature, and for BWRs calculate hydrotest pressure as a function of temperature.

Calculation Case 2

Same parameters as Case 1, but Toughness = K_{IC}

From ORNL Favor Code, per Terry Dickson, 7-9-98

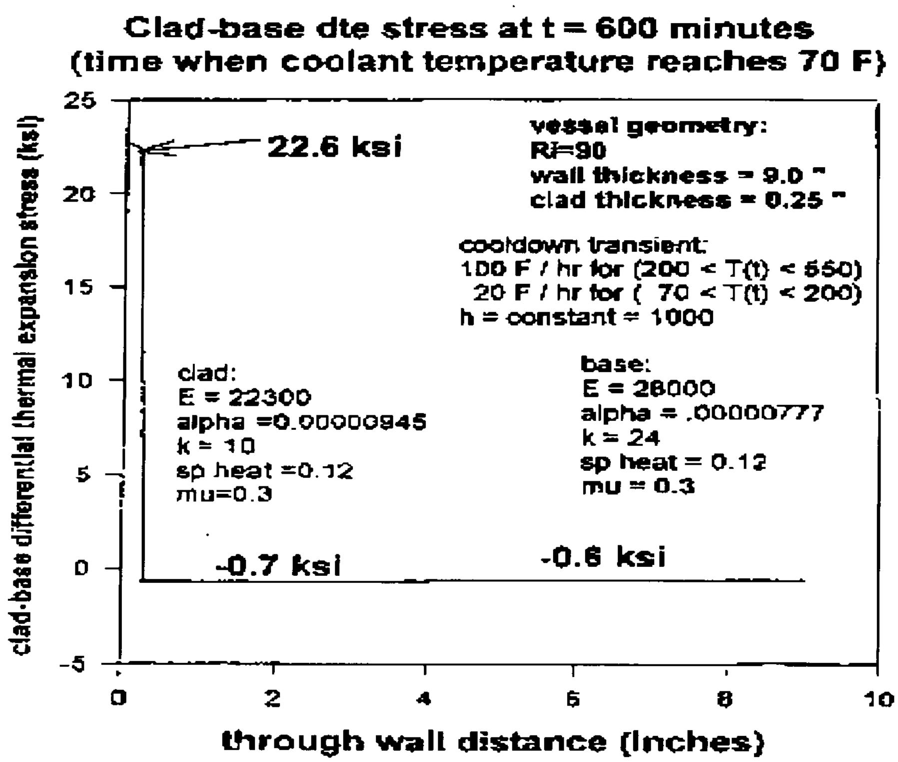


Figure A-1: Clad-base metal stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

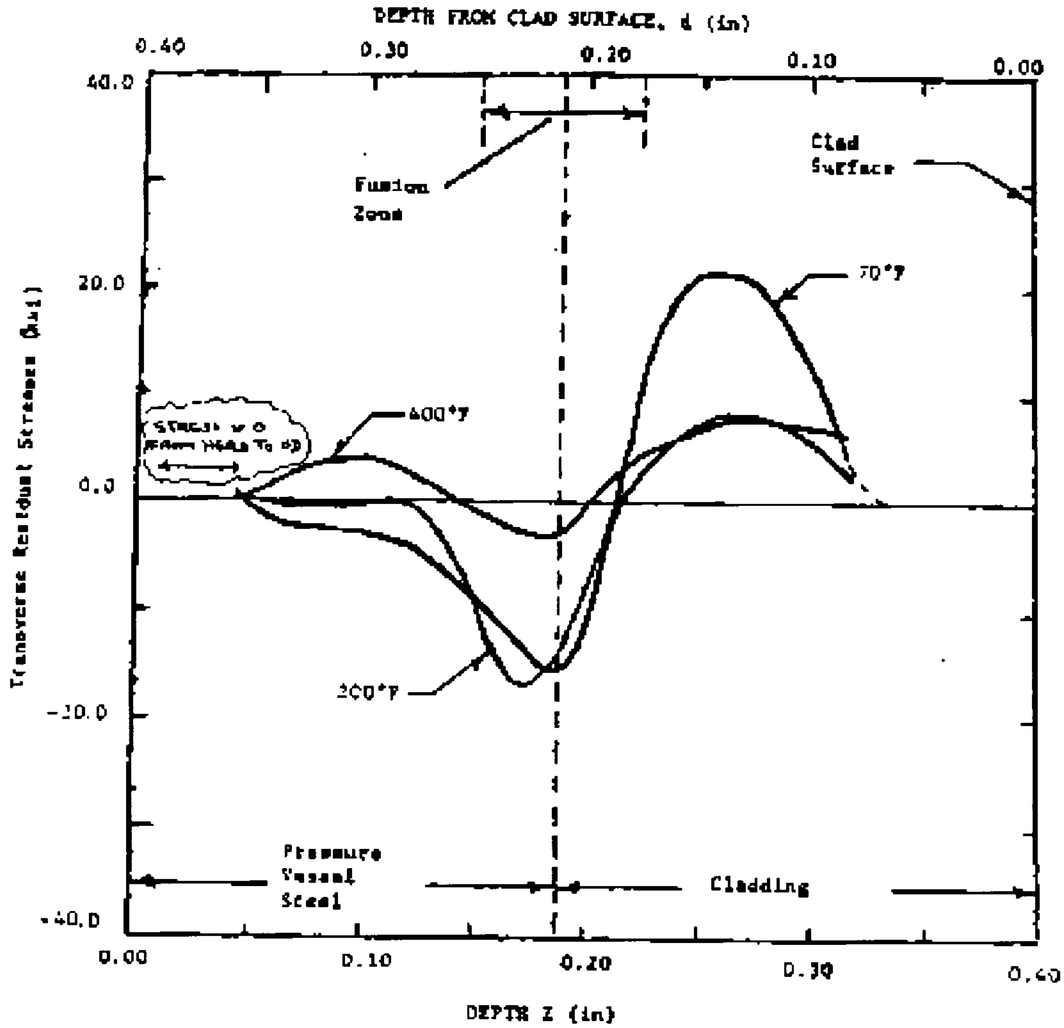


Figure A-2: Residual Stresses Transverse to Direction of Welding

ATTACHMENT 7

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

General Electric Company Report
GE-NE-B11-00836-00-02a NP
(non-proprietary information)

GE Nuclear Energy

Engineering and Technology
General Electric Company
175 Curtner Avenue,
San Jose, CA 95125

GE-NE-B11-00836-00-02a NP
Revision 0
Class I
July 2000

Pressure-Temperature Curves

For

PECO Energy Company

Limerick Unit 2

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT PLEASE READ CAREFULLY

EXECUTIVE SUMMARY

This report provides the pressure-temperature curves (P-T curves) developed to present steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. The methodology used to generate the P-T curves in this report is similar to the methodology used to generate the P-T curves in 1993 [1]. Several improvements were made to the P-T curve methodology; the improvements include, but are not limited to, the incorporation of ASME Code Cases N-640 and N-588. ASME Code Case N-640 allows the use of K_{IC} rather than K_{Ia} to determine T-RT_{NDT}. ASME Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors of Appendix G for axial and circumferential welds. Descriptions of other improvements are included in the P-T curve methodology section.

CONCLUSIONS

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) non-nuclear heatup/cooldown and low-level physics tests, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

There are four vessel regions that should be monitored against the P-T curve operating limits; these regions are defined on the thermal cycle diagram [2]:

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

For the core not critical and the core critical curve, the P-T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram [2] and the nozzle thermal cycle diagrams [3]. The bounding transients used to develop the curves are described in this report. For the hydrostatic pressure and leak test curve, a coolant heatup and cooldown temperature rate of 20°F/hr or less must be maintained at all times.

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

Composite P-T curves were generated for each of the Pressure Test, Core Not Critical and Core Critical conditions at 32 effective full power years (EFPY). The composite curves were generated by enveloping the most restrictive P-T limits from the separate bottom head, beltline, upper vessel and closure assembly P-T limits. Separate P-T curves were developed for the upper vessel, beltline (at 22 and 32 EFPY), and bottom head for the Pressure Test and Core Not Critical conditions. A composite P-T curve was also generated for the Core Critical condition at 22 EFPY.

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1.0 INTRODUCTION

The pressure-temperature (P-T) curves included in this report have been developed to present steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. Complete P-T curves were developed for 22 and 32 effective full power years (EFPY). The P-T curves are provided in Section 5.0 and a tabulation of the curves is included in Appendix B.

The methodology used to generate the P-T curves in this report is presented in Section 4.3 and is similar to the methodology used to generate the P-T curves in 1993 [1]. Several improvements were made to the P-T curve methodology; the improvements include, but are not limited to, the incorporation of ASME Code Cases N-640 and N-588. ASME Code Case N-640 allows the use of K_{IC} rather than K_{Ia} to determine T-RT_{NDT}. ASME Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors of Appendix G for axial and circumferential welds. Descriptions of other improvements are included in the P-T curve methodology section. P-T curves are developed using geometry of the RPV shells and discontinuities, the initial RT_{NDT} of the RPV materials, and the adjusted reference temperature (ART) for the beltline materials.

The initial RT_{NDT} is the reference for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. The Charpy energy data used to determine the initial RT_{NDT} values were tabulated from the Certified Material Test Report (CMTRs). The data and methodology used to determine initial RT_{NDT} is documented in Section 4.1.

Adjusted Reference Temperature (ART) is the reference temperature when including irradiation shift and a margin term. Regulatory Guide 1.99, Rev. 2 [7] provides the methods for calculating ART. The value of ART is a function of RPV 1/4T fluence and beltline material chemistry. The ART calculation, methodology, and ART tables for 22 and 32 EFPY are included in Section 4.2. The 32 EFPY fluence value of 1.9×10^{18} n/cm^2 used in this report is that used for the currently licensed Limerick Unit 2 P-T curves. A discussion of fluence is included in Section 4.2.1.2. Beltline chemistry values were updated to include the current data available from the vessel fabricator. The chemistry data is discussed in Section 4.2.1.1.

Comprehensive documentation of the RPV discontinuities that are considered in this report is included in Appendix A. This appendix also includes a table that documents which non-beltline discontinuity curves are used to protect the discontinuities.

Guidelines and requirements for operating and temperature monitoring are included in Appendix C. Finally, GE SIL 430, a GE service information letter regarding Reactor Pressure Vessel Temperature Monitoring is included in Appendix D.

2.0 SCOPE OF THE ANALYSIS

The methodology used to generate the P-T curves in this report is similar to the methodology used to generate the P-T curves in 1993 [1]. A detailed description of the P-T curve bases is included in Section 4.3. Several improvements were made to the P-T curve methodology; the improvements included the incorporation of ASME Code Cases N-640 [4] and N-588 [5]. ASME Code Case N-640 allows the use of K_{IC} rather than K_{Ia} to determine T-RT_{NDT}. ASME Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors to consider attenuation to reference flaw orientation of Appendix G for circumferential welds. This Code Case also provides an alternative procedure for calculating the applied stress intensity factor for axial welds. Other improvements include, but are not limited to the following:

- Generation of separate curves for the upper vessel in addition to those generated for the beltline, and bottom head.
- Comprehensive description of discontinuities used to develop the non-beltline curves (see Appendix A).

The pressure-temperature (P-T) curves are established to the requirements of 10CFR50, Appendix G [6] to assure that brittle fracture of the reactor vessel is prevented. Part of the analysis involved in developing the P-T curves is to account for irradiation embrittlement effects in the core region, or beltline. The method used to account for irradiation embrittlement is described in Regulatory Guide 1.99, Rev. 2 [7].

In addition to beltline considerations, there are non-beltline discontinuity limits such as nozzles, penetrations, and flanges that influence the construction of P-T curves. The non-beltline limits are based on generic analyses that are adjusted to the maximum reference temperature of nil ductility transition (RT_{NDT}) for the applicable Limerick Unit 2 vessel components. The non-beltline limits are discussed in Section 4.3 and are also governed by requirements in [6].

Furthermore, curves are included to allow monitoring of the vessel bottom head and upper vessel regions separate from the beltline region. This refinement could minimize heating requirements prior to pressure testing. Operating and temperature monitoring requirements are found in Appendix C. Temperature monitoring requirements and methods are available in GE Services Information Letter (SIL) 430 contained in Appendix D.

3.0 ANALYSIS ASSUMPTIONS

The following assumption is made for this analysis:

For end-of-license (32 EFPY) fluence an 80% capacity factor is used to determine the EFPY for a 40-year plant life. The 80% capacity factor is based on the objective to have BWR's available for full power production 80% of the year (refueling outages, etc. ~20% of the year).

4.0 ANALYSIS

4.1 INITIAL REFERENCE TEMPERATURE

4.1.1 Background

The initial RT_{NDT} values for all low alloy steel vessel components are needed to develop the vessel P-T limits. The requirements for establishing the vessel component toughness prior to 1972 were per the ASME Code Section III, Subsection NB-2300 and are summarized as follows:

a. Test specimens shall be longitudinally oriented CVN specimens.

b. At the qualification test temperature (specified in the vessel purchase specification), no impact test result shall be less than 25 ft-lb., and the average of three test results shall be at least 30 ft-lb.

c. Pressure tests shall be conducted at a temperature at least 60°F above the qualification test temperature for the vessel materials.

The current requirements used to establish an initial RT_{NDT} value are significantly different. For plants constructed according to the ASME Code after Summer 1972, the requirements per the ASME Code Section III, Subsection NB-2300 are as follows:

a. Test specimens shall be transversely oriented (normal to the rolling direction) CVN specimens.

b. RT_{NDT} is defined as the higher of the dropweight NDT or 60°F below the temperature at which Charpy V-Notch 50 ft-lb. energy and 35 mils lateral expansion are met.

c. Bolt-up in preparation for a pressure test or normal operation shall be performed at or above the highest RT_{NDT} of the materials in the closure flange region or lowest service temperature (LST) of the bolting material, whichever is greater.

10CFR50 Appendix G [6] states that for vessels constructed to a version of the ASME Code prior to the Summer 1972 Addendum, fracture toughness data and data analyses must be supplemented in an approved manner. GE developed methods for analytically converting fracture toughness data for vessels constructed before 1972 to comply with current requirements. These methods were developed from data in WRC Bulletin 217 [11] and from data collected to respond to NRC questions on FSAR submittals in the late 1970s. In 1994, these methods of estimating RT_{NDT} were submitted for generic approval by the BWR Owners' Group [12], and approved by the NRC for generic use [13].

4.1.2 Values of Initial RT_{NDT} and Lowest Service Temperature (LST)

To establish the initial RT_{NDT} temperatures for the Limerick Unit 2 vessel per the current requirements, calculations were performed in accordance with the GE method for determining RT_{NDT}. Example RT_{NDT} calculations for vessel plate, weld, HAZ, and forging, and bolting material LST are summarized in the remainder of this section.

For vessel plate material, the first step in calculating RT_{NDT} is to establish the 50 ft-lb. transverse test temperature from longitudinal test specimen data (obtained from certified material test reports, CMTRs). For Limerick Unit 2 CMTRs, typically six energy values were listed at a given test temperature, corresponding to two sets of Charpy tests. The lowest energy Charpy value is adjusted by adding 2°F per ft-lb. energy difference from 50 ft-lb.

For example, for plate heat B3416-1 in the intermediate shell course of Limerick Unit 2, the lowest Charpy energy and test temperature from the CMTRs is 35.0 ft-lb. at 40°F. The estimated 50 ft-lb. longitudinal test temperature is:

$$T_{50L} = 40°F + [(50 - 35) \text{ ft-lb.} \cdot 2°F/\text{ft-lb.}] = 70°F$$

The transition from longitudinal data to transverse data is made by adding 30°F to the 50 ft-lb. transverse test temperature; thus, for this case above,

$$T_{50T} = 70°F + 30°F = 100°F$$

The initial RT_{NDT} is the greater of nil-ductility transition temperature (NDT) or (T_{50T}- 60°F). Dropweight testing to establish NDT for plate material was listed in the CMTR; the NDT for the case above was 10°F. Thus, the initial RT_{NDT} for plate heat B3416-1 was 40°F.

For the Limerick Unit 2 beltline weld Heat 432A2671 (contained in Shell Rings #1 and #2), the CVN results were used to calculate the RT_{NDT}. The 50 ft-lb. test temperature is applicable to the weld material, but the 30°F adjustment to convert longitudinal data to transverse data is not applicable to weld material. For Heat 432A2671, the NDT was not available and the lowest Charpy energy was 31 ft-lb. at 10°F as recorded in the weld qualification records. Therefore, the initial RT_{NDT} is:

$$(T_{50T} - 60°F) = 10°F + [(50 - 31.0)\text{ ft-lb.} \cdot 2°F/\text{ft-lb.}] - 60°F = -12°F$$

For the vessel HAZ material, the RT_{NDT} is assumed to be the same as for the base material since ASME Code weld procedure qualification test requirements and post-weld heat treat data indicate this assumption is valid.

For vessel forging material, such as nozzles and closure flanges, the method for establishing RT_{NDT} is the same as for vessel plate material. For the feedwater nozzle at Limerick Unit 2, the NDT was –20°F and the lowest CVN data was 25 ft-lb. at -20°F. The corresponding value of (T_{50T} – 60°F) was:

$$(T_{50T} - 60°F) = \{[-20 + (50 - 25)\text{ ft-lb.} \cdot 2°F/\text{ft-lb.}] + 30\} - 60°F = 0°F.$$

Therefore, the initial RT_{NDT} was 0°F.

In the bottom head region of the vessel, the vessel plate method is applied for estimating RT_{NDT}. For the bottom head side plate of Limerick Unit 2 (Heat C9362-2), the NDT was not available and the lowest CVN data was 41 ft-lb. at 40°F The corresponding value of (T_{50T} – 60°F) was:

$$(T_{50T} - 60°F) = \{[40 + (50 - 41)\text{ ft-lb.} \cdot 2°F/\text{ft-lb.}] + 30°F\} - 60°F = 28°F.$$

Therefore, the initial RT_{NDT} was 28°F.

For bolting material, the current ASME Code requirements define the lowest service temperature (LST) as the temperature at which transverse CVN energy of 45 ft-lb. and 25 mils lateral expansion (MLE) were achieved. If the required Charpy results are not met, or are not reported, but the CVN energy reported is above 30 ft-lb., the requirements of the ASME Code Section III, Subsection NB-2300 at construction are applied, namely that the 30 ft-lb. test temperature plus 60°F is the LST for the bolting materials. Charpy data for the Limerick Unit 2 closure studs meet the 45 ft-lb., 25 MLE requirement at 10°F. Therefore, the LST for the bolting material is 10°F. The highest RT_{NDT} in the closure flange region is 10°F, for the top head and vessel shell flange materials. Thus, the higher of the LST and the RT_{NDT} +60°F is 70°F, the boltup limit in the closure flange region.

The initial RT_{NDT} values for the Limerick Unit 2 reactor vessel materials are listed in Tables 4-1, 4-2, and 4-3. This tabulation includes beltline, closure flange, feedwater nozzle, and bottom head materials that were considered in generating the P-T curves.

Table 4-1: RT_{NDT} values for Limerick Unit 2 Vessel Materials

COMPONENT	HEAT	TEST TEMP. (°F)	CHARPY ENERGY (FT-LB)			(T_{50T}-60) (°F)	DROP WEIGHT NDT	RT_{NDT} (°F)
PLATES & FORGINGS:								
Top Head & Flange								
Shell Flange (Vessel)	2L2058	10	167	142	121	-20	10	10
Head Flange	123B300	10	130	125	118	-20	10	10
Top Head Dome	C9491	40	49	48	44	22	N/A	22
Top Head Side Plates	C9646-2	10	67	77	71	-20	N/A	-20
	C9479-1	10	109	101	118	-20	N/A	-20
Shell Courses								
Shell Ring #5	C9632-1	10	75	77	75	-20	N/A	-20
	C9800-2	10	55	48	54	-16	N/A	-16
Shell Ring #4	B3582	40	65	67	69	10	N/A	10
	C9633	40	40	47	62	30	N/A	30
Shell Ring #3	C9590-2	40	55	76	58	10	-40	10
	C9590-1	40	44	42	43	26	-10	26
	C9601	40	76	67	71	10	-10	10
Shell Ring #2	C9569-2	40	68	62	68	10	-20	10
	C9526-1	40	65	60	66	10	-40	10
	C9526-2	40	71	74	87	10	-30	10
Shell Ring #1	B3312-1	40	58	63	68	10	-20	10
	B3416-1	40	61	35	37	40	10	40
	C9621-2	40	44	47	60	22	0	22
Bottom Head								
Center Plate	C9306-1	10	65	43	31	18	N/A	18
	C9245-2	40	55	44	64	22	N/A	22
Side Plate	C9269-2	40	51	56	44	22	N/A	22
	C9362-2	40	68	63	41	28	N/A	28
Support Skirt	B3701-3A	40	56	48	41	28	N/A	28
Refueling Bellows	A8792	10	40	42	40	0	N/A	0
STUDS:	19626	10	46	48	48	10		
NUTS and WASHERS:	17501	10	52	50	50	10		

Table 4-2: RT_{NDT} values for Limerick Unit 2 Nozzle Materials

COMPONENT	HEAT	TEST TEMP. (°F)	CHARPY ENERGY (FT-LB)			(T_{50T}-60) (°F)	DROP WEIGHT NDT	RT_{NDT} (°F)
NOZZLES								
Recirc Outlet, N1	Q2Q30W	40	52	52	62	10	40	40
	Q2Q37W	40	41	52	36	38	40	40
Recirc Inlet, N2	Q2Q29W	40	40	32	34	46	40	46
Steam Outlet, N3	Q2Q23W	40	60	52	57	10	40	40
Feedwater, N4	Q2Q29W	-20	25	42	51	0	-20	0
Core Spray, N5	Q2Q42W	-20	70	56	62	-50	-20	-20
	Q2Q52W	-20	54	50	54	-50	-20	-20
Head Spray and Spare, N6	Q2Q33W	40	107	139	147	10	40	40
Vent, N7	Q2Q20W	40	39	45	60	32	40	40
Jet Pump Instrumentation, N8	Q2Q32W	40	46	36	49	38	40	40
CRD Hyd. System Return, N9	Q2Q32W	-10	29	32	47	2	-10	2
Drain, N15	B2W	-10	34	32	34	-4	N/A	-4
LPCI, N17	Q2Q33W	-20	32	27	53	-4	-20	-4

Table 4-3: RT_{NDT} values for Limerick Unit 2 Weld Materials

COMPONENT	HEAT	TEST TEMP. (°F)	CHARPY ENERGY (FT-LB)			(T_{50T}-60) (°F)	DROP WEIGHT NDT	RT_{NDT} (°F)
WELDS:								
Vertical Welds and LPCI Nozzle								
BA,BB,BC,BD,BE,BF,BN,BP	662A746	10	35	38	47	-20	N/A	-20
BA,BB,BC,BK,BM	03R728T	10	64	71	72	-50	N/A	-50
BA,BB,BC,BD,BE,BF,BN,BP	3P4000 Lot 3933	10	90	86	91	-50	N/A	-50
BA,BB,BC, LPCI,BD,BE,BF	432A2671	10	31	31	33	-12	N/A	-12
BA,BB,BC,LPCI	07L669	10	50	50	54	-50	N/A	-50
BA,BB,BC	402A0462	10	75	77	86	-50	N/A	-50
BA,BB,BC	401Z9711	10	98	99	104	-50	N/A	-50
LPCI	S3986	10	46	51	49	-42	N/A	-42
LPCI,BD,BE,BF	09L853	10	78	78	79	-50	N/A	-50
BG,BH,BJ, LPCI	4P4784 Lot 3930	10	79	75	81	-50	N/A	-50
LPCI	C3L46C	10	35	39	40	-20	N/A	-20
LPCI	422B7201	10	65	66	72	-50	N/A	-50
BG,BH,BJ	422A7011	10	127	135	135	-50	N/A	-50
BK,BM	04M569	-20	100	102	105	-80	N/A	-50
BK,BM	06M547	-20	89	133	158	-80	N/A	-50
FIELD GIRTH WELDS:	07L857	10	28	36	39	-6	N/A	-6
	402C4371	10	82	84	92	-50	N/A	-50
	411A3531	10	60	60	68	-50	N/A	-50
	09M057	10	43	43	44	-36	N/A	-36
	412P3611	-20	52	65	69	-80	-80	-80
	03M014	10	42	44	47	-34	N/A	-34
	L83355	-10	51	52	63	-70	-90	-70
	640892	0	55	62	62	-60	-70	-60
	401P6741	0	51	57	68	-60	-70	-60
Top and Bottom Head Assembly Welds								
DA,DB,DC,DD,DE,DF,DG	662A746	10	97	95	88	-50	N/A	-50
DA,DB,DC,DD,DE,DF,DG	03R728T	10	35	38	47	-20	N/A	-20
DA,DB,DC,DD,DE,DF,DG, DH,AF,AG,AH,AJ	3P4000 Lot 3933	10	60	60	68	-50	N/A	-50
DA,DB,DC,DD,DE,DF,DG, AH,AJ	432A2671	10	94	91	90	-50	N/A	-50
DH,DJ,DK,DM,DN,DP,DG, AG,AJ	07L669	10	102	93	95	-50	N/A	-50
DG,AF,AG,AJ	402A0462	10	50	50	54	-50	N/A	-50
DH,DJ,DK,DM,DN,DP,AJ	401Z9711	10	98	99	104	-50	N/A	-50
AG,AJ	09L853	10	35	39	40	-20	N/A	-20
AJ	4P4784 Lot 3930	10	31	31	33	-12	N/A	-12
DG	C3L46C	10	75	77	80	-50	N/A	-50
DG	1P4218 Lot 3932	10	56	71	60	-50	N/A	-50

4.2 ADJUSTED REFERENCE TEMPERATURE FOR BELTLINE

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2 (Rev 2) provides the methods for determining the ART. The Rev 2 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section. An evaluation of ART for all beltline plates and several beltline welds was made and summarized in Table 4-4 for 22 EFPY and Table 4-5 for 32 EFPY.

4.2.1 Regulatory Guide 1.99, Revision 2 (Rev 2) Methods

The value of ART is computed by adding the SHIFT term for a given value of effective full power years (EFPY) to the initial RT_{NDT}. For Rev 2, the SHIFT equation consists of two terms:

$$SHIFT = \Delta RT_{NDT} + Margin$$

where,

$$\Delta RT_{NDT} = [CF] \cdot f^{(0.28 - 0.10 \log f)}$$

$$Margin = 2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$$

$$f = 1/4\ T\ fluence / 10^{19}$$

$$ART = Initial\ RT_{NDT} + SHIFT$$

4.2.1.1 Chemistry

The vessel beltline chemistries were obtained from sources including CMTRs [14] and a recent NRC submittal [15]. The copper (Cu) and nickel (Ni) values were used with Tables 1 and 2 of Rev 2, to determine a chemistry factor (CF) per Paragraph 1.1 of Rev 2 for welds and plates, respectively. The margin term σ_Δ has constant values in Rev 2 of 17°F for plate and 28°F for weld. However, σ_Δ need not be greater than $0.5 \cdot \Delta RT_{NDT}$. Since the GE/BWROG method of estimating RT_{NDT} operates on the lowest Charpy energy value (as described in Section 4.1.2) and provides a conservative adjustment to the 50 ft-lb. level, the value of σ_I is taken to be 0°F for the vessel plate and weld materials.

4.2.1.2 Fluence

The 32 EFPY fluence at the 1/4T depth of vessel wall for the Limerick Unit 2 vessel was obtained from currently licensed P-T curves. The currently licensed, 32 EFPY, peak ID fluence value for Limerick Unit 2 is 1.9×10^{18} n/cm^2.

4.2.2 Limiting Beltline Material

The limiting beltline material signifies the material that is estimated to receive the greatest embrittlement due to irradiation effects combined with initial RT_{NDT}. Using initial RT_{NDT}, chemistry, and fluence as inputs, Rev 2 was applied to compute ART. Table 4-4 lists values of beltline ART for 22 EFPY and Table 4-5 lists the values for 32 EFPY.

Plates and Welds

Thickness = 6.19 inches

32 EFPY Peak I.D. fluence = 1.9E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.3E+18 n/cm^2
22 EFPY Peak 1/4 T fluence = 9.0E+17 n/cm^2

LPCI Nozzle

Thickness = 6.19 inches

32 EFPY Peak I.D. fluence = 2.8E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.9E+17 n/cm^2
22 EFPY Peak 1/4 T fluence = 1.3E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	22 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	22 EFPY Shift °F	22 EFPY ART °F
PLATES:												
Shell Course #1												
14-1	B3312-1	0.13	0.58	90	10	9.0E+17	36	0	17	34	70	80
14-2	B3416-1	0.14	0.65	101	40	9.0E+17	40	0	17	34	74	114
14-3	C9621-2	0.15	0.60	110	22	9.0E+17	44	0	17	34	78	100
Shell Course #2												
17-1	C9569-2	0.11	0.51	73	10	9.0E+17	29	0	14	29	58	68
17-2	C9526-1	0.11	0.56	74	10	9.0E+17	29	0	15	29	59	69
17-3	C9526-2	0.11	0.56	74	10	9.0E+17	29	0	15	29	59	69
WELDS:												
seams BA,BB,BD,BE,BF,KA	432A2671/H019A27A	0.04	1.08	54	-12	9.0E+17	21	0	11	21	43	31
seams BA,BC	03R728/L910A27A*	0.03	0.92	41	-50	9.0E+17	16	0	8	16	32	-18
seams BA,BB,BC,BD,BE,BF	3P4000/3933	0.02	0.928	27	-50	9.0E+17	11	0	5	11	21	-29
seam BB	401Z9711/A022A27A	0.02	0.83	27	-50	9.0E+17	11	0	5	11	21	-29
seam BC	662A746/H013A27A	0.03	0.88	41	-20	9.0E+17	16	0	8	16	32	12
seam BC	402A0462/B023A27A	0.02	0.90	27	-50	9.0E+17	11	0	5	11	21	-29
seams BD,BE,KA	09L853/A111A27A	0.03	0.86	41	-50	9.0E+17	16	0	8	16	32	-18
seams BC,BD,BE,BF,KA	07L669/K004A27A	0.03	1.02	41	-50	9.0E+17	16	0	8	16	32	-18
seam AB	07L857/B101A27A	0.03	0.97	41	-6	9.0E+17	16	0	8	16	32	26
seam AB	L83355/S411B27AD	0.03	1.08	41	-70	9.0E+17	16	0	8	16	32	-38
seam AB	402C4371/C115A27A	0.02	0.92	27	-50	9.0E+17	11	0	5	11	21	-29
seam AB	03M014/C118A27A	0.01	0.94	20	-34	9.0E+17	8	0	4	8	16	-18
seam AB	411A3531/H004A27A	0.02	0.96	27	-50	9.0E+17	11	0	5	11	21	-29
seam AB	09M057/C109A27A	0.03	0.89	41	-36	9.0E+17	16	0	8	16	32	-4
seam AB	640892/J424B27AE	0.09	1.00	122	-60	9.0E+17	48	0	24	48	97	37
seam AB	401P6741/S419B27AG	0.03	0.92	41	-60	9.0E+17	16	0	8	16	32	-28
seam AB	412P3611/J417B27AF	0.03	0.93	41	-80	9.0E+17	16	0	8	16	32	-48
LPCI NOZZLE:												
Seam KA	432A2671/H019A27A	0.04	1.08	54	-12	1.3E+17	7	0	4	7	14	2
Seam KA	07L669/K004A27A	0.03	1.02	41	-50	1.3E+17	5	0	3	5	11	-39
Seam KA	C3L46C/J020A27A	0.02	0.87	27	-20	1.3E+17	4	0	2	4	7	-13
Seam KA	422B7201/L030A27A	0.04	0.90	54	-40	1.3E+17	7	0	4	7	14	-26
Seam KA	09L853/A111A27A	0.03	0.86	41	-50	1.3E+17	5	0	3	5	11	-39
Seam KA	4P4784/3930 (single wire)	0.06	0.87	82	-50	1.3E+17	11	0	5	11	22	-28
Seam KA	4P4784/3930 (tandem wire)	0.06	0.87	82	-20	1.3E+17	11	0	5	11	22	2
Forging 892L-1	Q2Q33W	0.15	0.83	115	-20	1.3E+17	15	0	8	15	30	10
Forging 892L-2	Q2Q33W	0.15	0.81	115	-6	1.3E+17	15	0	8	15	30	24
Forging 892L-3	Q2Q33W	0.15	0.82	115	-4	1.3E+17	15	0	8	15	30	26
Forging 892L-4	Q2Q33W	0.15	0.82	115	-20	1.3E+17	15	0	8	15	30	10

Table 4-4: Limerick Unit 2 Beltline ART Values (22 EFPY)

Plates and Welds

Thickness = 6.19 inches

32 EFPY Peak I.D. fluence = 1.9E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.3E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.3E+18 n/cm^2

LPCI Nozzle

Thickness = 6.19 inches

32 EFPY Peak I.D. fluence = 2.8E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.9E+17 n/cm^2
32 EFPY Peak 1/4 T fluence = 1.9E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	32 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLATES:												
Shell Course #1												
14-1	B3312-1	0.13	0.58	90	10	1.3E+18	43	0	17	34	77	87
14-2	B3416-1	0.14	0.65	101	40	1.3E+18	48	0	17	34	82	122
14-3	C9621-2	0.15	0.60	110	22	1.3E+18	52	0	17	34	86	108
Shell Course #2												
17-1	C9569-2	0.11	0.51	73	10	1.3E+18	35	0	17	34	69	79
17-2	C9526-1	0.11	0.56	74	10	1.3E+18	35	0	17	34	69	79
17-3	C9526-2	0.11	0.56	74	10	1.3E+18	35	0	17	34	69	79
WELDS:												
seams BA,BB,BD,BE,BF,KA	432A2671/H019A27A	0.04	1.08	54	-12	1.3E+18	26	0	13	26	51	39
seams BA,BC	03R728/L910A27A*	0.03	0.92	41	-50	1.3E+18	19	0	10	19	39	-11
seams BA,BB,BC,BD,BE,BF	3P4000/3933	0.02	0.928	27	-50	1.3E+18	13	0	6	13	26	-24
seam BB	401Z9711/A022A27A	0.02	0.83	27	-50	1.3E+18	13	0	6	13	26	-24
seam BC	662A746/H013A27A	0.03	0.88	41	-20	1.3E+18	19	0	10	19	39	19
seam BC	402A0462/B023A27A	0.02	0.90	27	-50	1.3E+18	13	0	6	13	26	-24
seams BD,BE,KA	09L853/A111A27A	0.03	0.86	41	-50	1.3E+18	19	0	10	19	39	-11
seams BC,BD,BE,BF,KA	07L669/K004A27A	0.03	1.02	41	-50	1.3E+18	19	0	10	19	39	-11
seam AB	07L857/B101A27A	0.03	0.97	41	-6	1.3E+18	19	0	10	19	39	33
seam AB	L83355/S411B27AD	0.03	1.08	41	-70	1.3E+18	19	0	10	19	39	-31
seam AB	402C4371/C115A27A	0.02	0.92	27	-50	1.3E+18	13	0	6	13	26	-24
seam AB	03M014/C118A27A	0.01	0.94	20	-34	1.3E+18	9	0	5	9	19	-15
seam AB	411A3531/H004A27A	0.02	0.96	27	-50	1.3E+18	13	0	6	13	26	-24
seam AB	09M057/C109A27A	0.03	0.89	41	-36	1.3E+18	19	0	10	19	39	3
seam AB	640892/J424B27AE	0.09	1.00	122	-60	1.3E+18	58	0	28	56	114	54
seam AB	401P6741/S419B27AG	0.03	0.92	41	-60	1.3E+18	19	0	10	19	39	-21
seam AB	412P3611/J417B27AF	0.03	0.93	41	-80	1.3E+18	19	0	10	19	39	-41
LPCI NOZZLE:												
Seam KA	432A2671/H019A27A	0.04	1.08	54	-12	1.9E+17	9	0	5	9	18	6
Seam KA	07L669/K004A27A	0.03	1.02	41	-50	1.9E+17	7	0	3	7	14	-36
Seam KA	C3L46C/J020A27A	0.02	0.87	27	-20	1.9E+17	5	0	2	5	9	-11
Seam KA	422B7201/L030A27A	0.04	0.90	54	-40	1.9E+17	9	0	5	9	18	-22
Seam KA	09L853/A111A27A	0.03	0.86	41	-50	1.9E+17	7	0	3	7	14	-36
Seam KA	4P4784/3930 (single wire)	0.06	0.87	82	-50	1.9E+17	14	0	7	14	28	-22
Seam KA	4P4784/3930 (tandem wire)	0.06	0.87	82	-20	1.9E+17	14	0	7	14	28	8
Forging 892L-1	Q2Q33W	0.15	0.83	115	-20	1.9E+17	19	0	10	19	39	19
Forging 892L-2	Q2Q33W	0.15	0.81	115	-6	1.9E+17	19	0	10	19	39	33
Forging 892L-3	Q2Q33W	0.15	0.82	115	-4	1.9E+17	19	0	10	19	39	35
Forging 892L-4	Q2Q33W	0.15	0.82	115	-20	1.9E+17	19	0	10	19	39	19

Table 4-5: Limerick Unit 2 Beltline ART Values (32 EFPY)

4.3 PRESSURE-TEMPERATURE CURVE METHODOLOGY

4.3.1 Background

Nuclear Regulatory Commission (NRC) 10CFR50 Appendix G [6] specifies fracture toughness requirements to provide adequate margins of safety during the operating conditions that a pressure-retaining component may be subjected to over its service lifetime. The ASME Code (Appendix G of Section XI of the ASME Code [9]) forms the basis for the requirements of 10CFR50 Appendix G. The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) non-nuclear heatup/cooldown and low-level physics tests, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

There are four vessel regions that should be monitored against the P-T curve operating limits; these regions are defined on the thermal cycle diagram [2]:

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

The closure flange region includes the bolts, top head flange, and adjacent plates and welds. The core beltline is the vessel location adjacent to the active fuel, such that the neutron fluence is sufficient to cause a significant shift of RT_{NDT}. The remaining portion of the vessel (i.e., upper vessel, lower vessel) include shells, components like the nozzles, the support skirt, and stabilizer brackets; these regions will also be called the non-beltline region.)

For the core not critical and the core critical curves, the P-T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram [2] and the nozzle thermal cycle diagrams [3]. The bounding transients used to develop the curves are described in the sections below. For the hydrostatic pressure and leak test curve, a coolant heatup and cooldown temperature rate of 20°F/hr or less must be maintained at all times.

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

The applicable temperature is the greater of the 10CFR50 Appendix G minimum temperature requirement or the ASME Appendix G limits. A summary of the requirements is as follows in Table 4-6:

Table 4-6: Summary of the 10CFR50 Appendix G Requirements

Operating Condition and Pressure	Minimum Temperature Requirement
I. Hydrostatic Pressure Test & Leak Test (Core is Not Critical) - Curve A	
1. At ≤ 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 60°F*
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 90°F
II. Normal operation (heatup and cooldown), including anticipated operational occurrences	
a. Core not critical - Curve B	
1. At ≤ 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 60°F*
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits or of highest closure flange region initial RT_{NDT} + 120°F
b. Core critical - Curve C	
1. At ≤ 20% of preservice hydrotest pressure, with the water level within the normal range for power operation	Larger of ASME Limits + 40°F or of a.1
2. At > 20% of preservice hydrotest pressure	Larger of ASME Limits + 40°F or of a.2 + 40°F or the minimum permissible temperature for the inservice system hydrostatic pressure test

* 60°F adder is included by GE as an additional conservatism as discussed in Section 4.3.2.3

There are four vessel regions that affect the operating limits: the closure flange region, the core beltline region, and the two regions in the remainder of the vessel (i.e., the upper vessel and lower vessel non-beltline regions). The closure flange region limits are controlling at lower pressures primarily because of 10CFR50 Appendix G [6] requirements. The non-beltline and beltline region operating limits are evaluated according to procedures in 10CFR50 Appendix G [6], ASME Code Appendix G [9], and Welding Research Council (WRC) Bulletin 175 [10]. The beltline region minimum temperature limits are adjusted to account for vessel irradiation.

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4.3.2 P-T Curve Methodology

4.3.2.1 Non-Beltline Regions

Non-beltline regions are defined as the vessel locations that are remote from the active fuel and where the neutron fluence is not sufficient to cause any significant shift of RT_{NDT}. Non-beltline components include most nozzles, the closure flanges, some shell plates, the top and bottom head plates and the control rod drive (CRD) penetrations.

Detailed stress analyses of the non-beltline components were performed for the BWR/6 specifically for the purpose of fracture toughness analysis. The analyses took into account all mechanical loading and anticipated thermal transients. Transients considered include 100°F/hr start-up and shutdown, SCRAM, loss of feedwater heaters or flow, loss of recirculation pump flow, and all transients involving emergency core cooling injections. Primary membrane and bending stresses and secondary membrane and bending stresses due to the most severe of these transients were used according to the ASME Code [9] to develop plots of allowable pressure (P) versus temperature relative to the reference temperature ($T - RT_{NDT}$). Plots were developed for the limiting BWR/6 components: the feedwater nozzle (FW) and the CRD penetration (bottom head). All other components in the non-beltline regions are categorized under one of these two components as described in Tables 4-7 and 4-8.

Table 4-7: Applicable BWR/4 Discontinuity Components FOR USE WITH FW CURVES A & B

Discontinuity Identification
FW Nozzle
CRD HYD System Return
Core Spray Nozzle
Recirculation Inlet Nozzle
Steam Outlet Nozzle
Main Closure Flange
Support Skirt
Stabilizer Brackets
Shroud Support Attachments
Core ΔP and Liquid Control Nozzle
Steam Water Interface
Jet Pump Instrumentation Nozzle
Shell
CRD and Bottom Head (B only)
Top Head Nozzles (B only)
Recirculation Outlet Nozzle (B only)

Table 4-8: Applicable BWR/4 Discontinuity Components FOR USE WITH CRD CURVES A&B

Discontinuity Identification
CRD and Bottom Head
Top Head Nozzle
Recirculation Outlet Nozzle

The P-T curves for the non-beltline region were conservatively developed for a large BWR/6 (nominal inside diameter of 251 inches). The analysis is considered appropriate for Limerick Unit 2 as the plant specific geometric values are bounded by the generic analysis for a large BWR/6, as determined in Section 4.3.2.1.1 through

Section 4.3.2.1.4. The generic value was adapted to the conditions at Limerick Unit 2 by using plant specific RT_{NDT} values for the reactor pressure vessel (RPV). The presence of nozzles and CRD penetration holes of the upper vessel and bottom head, respectively, has made the analysis different from a shell analysis such as the beltline. This was the result of the stress concentrations and higher thermal stress for certain transient conditions experienced by the upper vessel and the bottom head.

4.3.2.1.1 Pressure Test - Non-Beltline, Curve A (Using Bottom Head)

In a finite element analysis [ ], the CRD penetration region was modeled to compute the local stresses for determination of the stress intensity factor, K_I. The evaluation was modified to consider the new requirement for M_m as discussed in ASME Code Case N-588 and shown below. The results of that computation were K_I = 143.6 ksi-in$^{1/2}$ for an applied pressure of 1593 psig (1563 psig preservice hydrotest pressure at the top of the vessel plus 30 psig hydrostatic pressure at the bottom of the vessel). The computed value of (T - RT_{NDT}) was 84°F.

The limit for the coolant temperature change rate is 20°F/hr or less.

The value of M_m for an inside axial postulated surface flaw from Paragraph 2214-1 [5] was based on a thickness of 8.0 inches; hence, $t^{1/2} = 2.83$. The resulting value obtained was:

$$M_m = 1.85 \text{ for } \sqrt{t} \leq 2$$

$$M_m = 0.926 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464 = 2.6206$$

$$M_m = 3.21 \text{ for } \sqrt{t} > 3.464$$

K_{Im} is calculated from the equation in Paragraph G-2214.1 [5] and K_{Ib} is calculated from the equation in Paragraph G-2214.2 [5]:

$$K_{Im} = M_m \cdot \sigma_{pm} = \quad \text{ksi-in}^{1/2}$$

$$K_{Ib} = (2/3)\, M_m \cdot \sigma_{pb} = \quad \text{ksi-in}^{1/2}$$

The total K_I is therefore:

$$K_I = 1.5\,(K_{Im} + K_{Ib}) + M_m \cdot (\sigma_{sm} + (2/3) \cdot \sigma_{sb}) = 143.6 \text{ ksi-in}^{1/2}$$

This equation includes a safety factor of 1.5 on primary stress. The method to solve for $(T - RT_{NDT})$ for a specific K_I is based on the K_{Ic} the equation of Paragraph A-4200 in ASME Appendix A [8]:

$(T - RT_{NDT}) = \ln [(K_I - 33.2) / 20.734] / 0.02$

$(T - RT_{NDT}) = \ln [(144 - 33.2) / 20.734] / 0.02$

$(T - RT_{NDT}) = 84°F$

The generic curve was generated by scaling 143.6 ksi-in$^{1/2}$ by the nominal pressures and calculating the associated $(T - RT_{NDT})$:

Pressure Test CRD Penetration K_I and $(T - RT_{NDT})$ as a Function Of Pressure

Nominal Pressure (psig)	K_I (ksi-in$^{1/2}$)	$T - RT_{NDT}$ (°F)
1563	144	84
1400	129	77
1200	111	66
1000	92	52
800	74	33
600	55	3
400	37	-88

The highest RT_{NDT} for the bottom head plates and welds is 28°F, as shown in Tables 4-1 and 4-3.

Second, the P-T curve is dependent on the calculated K_I value, and the K_I value is proportional to the stress and the crack depth as shown below:

$$K_I \propto \sigma (\pi a)^{1/2} \tag{4-1}$$

The stress is proportional to R/t and, for the P-T curves, crack depth, *a*, is t/4. Thus, K_I is proportional to $R/(t)^{1/2}$. The generic curve value of $R/(t)^{1/2}$, based on the generic BWR/6 bottom head dimensions, is:

Generic: $$R / (t)^{1/2} = 138 / (8)^{1/2} = 49 \text{ inch}^{1/2} \tag{4-2}$$

The Limerick Unit 2 specific bottom head dimensions are R = 126.7 inches and t = 7.125 inches minimum [26], resulting in:

Limerick Unit 2 specific: $R / (t)^{1/2} = 126.7 / (7.125)^{1/2} = 47.5 \text{ inch}^{1/2}$ (4-3)

Since the generic value of $R/(t)^{1/2}$ is larger, the generic P-T curve is conservative when applied to the Limerick Unit 2 bottom head.

4.3.2.1.2 Core Not Critical Heatup/Cooldown - Non-Beltline Curve B (Using Bottom Head)

As discussed previously, the CRD penetration region limits were established primarily for consideration of bottom head discontinuity stresses during pressure testing. Heatup/cooldown limits were calculated by increasing the safety factor in the pressure testing stresses (Section 4.3.2.1.1) from 1.5 to 2.0.

The calculated value of K_I for pressure test is multiplied by a safety factor (SF) of 1.5, per ASME Appendix G [9] for comparison with K_{IR}, the material fracture toughness. A safety factor of 2.0 is used for the core not critical. Therefore, the K_I value for the core not critical condition is $(143.6 / 1.5) \cdot 2.0 = 191.5$ ksi-in$^{1/2}$.

Therefore, the method to solve for $(T - RT_{NDT})$ for a specific K_I is based on the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [8] for the core not critical curve:

$$(T - RT_{NDT}) = \ln [(K_I - 33.2) / 20.734] / 0.02$$
$$(T - RT_{NDT}) = \ln [(191.5 - 33.2) / 20.734] / 0.02$$
$$(T - RT_{NDT}) = 102°F$$

The generic curve was generated by scaling 192 ksi-in$^{1/2}$ by the nominal pressures and calculating the associated $(T - RT_{NDT})$:

Core Not Critical CRD Penetration K_I and $(T - RT_{NDT})$ as a Function of Pressure

Nominal Pressure (psig)	K_I (ksi-in$^{1/2}$)	$T - RT_{NDT}$ (°F)
1563	192	102
1400	172	95
1200	147	85
1000	123	73
800	98	57
600	74	33
400	49	-14

The highest RT_{NDT} for the bottom head plates and welds is 28°F, as shown in Tables 4-1 and 4-3.

As discussed in Section 4.3.2.1.1 an evaluation is performed to assure that the CRD discontinuity bounds the other discontinuities that are to be protected by the CRD curve with respect to pressure stresses (see Tables 4-7, 4-8, and Appendix A). With respect to thermal stresses, the transients evaluated for the CRD are similar to or more severe than those of the other components being bounded. Therefore, for heatup/cooldown conditions, the CRD penetration provides bounding limits.

4.3.2.1.3 Pressure Test - Non-Beltline Curve A (Using Feedwater Nozzle/Upper Vessel Region)

The stress intensity factor, K_I, for the feedwater nozzle was computed using the methods from WRC 175 [10] together with the nozzle dimension for a generic 251-inch BWR/6 feedwater nozzle. The result of that computation was K_I = 200 ksi-in$^{1/2}$ for an applied pressure of 1563 psig preservice hydrotest pressure.

The respective flaw depth and orientation used in this calculation is perpendicular to the maximum stress (hoop) at a depth of 1/4T through the corner thickness.

To evaluate the results, K_I is calculated for the upper vessel nominal stress, PR/t, according to the methods in ASME Code Appendix G (Section III or XI). The result is compared to that determined by CBIN in order to quantify the K magnification associated with the stress concentration created by the feedwater nozzles. A calculation of K_I is shown below using the BWR/6, 251-inch dimensions:

Vessel Radius, R_v	126.7 inches
Vessel Thickness, t_v	6.1875 inches
Vessel Pressure, P_v	1563 psig

Pressure stress: σ = PR / t = 1563 psig · 126.7 inches / (6.1875 inches) = 32,005 psi. The Dead weight and thermal RFE stress of 2.967 ksi is conservatively added yielding σ = 34.97 ksi. The factor F (a/r_n) from Figure A5-1 of WRC-175 is 1.4 where :

$a = \frac{1}{4}(t_n^2 + t_v^2)^{1/2}$ =2.36 inches

t_n =	thickness of nozzle	= 7.125 inches
t_v =	thickness of vessel	= 6.1875 inches
r_n =	apparent radius of nozzle	= r_i + 0.29 r_c=7.09 inches
r_i =	actual inner radius of nozzle	= 6.0 inches
r_c =	nozzle radius (nozzle corner radius)	= 3.75 inches

Thus, a/r_n = 2.36 / 7.09 = 0.33. The value $F(a/r_n)$, taken from Figure A5-1 of WRC Bulletin 175 for an a/r_n of 0.33, is 1.4. Including the safety factor of 1.5, the stress intensity factor, K_I, is $1.5\ \sigma\ (\pi a)^{1/2} \cdot F(a/r_n)$:

Nominal $K_I = 1.5\quad 34.97 \cdot (\pi \cdot 2.36)^{1/2} \cdot 1.4 = 200$ ksi-in$^{1/2}$

The method to solve for $(T - RT_{NDT})$ for a specific K_I is based on the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [8] for the pressure test condition:

$$(T - RT_{NDT}) = \ln [(K_I - 33.2) / 20.734] / 0.02$$
$$(T - RT_{NDT}) = \ln [(200 - 33.2) / 20.734] / 0.02$$
$$(T - RT_{NDT}) = 104.2°F$$

The generic pressure test P-T curve was generated by scaling 200 ksi-in$^{1/2}$ by the nominal pressures and calculating the associated $(T - RT_{NDT})$.

The highest RT_{NDT} for the feedwater nozzle materials is 0°F as shown in Table 4-2. However, the RT_{NDT} was increased to 39°F to consider the stresses in the steam outlet nozzle together with the initial RT_{NDT} as described below. The generic pressure test P-T curve is applied to the Limerick Unit 2 feedwater nozzle curve by shifting the P vs. (T - RT_{NDT}) values above to reflect the RT_{NDT} value of 39°F.

Second, the P-T curve is dependent on the K_I value calculated. The Limerick Unit 2 specific vessel shell and nozzle dimensions applicable to the feedwater nozzle location and K_I are shown below:

Vessel Radius, R_v	126.9 inches
Vessel Thickness, t_v	6.225 inches
Vessel Pressure, P_v	1563 psig

Pressure stress: $\sigma = PR / t$ = 1563 psig · 126.9 inches / (6.225 inches) = 31,863 psi. The Dead weight and thermal RFE stress of 2.967 ksi is conservatively added yielding σ = 34.83 ksi. The factor F (a/r_n) from Figure A5-1 of WRC-175 is 1.39 where:

a =	$\frac{1}{4}(t_n^2 + t_v^2)^{1/2}$	=2.37 inches
t_n =	thickness of nozzle	= 7.125 inches
t_v =	thickness of vessel	= 6.225 inches
r_n =	apparent radius of nozzle	= $r_i + 0.29\ r_c$=6.9 inches
r_i =	actual inner radius of nozzle	= 6.0 inches
r_c =	nozzle radius (nozzle corner radius)	= 2.94 inches

Thus, a/r_n = 2.37 / 6.9 = 0.34. The value F(a/r_n), taken from Figure A5-1 of WRC Bulletin 175 for an a/r_n of 0.34, is 1.4. Including the safety factor of 1.5, the stress intensity factor, K_I, is $1.5\ \sigma\ (\pi a)^{1/2} \cdot F(a/r_n)$:

Nominal $K_I = 1.5 \cdot 34.83 \cdot (\pi \cdot 2.37)^{1/2} \cdot 1.4 = 200$ ksi-in$^{1/2}$

4.3.2.1.4 Core Not Critical Heatup/Cooldown - Non-Beltline Curve B (Using Feedwater Nozzle/Upper Vessel Region)

The feedwater nozzle was selected to represent non-beltline components for fracture toughness analyses because the stress conditions are the most severe experienced in the vessel. In addition to the pressure and piping load stresses resulting from the nozzle discontinuity, the feedwater nozzle region experiences relatively cold feedwater flow in hotter vessel coolant.

Stresses were taken from a finite element analysis done specifically for the purpose of fracture toughness analysis [ ]. Analyses were performed for all feedwater nozzle transients that involved rapid temperature changes. The most severe of these was normal operation with cold 40°F feedwater injection, which is equivalent to hot standby.

The non-beltline curves based on feedwater nozzle limits were calculated according to the methods for nozzles in Appendix 5 of the Welding Research Council (WRC) Bulletin 175 [10].

The stress intensity factor for a nozzle flaw under primary stress conditions (K_{IP}) is given in WRC Bulletin 175 Appendix 5 by the expression for a flaw at a hole in a flat plate:

$$K_{IP} = SF \cdot \sigma\,(\pi a)^{1/2} \cdot F(a/r_n) \qquad (4\text{-}4)$$

where SF is the safety factor applied per WRC Bulletin 175 recommended ranges, and $F(a/r_n)$ is the shape correction factor.

Finite element analysis of a nozzle corner flaw was performed to determine appropriate values of $F(a/r_n)$ for Equation 4-4. These values are shown in Figure A5-1 of WRC Bulletin 175 [10].

The stresses used in Equation 4-4 were taken from design stress reports for the feedwater nozzle. The stresses considered are primary membrane, σ_{pm}, and primary bending, σ_{pb}. Secondary membrane, σ_{sm}, and secondary bending, σ_{sb}, stresses are included in the total K_I by using ASME Appendix G [9] methods for secondary portion, K_{Is}:

$$K_{Is} = M_m (\sigma_{sm} + (2/3) \cdot \sigma_{sb}) \quad (4\text{-}5)$$

In the case where the total stress exceeded yield stress, a plasticity correction factor was applied based on the recommendations of WRC Bulletin 175 Section 5.C.3 [10]. However, the correction was not applied to primary membrane stresses because primary stresses satisfy the laws of equilibrium and are not self-limiting. K_{IP} and K_{Is} are added to obtain the total value of stress intensity factor, K_I. A safety factor of 2.0 is applied to primary stresses for core not critical heatup/cooldown conditions.

Once K_I was calculated, the following relationship was used to determine $(T - RT_{NDT})$. The method to solve for $(T - RT_{NDT})$ for a specific K_I is based on the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [8]. The highest RT_{NDT} for the appropriate non-beltline components was then used to establish the P-T curves.

$$(T - RT_{NDT}) = \ln [(K_I - 33.2) / 20.734] / 0.02 \quad (4\text{-}6)$$

Example Core Not Critical Heatup/Cooldown Calculation for Feedwater Nozzle/Upper Vessel Region

The non-beltline core not critical heatup/cooldown curve was based on the feedwater nozzle analysis, where feedwater injection of 40°F into the vessel while at operating conditions (551.4°F and 1050 psig) was the limiting normal or upset condition from a brittle fracture perspective. The feedwater nozzle corner stresses were obtained from finite element analysis [ ]. To produce conservative thermal stresses, a vessel and nozzle thickness of 7.5 inches was used in the evaluation. However, a thickness of 7.5 inches is not conservative for the pressure stress evaluation. Therefore, the pressure stress (σ_{pm}) was adjusted for the actual vessel thickness of 6.1875 inches (i.e., σ_{pm} = 20.49 ksi was revised to 20.49 ksi · 7.5 inches/6.1875 inches = 24.84 ksi). These stresses, and other inputs used in the generic calculations, are shown below:

σ_{pm} = 24.84 ksi	σ_{sm} = 16.19 ksi	σ_{ys} = 45.0 ksi	t_v = 6.1875 inch
σ_{pb} = 0.22 ksi	σ_{sb} = 19.04 ksi	a = 2.36 inch	r_n = 7.08 inch
t_n = 7.125 inch			

In this case the total stress, 60.29 ksi, exceeds the yield stress, σ_{ys}, so the correction factor, R, is calculated to consider the nonlinear effects in the plastic region according to the following equation based on the assumptions and recommendation of WRC Bulletin 175 [10]. (The value of specified yield stress is for the material at the temperature under consideration. For conservatism, the temperature assumed for the crack root is the inside surface temperature.)

$$R = [\sigma_{ys} - \sigma_{pm} + ((\sigma_{total} - \sigma_{ys}) / 30)] / (\sigma_{total} - \sigma_{pm}) \quad (4\text{-}7)$$

For the stresses given, the ratio, R = 0.583. Therefore, all the stresses are adjusted by the factor 0.583, except for σ_{pm}. The resulting stresses are:

σ_{pm} = 24.84 ksi σ_{sm} = 9.44 ksi

σ_{pb} = 0.13 ksi σ_{sb} = 11.10 ksi

The value of M_m for an inside axial postulated surface flaw from Paragraph 2214-1 [5] was based on the 4a thickness ; hence, $t^{1/2} = 3.072$. The resulting value obtained was:

$$M_m = 1.85 \text{ for } \sqrt{t} \leq 2$$

$$\mathbf{M_m} = 0.926\ \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464 = \mathbf{2.845}$$

$$M_m = 3.21 \text{ for } \sqrt{t} > 3.464$$

The value $F(a/r_n)$, taken from Figure A5-1 of WRC Bulletin 175 for an a/r_n of 0.33, is therefore,

$F(a / r_n) = 1.4$

K_{IP} is calculated from Equation 4-4:

$K_{IP} = 2.0 \cdot (24.84 + 0.13) \cdot (\pi \cdot 2.36)^{1/2} \cdot 1.4$

$K_{IP} = 190.4$ ksi-in$^{1/2}$

K_{Is} is calculated from Equation 4-5:

$K_{Is} = 2.845 \cdot (9.44 + 2/3 \cdot 11.10)$

$K_{Is} = 47.9$ ksi-in$^{1/2}$

The total K_I is, therefore, 238.3 ksi-in$^{1/2}$.

The total K_I is substituted into Equation 4-6 to solve for $(T - RT_{NDT})$:

$(T - RT_{NDT}) = \ln[(238.3 - 33.2) / 20.734] / 0.02$

$(T - RT_{NDT}) = 115°F$

The curve was generated by scaling the stresses used to determine the K_I; this scaling was performed after the adjustment to stresses above yield. The primary stresses were scaled by the nominal pressures, while the secondary stresses were scaled by the temperature difference of the 40°F water injected into the hot reactor vessel nozzle. In the base case that yielded a K_I value of 238 ksi-in$^{1/2}$, the pressure is 1050 psig and the hot reactor vessel temperature is 551.4°F. Since the reactor vessel temperature follows the saturation temperature curve, the secondary stresses are scaled by

$(T_{saturation} - 40) / (551.4 - 40)$. From K_I the associated $(T - RT_{NDT})$ can be calculated:

Core Not Critical Feedwater Nozzle K_I and $(T - RT_{NDT})$ as a Function of Pressure

Nominal Pressure (psig)	Saturation Temp. (°F)	R	K_I* (ksi-in$^{1/2}$)	$(T - RT_{NDT})$ (°F)
1563	604	0.23	303	128
1400	588	0.34	283	124
1200	557	0.48	257	119
1050	551	0.58	238	115
1000	546	0.62	232	113
800	520	0.79	206	106
600	489	1.0	181	98
400	448	1.0	138	81

*Note: For each change in stress for each pressure and saturation temperature condition, there is a corresponding change to R that influences the determination of K_I.

The highest non-beltline RT_{NDT} for the feedwater nozzle at Limerick Unit 2 is 0°F as shown in Table 4-2. The generic curve is applied to the Limerick Unit 2 upper vessel by shifting the P vs. $(T - RT_{NDT})$ values above to reflect the RT_{NDT} value of 39°F as discussed in Section 4.3.2.1.3.

4.3.2.2 CORE BELTLINE REGION

The pressure-temperature (P-T) operating limits for the beltline region are determined according to the ASME Code. As the beltline fluence increases with the increase in operating life, the P-T curves shift to a higher temperature.

The stress intensity factors (K_I), calculated for the beltline region according to ASME Code Appendix G procedures [9], were based on a combination of pressure and thermal stresses for a 1/4T flaw in a flat plate. The pressure stresses were calculated using thin-walled cylinder equations. Thermal stresses were calculated assuming the through-wall temperature distribution of a flat plate; values were calculated for 100°F/hr coolant thermal gradient. The shift value of the most limiting ART material was used to adjust the RT_{NDT} values for the P-T limits.

4.3.2.2.1 Beltline Region - Pressure Test

The methods of ASME Code Section XI, Appendix G [9] are used to calculate the pressure test beltline limits. The vessel shell, with an inside radius (R) to minimum thickness (t_{min}) ratio of 15, is treated as a thin-walled cylinder. The maximum stress is the hoop stress, given as:

$$\sigma_m = PR / t_{min} \qquad (4\text{-}8)$$

The stress intensity factor, K_{Im}, is calculated using Paragraph 2214-1 of the ASME Code Case N-588 [5].

The calculated value of K_{Im} for pressure test is multiplied by a safety factor (SF) of 1.5, per ASME Appendix G [9] for comparison with K_{IC}, the material fracture toughness. A safety factor of 2.0 is used for the core not critical and core critical conditions.

The relationship between K_{IC} and temperature relative to reference temperature ($T - RT_{NDT}$) is based on the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [8] for the pressure test condition:

$$K_{Im} \cdot SF = K_{IC} = 20.734 \exp[0.02\,(T - RT_{NDT})] + 33.2 \qquad (4\text{-}9)$$

This relationship provides values of pressure versus temperature (from K_{IR} and ($T-RT_{NDT}$), respectively).

GE's current practice for the pressure test curve is to add a stress intensity factor, K_{It}, for a coolant heatup/cooldown rate of 20°F/hr to provide operating flexibility. For the core not critical and core critical condition curves, a stress intensity factor is added for a coolant heatup/cooldown rate of 100°F/hr. The K_{It} calculation for a coolant heatup/cooldown rate of 100°F/hr is described in Section 4.3.2.2.3 below.

4.3.2.2.2 Calculations for the Beltline Region - Pressure Test

This sample calculation is for a pressure test pressure of 1105 psig at 32 EFPY. The following inputs were used in the beltline limit calculation:

Adjusted RT_{NDT} = Initial RT_{NDT} + Shift	A = 40 + 82 = 122°F (Based on ART values in Section 4.2)
Vessel Height	H = 871 inches
Bottom of Active Fuel Height	B = 216.3 inches
Vessel Radius (to inside of clad)	R = 126.38 inches
Minimum Vessel Thickness (without clad)	t = 6.19 inches
Limiting Beltline Material Yield Strength	σ_y = 63.8 ksi

Pressure is calculated to include hydrostatic pressure for a full vessel:

$$P = 1105 \text{ psi} + (H - B)\, 0.0361 \text{ psi/inch} = P \text{ psig} \tag{4-10}$$
$$= 1105 + (871 - 216.3)\, 0.0361 = 1129 \text{ psig}$$

Pressure stress:

$$\sigma = PR/t \tag{4-11}$$
$$= 1.129 \cdot 126.38 / 6.19 = 23.1 \text{ ksi}$$

The value of M_m for an inside axial postulated surface flaw from Paragraph 2214-1 [5] was based on a thickness of 6.19 inches (the minimum thickness without cladding); hence, $t^{1/2} = 2.5$. The resulting value obtained was:

$$M_m = 1.85 \text{ for } \sqrt{t} \le 2$$

$$M_m = 0.926 \sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464 = 2.30$$

$$M_m = 3.21 \text{ for } \sqrt{t} > 3.464$$

The stress intensity factor for the pressure stress is $K_{Im} = M_m \cdot \sigma$. The stress intensity factor for the thermal stress, K_{It}, is calculated as described in Section 4.3.2.2.4 except that the value of "G" is 20°F/hr instead of 100°F/hr.

Equation 4-9 can be rearranged, and 1.5 K_{Im} substituted for K_{IC}, to solve for $(T - RT_{NDT})$. Using the K_{Ic} equation of Paragraph A-4200 in ASME Appendix A [8], K_{Im} = 53.1, and K_{It} = 2.28 for a 20°F/hr coolant heatup/cooldown rate with a vessel thickness, t, that includes cladding:

$$\begin{aligned}(T - RT_{NDT}) &= \ln[(1.5 \cdot K_{Im} + K_{It} - 33.2) / 20.734] / 0.02 \qquad (4\text{-}12)\\ &= \ln[(1.5 \cdot 53.1 + 2.28 - 33.2) / 20.734] / 0.02\\ &= 43°F\end{aligned}$$

T can be calculated by adding the adjusted RT_{NDT}:

$$T = 43 + 122 = 165°F \qquad \text{for } P = 1105 \text{ psig}$$

4.3.2.2.3 Beltline Region - Core Not Critical Heatup/Cooldown

The beltline curves for core not critical heatup/cooldown conditions are influenced by pressure stresses and thermal stresses, according to the relationship in ASME Section XI Appendix G [9]:

$$K_{IC} = 2.0 \cdot K_{Im} + K_{It} \qquad (4\text{-}13)$$

where K_{Im} is primary membrane K due to pressure and K_{It} is radial thermal gradient K due to heatup/cooldown.

The pressure stress intensity factor K_{Im} is calculated by the method described above, the only difference being the larger safety factor applied. The thermal gradient stress intensity factor calculation is described below.

The thermal stresses in the vessel wall are caused by a radial thermal gradient that is created by changes in the adjacent reactor coolant temperature in heatup or cooldown conditions. The stress intensity factor is computed by multiplying the coefficient M_t from Figure G-2214-2 of ASME Appendix G [9] by the through-wall temperature gradient ΔT_w, given that the temperature gradient has a through-wall shape similar to that shown in Figure G-2214-3 of ASME Appendix G [9]. The relationship used to compute the through-wall ΔT_w is based on one-dimensional heat conduction through an insulated flat plate:

$$\partial^2 T(x,t) / \partial x^2 = 1 / \beta \, (\partial T(x,t) / \partial t) \qquad (4\text{-}14)$$

where T(x,t) is temperature of the plate at depth x and time t, and β is the thermal diffusivity.

The maximum stress will occur when the radial thermal gradient reaches a quasi-steady state distribution, so that $\partial T(x,t) / \partial t = dT(t) / dt = G$, where G is the coolant heatup/cooldown rate, normally 100°F/hr. The differential equation is integrated over x for the following boundary conditions:

1. Vessel inside surface (x = 0) temperature is the same as coolant temperature, T_0.
2. Vessel outside surface (x = C) is perfectly insulated; the thermal gradient dT/dx = 0.

The integrated solution results in the following relationship for wall temperature:

$$T = Gx^2 / 2\beta - GCx / \beta + T_0 \qquad (4\text{-}15)$$

This equation is normalized to plot $(T - T_0) / \Delta T_w$ versus x / C.

The resulting through-wall gradient compares very closely with Figure G-2214-3 of ASME Appendix G [9]. Therefore, ΔT_w calculated from Equation 4-15 is used with the appropriate M_t of Figure G-2214-2 of ASME Appendix G [9] to compute K_{It} for heatup and cooldown.

The M_t relationships were derived in the Welding Research Council (WRC) Bulletin 175 [10] for infinitely long cracks of 1/4T and 1/8T. For the flat plate geometry and radial thermal gradient, orientation of the crack is not important.

4.3.2.2.4 Calculations for the Beltline Region Core Not Critical Heatup/Cooldown

This sample calculation is for a pressure of 1105 psig for 32 EFPY. The core not critical heatup/cooldown curve at 1105 psig uses the same K_{Im} as the pressure test curve, but with a safety factor of 2.0 instead of 1.5. The increased safety factor is used because the heatup/cooldown cycle represents an operational rather than test condition that necessitates a higher safety factor. In addition, there is a K_{It} term for the thermal stress. The additional inputs used to calculate K_{It} are:

Coolant heatup/cooldown rate, normally 100°F/hr	G = 100 °F/hr
Minimum vessel thickness, including clad thickness	C = 0.526 ft (6.315 inches)
Thermal diffusivity at 550°F (most conservative value)	$\beta = 0.354$ ft^2/ hr [28]

Equation 4-15 can be solved for the through-wall temperature (x = C), resulting in the absolute value of ΔT for heatup or cooldown of:

$$\Delta T = GC^2 / 2\beta \qquad (4\text{-}16)$$

$$= 100 \cdot (0.526)^2 / (2 \cdot 0.354) = 39°F$$

The analyzed case for thermal stress is a 1/4T flaw depth with wall thickness of C. The corresponding value of M_t (=0.2916) can be interpolated from ASME Appendix G, Figure G-2214-2 [9]. Thus the thermal stress intensity factor, $K_{It} = M_t \cdot \Delta T = 11.42$, can be calculated. K_{Im} has the same value as that calculated in Section 4.3.2.2.2.

The pressure and thermal stress terms are substituted into Equation 4-9 to solve for $(T - RT_{NDT})$:

$$\begin{aligned}(T - RT_{NDT}) &= \ln[((2 \cdot K_{Im} + K_{It}) - 33.2) / 20.734] / 0.2 \qquad (4\text{-}17)\\ &= \ln[(2 \cdot 53.1 + 11.42 - 33.2) / 20.734] / 0.02\\ &= 70\ °F\end{aligned}$$

T can be calculated by adding the adjusted RT_{NDT}:

T = 70 + 122 = 192 °F for P = 1105 psig

4.3.2.3 CLOSURE FLANGE REGION

10CFR50 Appendix G [6] sets several minimum requirements for pressure and temperature in addition to those outlined in the ASME Code, based on the closure flange region RT_{NDT}. In some cases, the results of analysis for other regions exceed these requirements and closure flange limits do not affect the shape of the P-T curves. However, some closure flange requirements do impact the curves, as is true with Limerick Unit 2 at low pressures.

The original ASME Code requirement for bolt-up was at qualification temperature (T_{30L}) plus 60°F. The ASME Code used for the currently licensed P-T curves is the 1989 ASME Code. The ASME Code requirements state in Paragraph G-2222(c) that, for application of full bolt preload and reactor pressure up to 20% of hydrostatic test pressure, the RPV metal temperature must be at RT_{NDT} or greater. The approach used for Limerick Unit 2 for the bolt-up temperature was based on a more conservative value of (RT_{NDT} + 60), or the LST of the bolting materials, whichever is greater. The 60°F adder is included by GE for two reasons: 1) the pre-1971 requirements of the ASME Code Section III, Subsection NA, Appendix G included the 60°F adder, and 2) inclusion of the additional 60°F requirement above the RT_{NDT} provides the additional assurance that a flaw size between 0.1 and 0.24 inches is acceptable. As shown in Tables 4-1 and 4-3, the limiting initial RT_{NDT} for the closure flange region was represented by both the top head and vessel shell flange materials at 10°F, and the LST of the closure studs was 10°F; therefore, the bolt-up temperature value used was 70°F. This conservatism is appropriate because bolt-up is one of the more limiting operating conditions (high stress and low temperature) for brittle fracture.

10CFR50 Appendix G, paragraph IV.A.2 [6] including Table 1, sets minimum temperature requirements for pressure above 20% hydrotest pressure based on the RT_{NDT} of the closure region. Curve A temperature must be no less than (RT_{NDT} + 90°F) and Curve B temperature no less than (RT_{NDT} + 120°F).

For pressures below 20% of preservice hydrostatic test pressure (312 psig) and with full bolt preload, the closure flange region metal temperature is required to be at RT_{NDT} or greater as described above. At low pressure, the ASME Code [9] allows the bottom head regions to experience even lower metal temperatures than the flange region RT_{NDT}. However, temperatures should not be permitted to be lower than 68°F for the reason discussed below.

The shutdown margin, provided in the Limerick Unit 2 Technical Specification, is calculated for a water temperature of 68°F. Shutdown margin is the quantity of reactivity needed for a reactor core to reach criticality with the strongest-worth control rod fully withdrawn and all other control rods fully inserted. Although it may be possible to safely allow the water temperature to fall below this 68°F limit, further extensive calculations would be required to justify a lower temperature. The 70°F limit applies when the head is on and tensioned and the 68°F limit for the bottom head curve and when the head is off, while fuel is in the vessel. When the head is not tensioned and fuel is not in the vessel, the requirements of 10CFR50 Appendix G [6] do not apply, and there are no limits on the vessel temperatures.

4.3.2.4 CORE CRITICAL OPERATION REQUIREMENTS OF 10CFR50, APPENDIX G

Curve C, the core critical operation curve, is generated from the requirements of 10CFR50 Appendix G [6], Table 1. Table 1 of [6] requires that core critical P-T limits be 40°F above any Curve A or B limits when pressure exceeds 20% of the pre-service system hydrotest pressure. Curve B is more limiting than Curve A, so limiting Curve C values are at least Curve B plus 40°F for pressures above 312 psig.

Table 1 of 10CFR50 Appendix G [6] indicates that for a BWR with water level within normal range for power operation, the allowed temperature for initial criticality at the closure flange region is (RT_{NDT} + 60°F) at pressures below 312 psig. This requirement makes the minimum criticality temperature 70°F, based on an RT_{NDT} of 10°F. In addition, above 312 psig the Curve C temperature must be at least the greater of RT_{NDT} of the closure region + 160°F or the temperature required for the hydrostatic pressure test (Curve A at 1105 psig). The requirement of closure region RT_{NDT} + 160°F does cause a temperature shift in Curve C at 312 psig.

5.0 CONCLUSIONS AND RECOMMENDATIONS

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) non-nuclear heatup/cooldown and low-level physics tests, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

There are four vessel regions that should be monitored against the P-T curve operating limits; these regions are defined on the thermal cycle diagram [2]:

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

For the core not critical and the core critical curve, the P-T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram [2] and the nozzle thermal cycle diagrams [3]. For the hydrostatic pressure and leak test curve, a coolant heatup and cooldown temperature rate of 20°F/hr or less must be maintained at all times.

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

The following P-T curves were generated for Limerick Unit 2.

- Composite P-T curves were generated for each of the Pressure Test and Core Not Critical conditions at 32 effective full power years (EFPY). The composite curves were generated by enveloping the most restrictive P-T limits from the separate beltline, upper vessel and closure assembly P-T limits. A separate Bottom Head Limits (CRD Nozzle) curve is also individually included with the composite curve for the Pressure Test and Core Not Critical condition.
- Separate P-T curves were developed for the upper vessel, beltline (at 22 and 32 EFPY), and bottom head for the Pressure Test and Core Not Critical conditions.
- A composite P-T curve was also generated for the Core Critical condition at 22 and 32 EFPY. The composite curves were generated by enveloping the most restrictive P-T limits from the separate beltline, upper vessel, bottom head, and closure assembly P-T limits.

Table 5-1 shows the figure numbers for each P-T curve. A tabulation of the curves is presented in Appendix B.

TABLE 5-1: COMPOSITE AND INDIVIDUAL CURVES USED TO CONSTRUCT COMPOSITE P-T CURVES

Curve	Curve Description	Figure Numbers for Presentation of the P-T Curves	Table Numbers for Presentation of the P-T Curves
Curve A			
	Bottom Head Limits (CRD Nozzle)	Figure 5-1	Table B-1 & 3
	Upper Vessel Limits (FW Nozzle)	Figure 5-2	Table B-1 & 3
	Beltline Limits for 22 EFPY	Figure 5-3	Table B-3
	Beltline Limits for 32 EFPY	Figure 5-4	Table B-1
Curve B			
	Bottom Head Limits (CRD Nozzle)	Figure 5-5	Table B-1 & 3
	Upper Vessel Limits (FW Nozzle)	Figure 5-6	Table B-1 & 3
	Beltline Limits for 22 EFPY	Figure 5-7	Table B-3
	Beltline Limits for 32 EFPY	Figure 5-8	Table B-1
Curve C			
	Composite Curve for 22 EFPY**	Figure 5-9	Table B-4
A, B, & C	**Composite Curves for 32 EFPY**		
	Bottom Head and Composite Curve A for 32 EFPY*	Figure 5-10	Table B-2
	Bottom Head and Composite Curve B for 32 EFPY*	Figure 5-11	Table B-2
	Composite Curve C for 32 EFPY**	Figure 5-12	Table B-2
	Limiting A,B&C Curves for 32 EFPY and A Curve for 22 EFPY	Figure 5-13	Table B-5

* The Composite Curve A & B curve is the more limiting of three limits: 10CFR50 Bolt-up Limits, Upper Vessel Limits (FW Nozzle), and Beltline Limits. A separate Bottom Head Limits (CRD Nozzle) curve is individually included on this figure.

** The Composite Curve C curve is the more limiting of four limits: 10CFR50 Bolt-up Limits, Bottom Head Limits (CRD Nozzle), Upper Vessel Limits (FW Nozzle), and Beltline Limits.

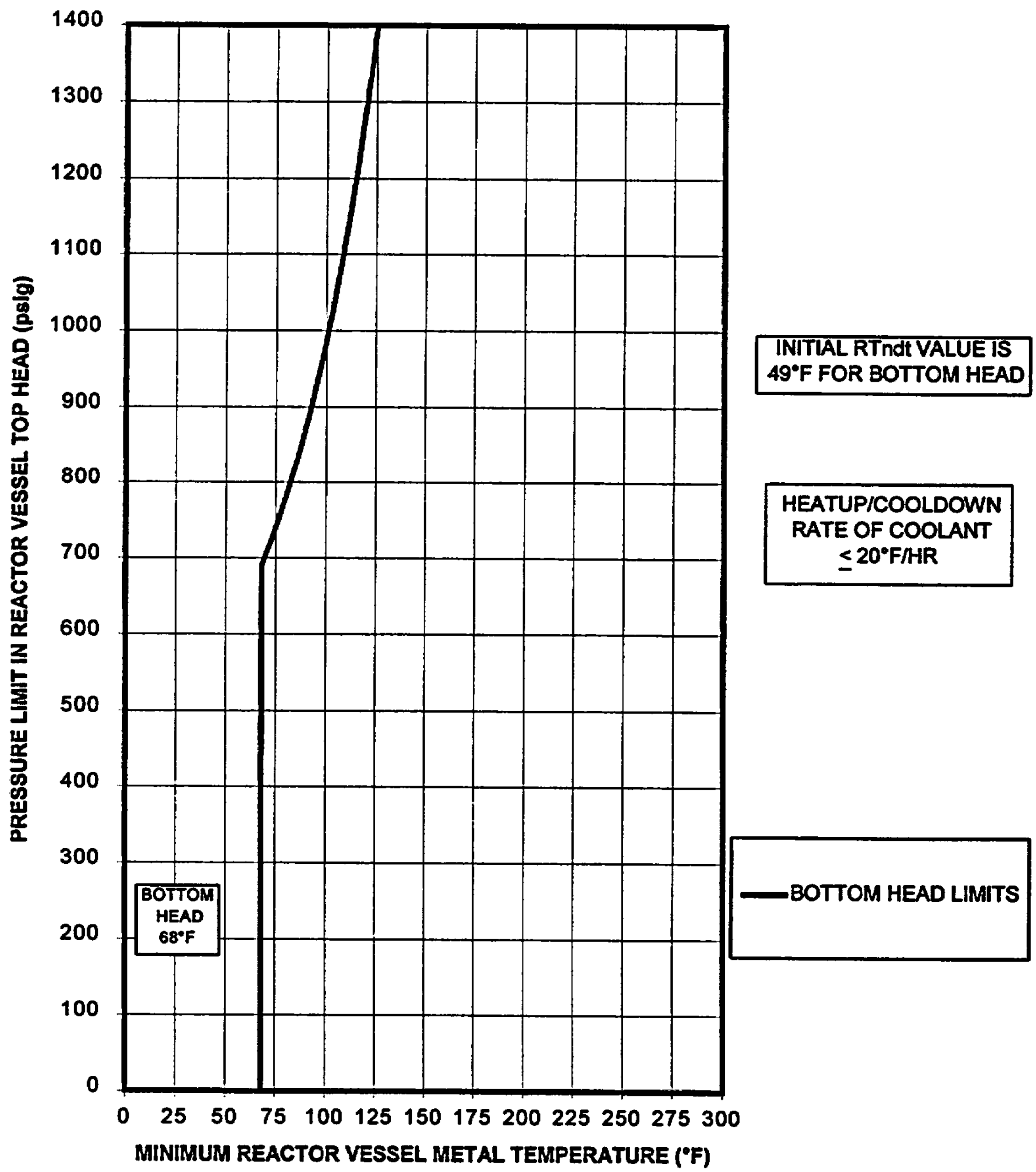

Figure 5-1: Bottom Head P-T Curve for Pressure Test [Curve A]

[20°F/hr or less coolant heatup/cooldown]

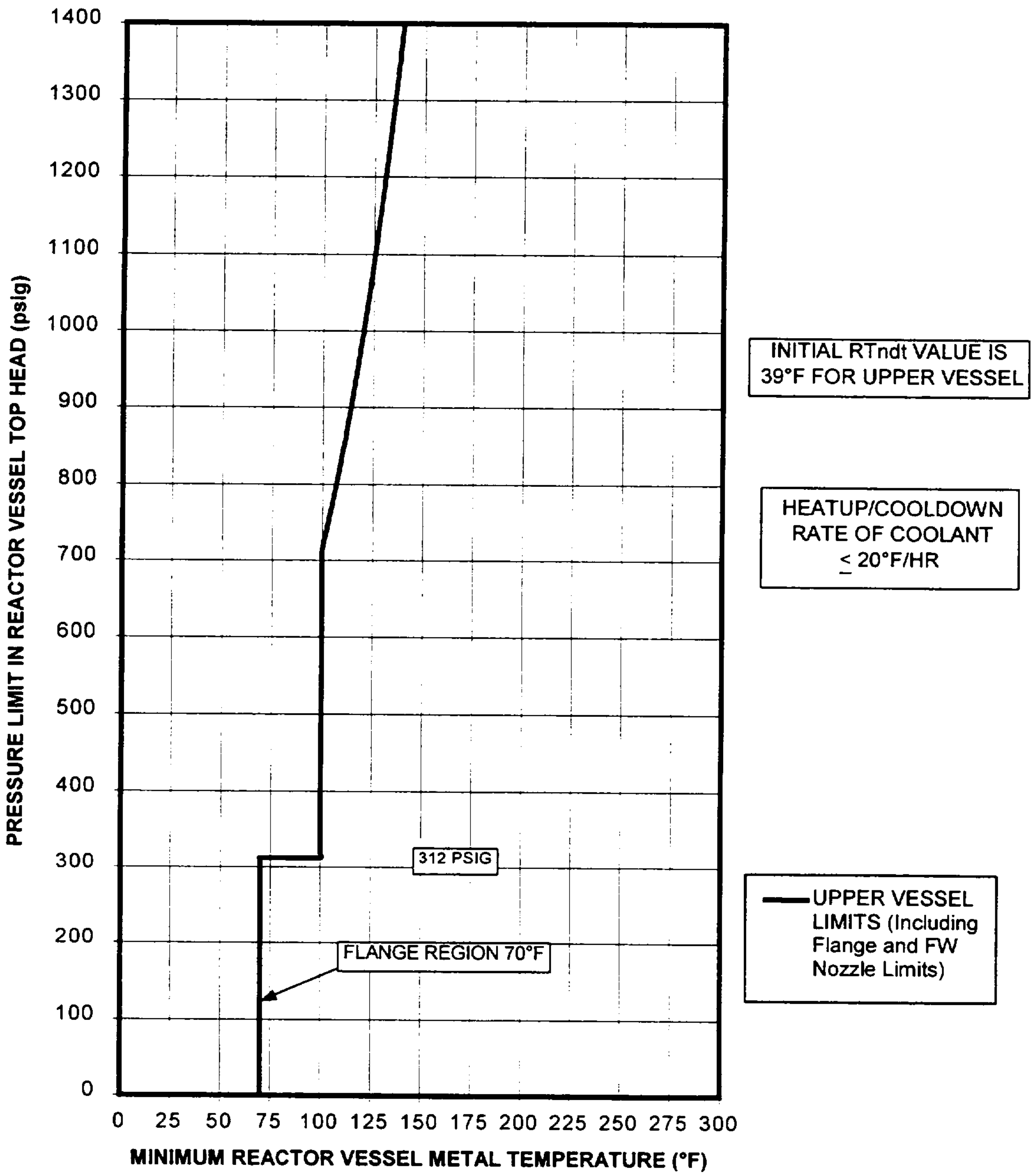

Figure 5-2: Upper Vessel P-T Curve for Pressure Test [Curve A]

[20°F/hr or less coolant heatup/cooldown]

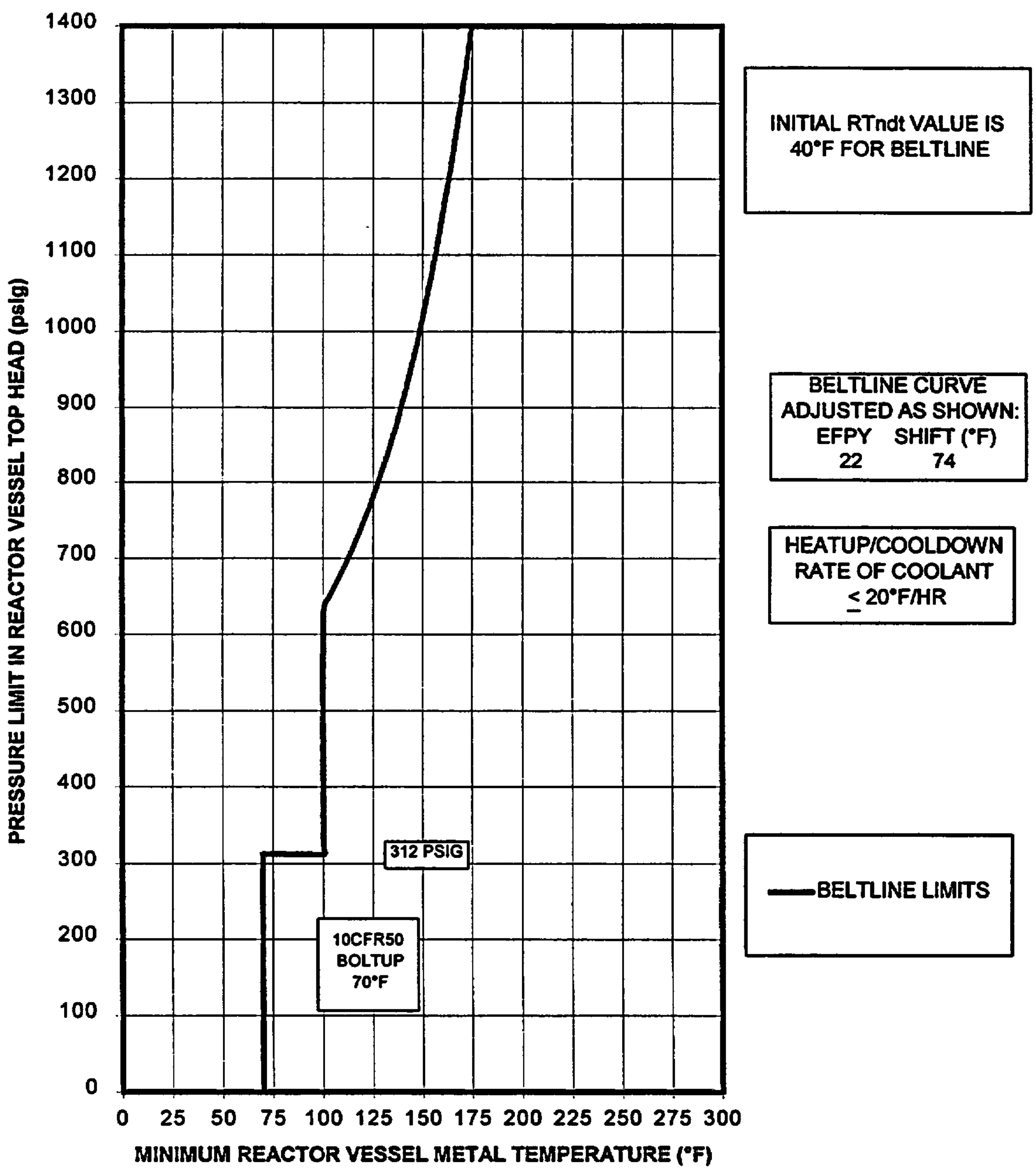

Figure 5-3: Beltline P-T Curve for Pressure Test [Curve A] up to 22 EFPY [20°F/hr or less coolant heatup/cooldown]

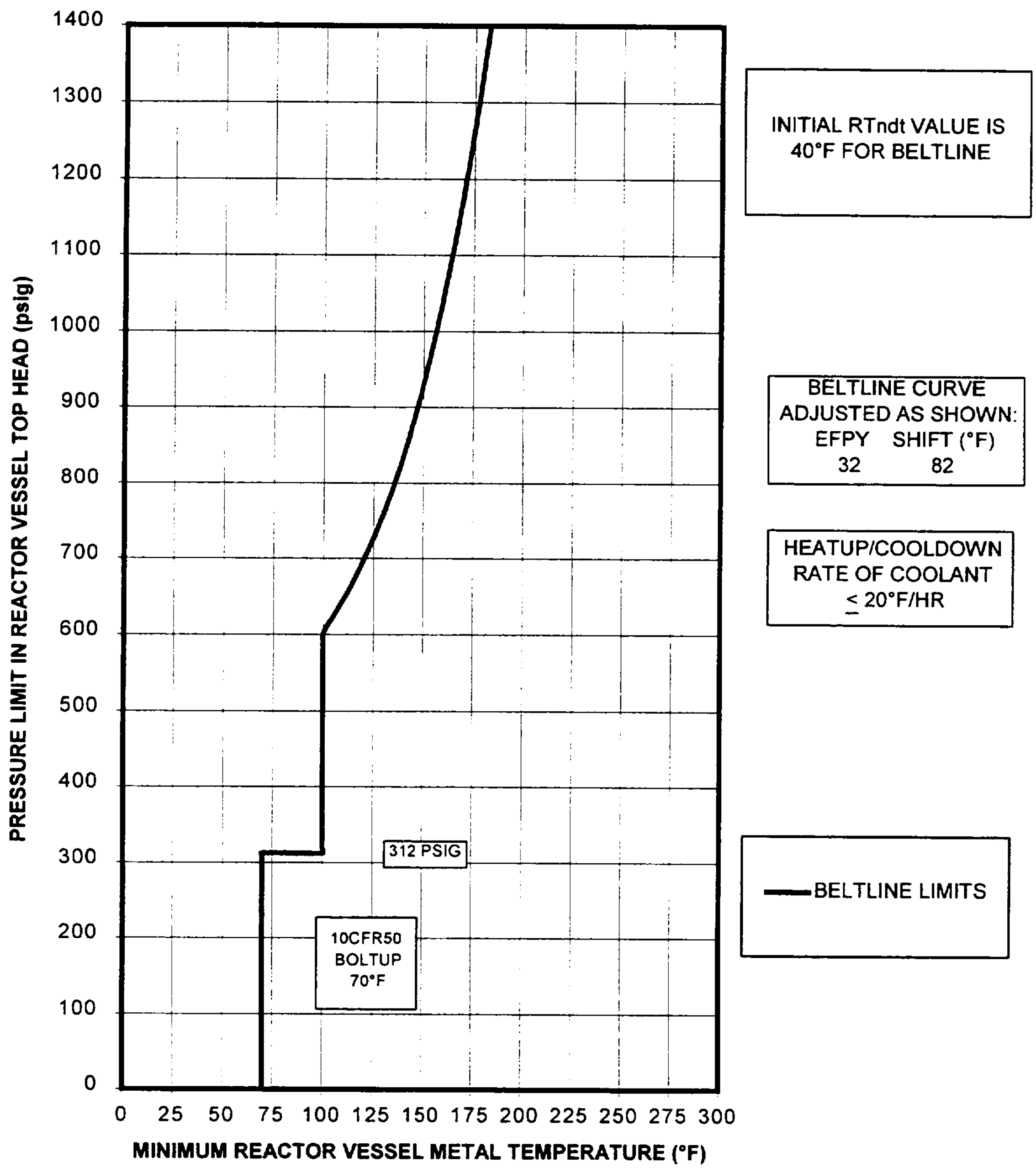

Figure 5-4: Beltline P-T Curve for Pressure Test [Curve A] up to 32 EFPY

[20°F/hr or less coolant heatup/cooldown]

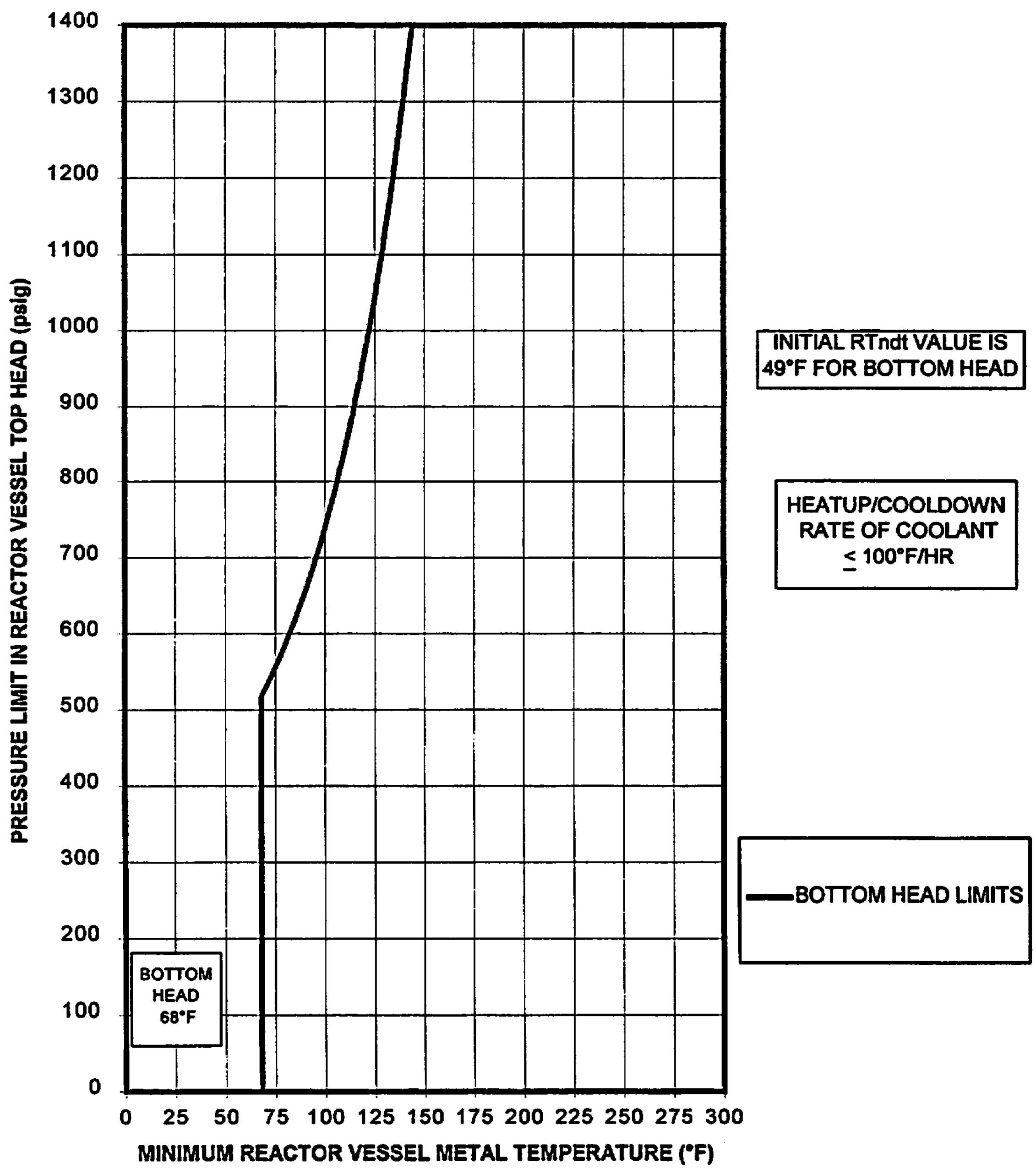

Figure 5-5: Bottom Head P-T Curve for Core Not Critical [Curve B]

[100°F/hr or less coolant heatup/cooldown]

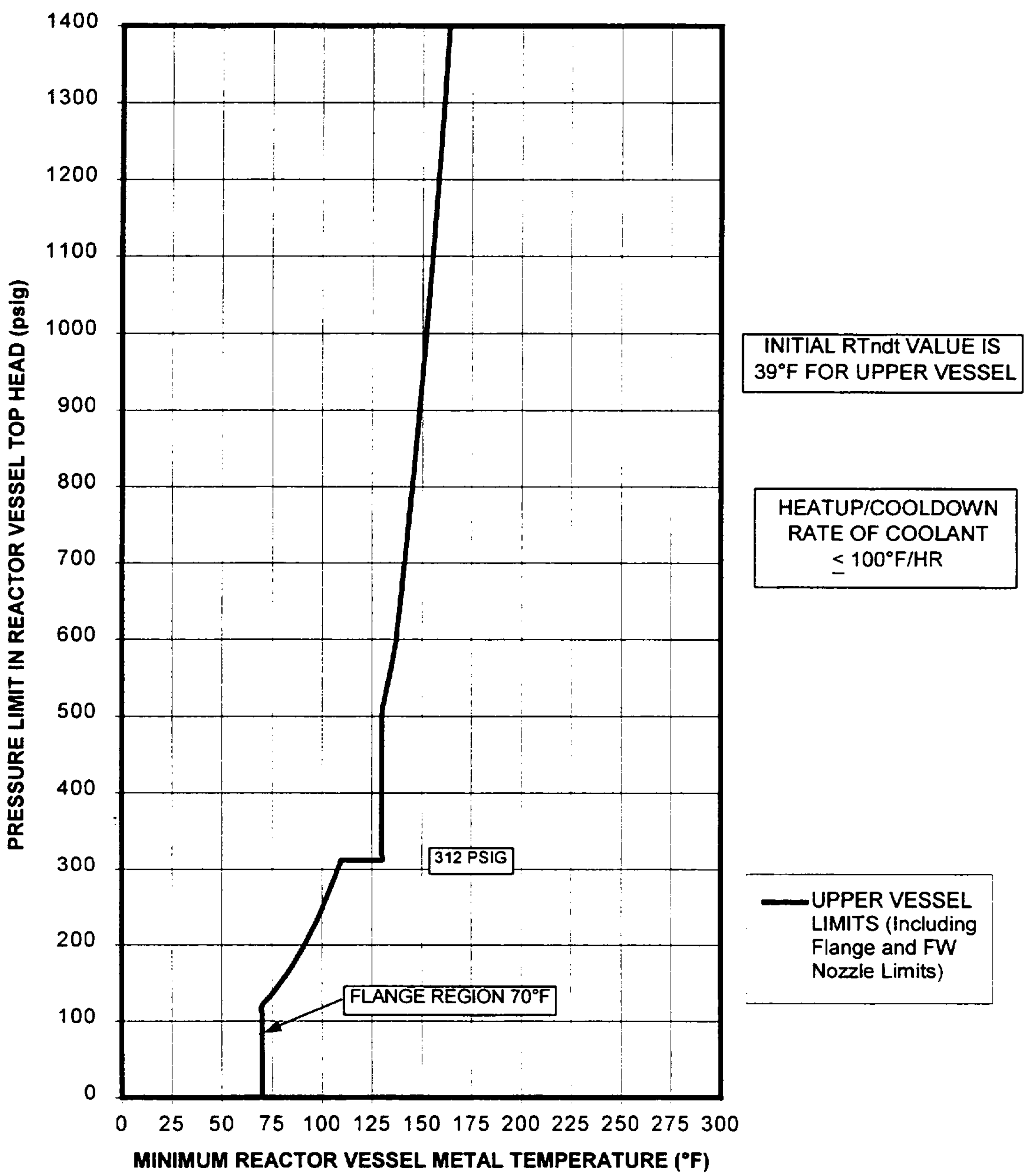

Figure 5-6: Upper Vessel P-T Curve for Core Not Critical [Curve B]

[100°F/hr or less coolant heatup/cooldown]

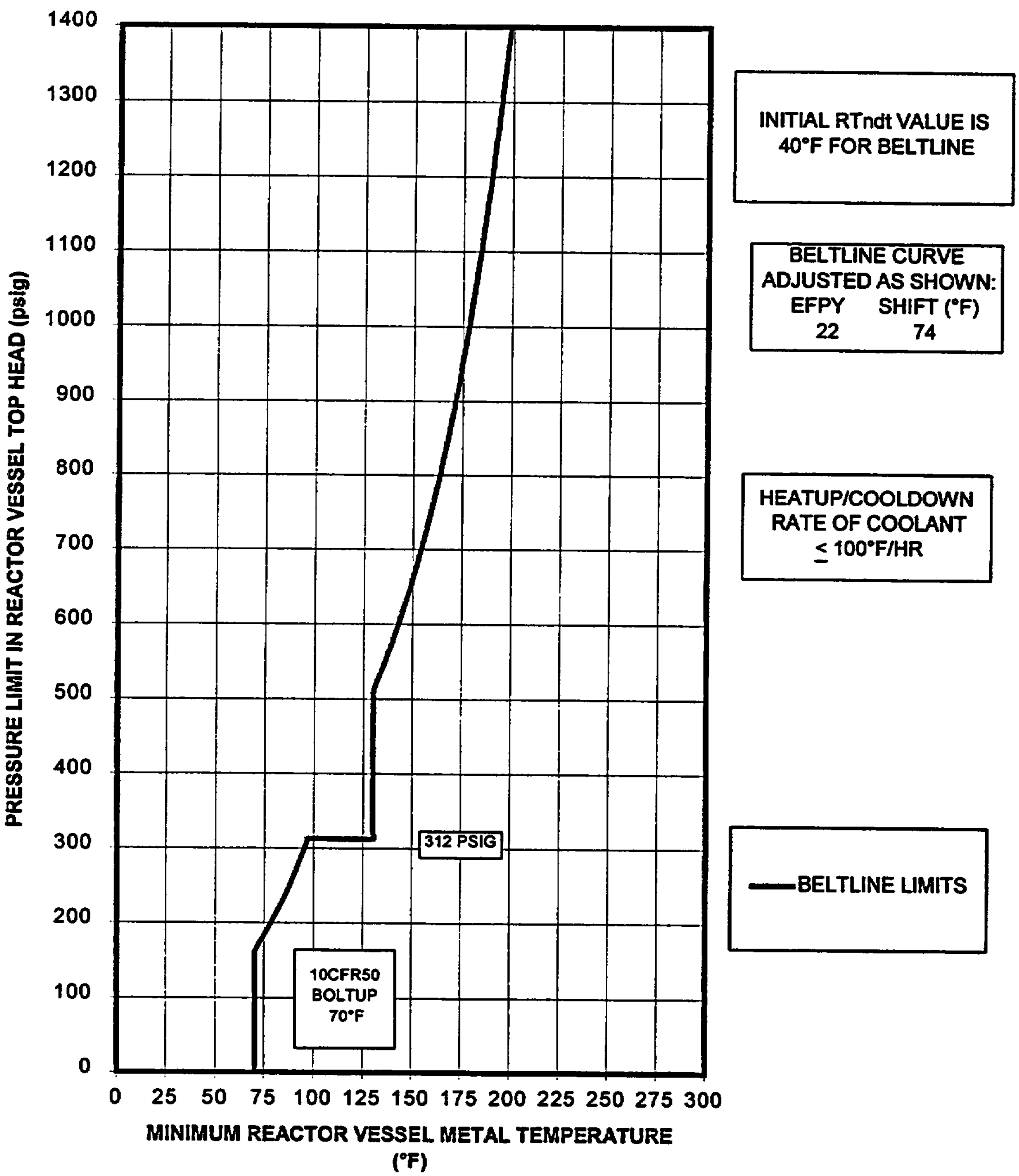

Figure 5-7: Beltline P-T Curve for Core Not Critical [Curve B] up to 22 EFPY [100°F/hr or less coolant heatup/cooldown]

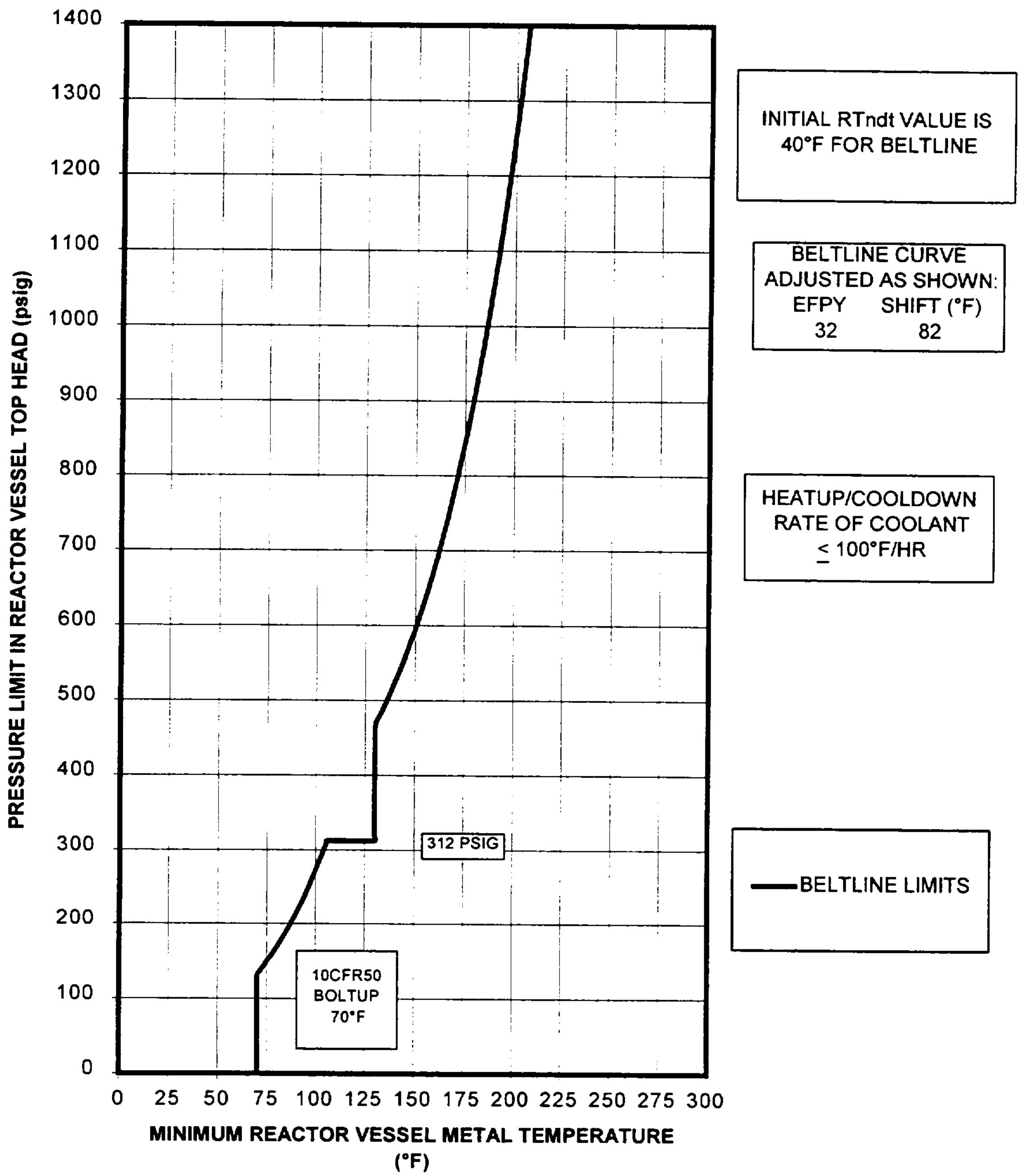

Figure 5-8: Beltline P-T Curves for Core Not Critical [Curve B] up to 32 EFPY [100°F/hr or less coolant heatup/cooldown]

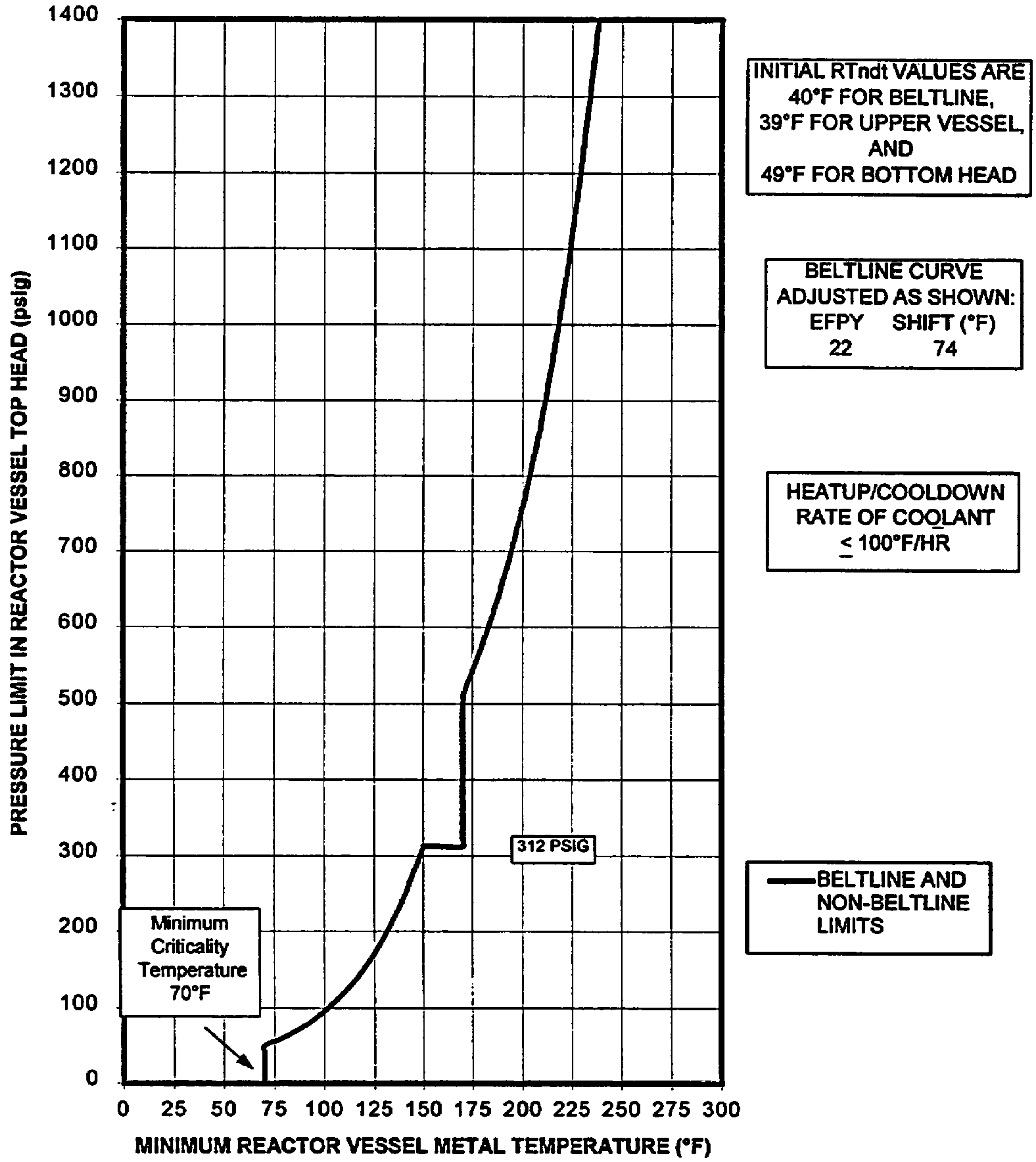

Figure 5-9: Core Critical P-T Curves [Curve C] up to 22 EFPY

[100°F/hr or less coolant heatup/cooldown]

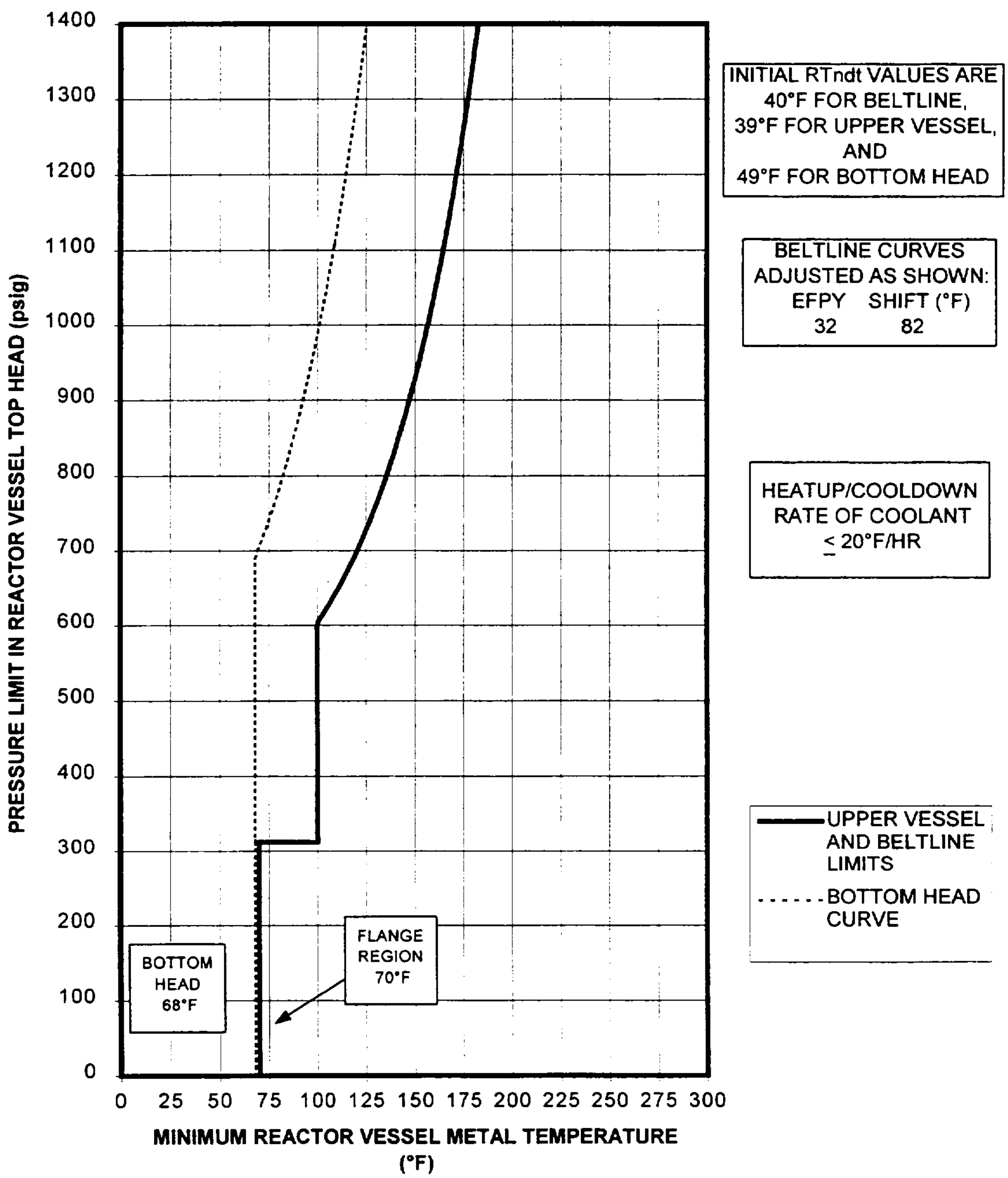

Figure 5-10: Pressure Test P-T Curves [Curve A] up to 32 EFPY

[20°F/hr or less coolant heatup/cooldown]

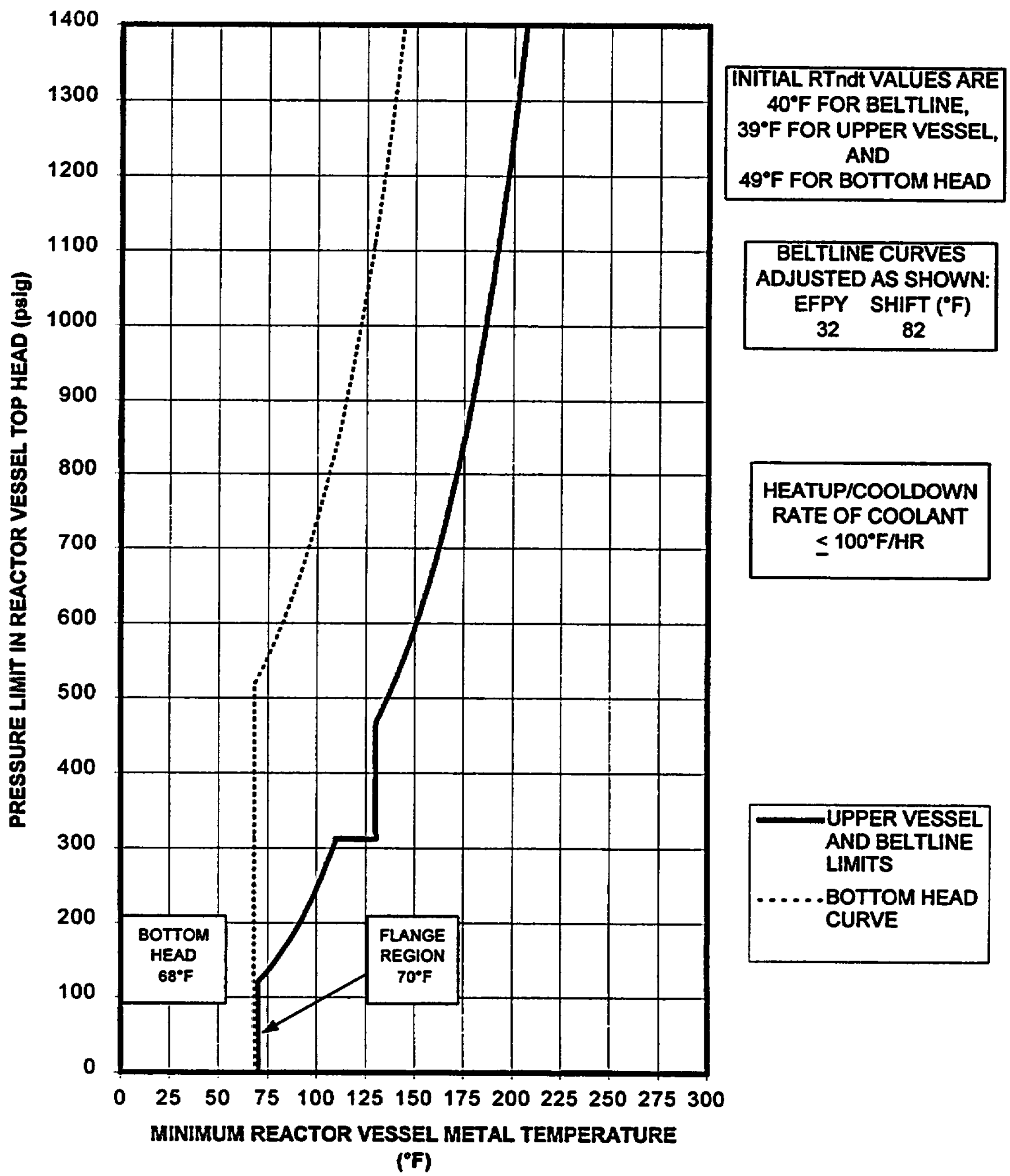

Figure 5-11: Core Not Critical P-T Curves [Curve B] up to 32 EFPY

[100°F/hr or less coolant heatup/cooldown]

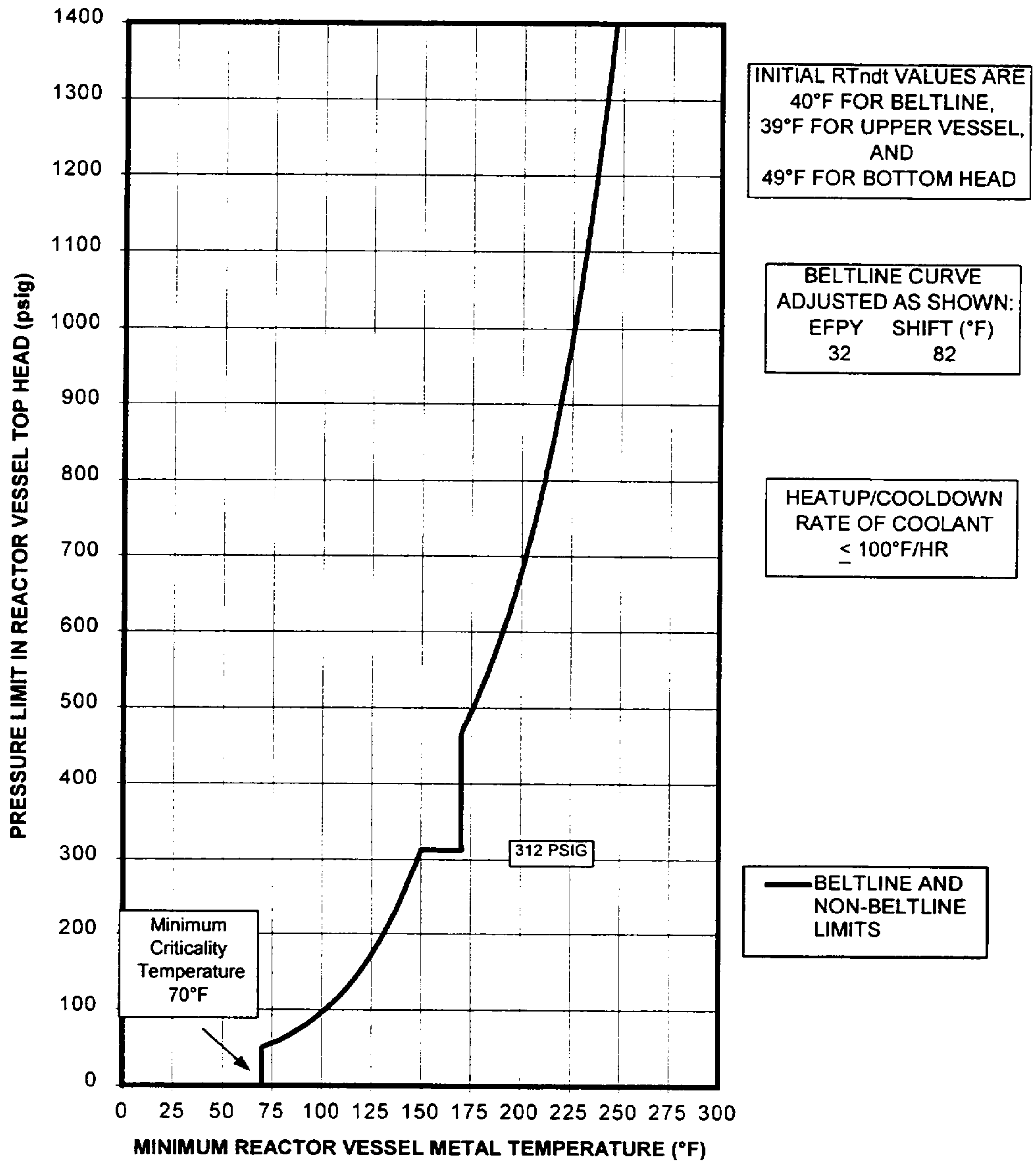

Figure 5-12: Core Critical P-T Curves [Curve C] up to 32 EFPY

[100°F/hr or less coolant heatup/cooldown]

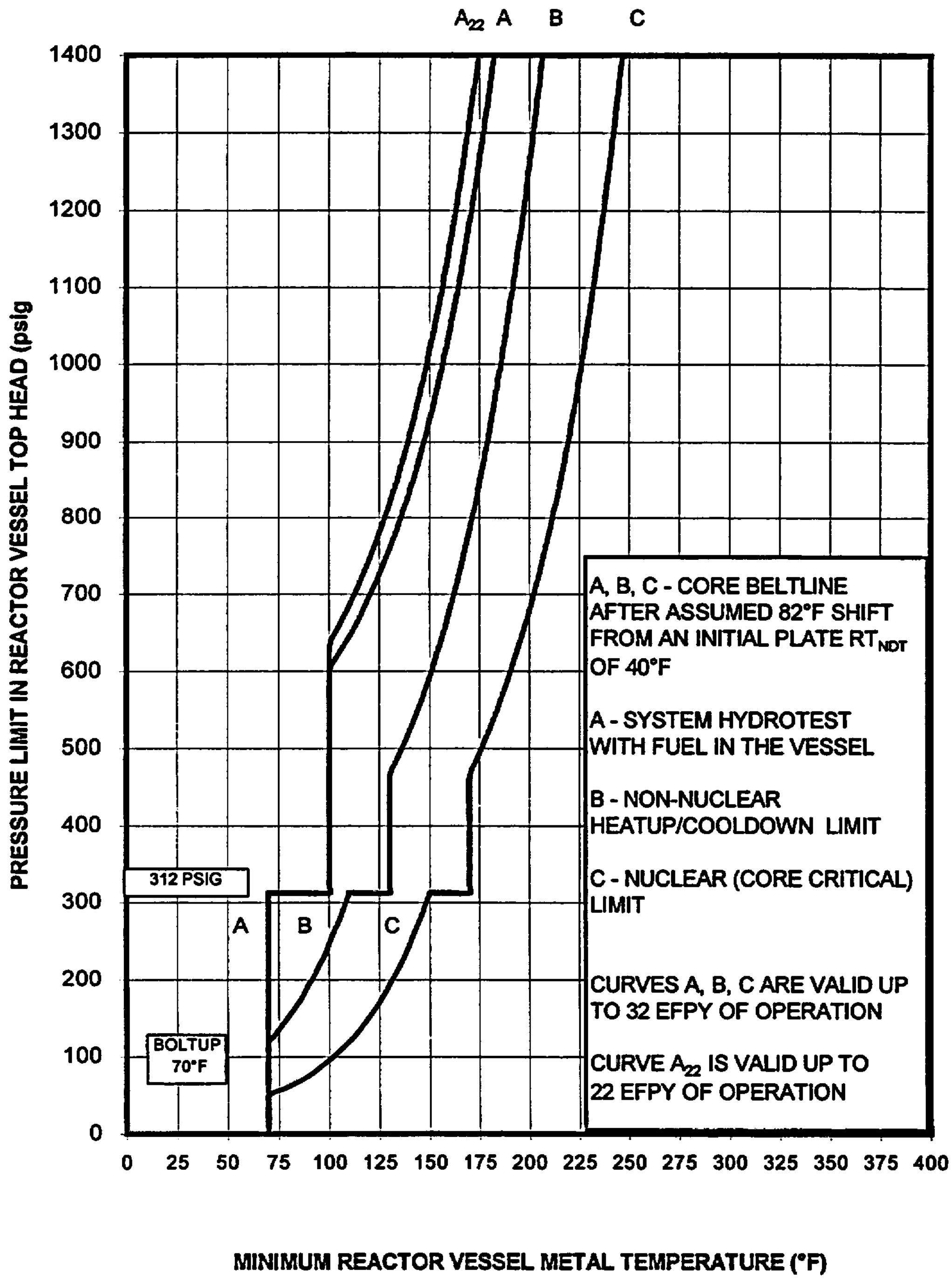

Figure 5-13: Limiting Pressure-Temperature Curves (A,B,C) up to 32 EFPY and Pressure Test Curve (A_{22}) up to 22 EFPY

6.0 REFERENCES

1. "Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 and 2," GE-NE, San Jose, CA, September 1993, (NEDC-32225P)
2. GE Drawing Number 761E708, "Reactor Vessel Thermal Cycles," GE-APED, San Jose, CA, Revision 0. Limerick RPV Thermal Cycle Diagram (GE Proprietary).
3. GE Drawing Number 158B8827, "Nozzle Thermal Cycles," GE-APED, San Jose, CA, Revision 7. Limerick Nozzle Thermal Cycle Diagram (GE Proprietary).
4. Alternative to Reference Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1," Code Case N-640 of the ASME Boiler & Pressure Vessel Code, Approval Date February 26, 1999
5. "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1," Code Case N-588 of the ASME Boiler & Pressure Vessel Code, Approval Date December 12, 1997.
6. "Fracture Toughness Requirements," Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.
7. "Radiation Embrittlement of Reactor Vessel Materials," USNRC Regulatory Guide 1.99, Revision 2, May 1988.
8. "Analysis of Flaws," Appendix A to Section XI of the ASME Boiler & Pressure Vessel Code, 1995 Edition with addenda through 1996.
9. "Fracture Toughness Criteria for Protection Against Failure," Appendix G to Section III or XI of the ASME Boiler & Pressure Vessel Code, 1995 Edition with addenda through 1996.
10. "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council Bulletin 175, August 1972.

11. Hodge, J. M., "Properties of Heavy Section Nuclear Reactor Steels," Welding Research Council Bulletin 217, July 1976.

12. GE Nuclear Energy, NEDC-32399-P, "Basis for GE RT_{NDT} Estimation Method," Report for BWR Owners' Group, San Jose, California, September 1994 (GE Proprietary).

13. Letter from B. Sheron to R.A. Pinelli,"Safety Assessment of Report NEDC-32399-P, Basis for GE RT_{NDT} Estimation Method, September 1994, " USNRC, December 16, 1994.

14. QA Records & RPV CMTR's: Limerick Unit 2 - (QA Records & RPV CMTR's Limerick Unit 2 GE PO# 205-AC122, Manufactured by CBIN), CBI Contract No. 69-5402.

15. "Response to NRC Request for Additional Information Regarding Reactor Pressure Vessel Structural Integrity at Limerick Generating Station Units 1 & 2", SIR-98-079, Revision A, July 1998.

16. Not used.

17. Not used.

18. Not used.

19. Not used.

20. Not used.

21. Not used.

22. Not used.

23. Not used.

24.

25.

26. CBI Stress Report, "Appendix C9 - 251" BWR/6 Vessel, Fracture Toughness Calculations for the Code 1563 psig System Hydrostatic Test," CBI Nuclear Company, (GE VPF 3519-735-2).

27.

28. CBI Stress Report, "Analysis of the Nozzle to Shell Junction for 10" Welded Thermal Sleeve Nozzles," CBI Nuclear Company, (GE VPF 3521-415-5 & 6).

29. "Materials - Properties," Part D to Section II of the ASME Boiler & Pressure Vessel Code, 1995 Edition with Addenda through 1996.

APPENDIX A

DESCRIPTION OF DISCONTINUITIES

Table A-2 - Geometric Discontinuities Not Requiring Fracture Toughness Evaluations

Per ASME Code Appendix G, Section G2223 (c), fracture toughness analysis to demonstrate protection against non-ductile failure is not required for portions of nozzles and appurtenances having a thickness of 2.5" or less provided the lowest service temperature is not lower than RT_{NDT} plus 60°F. Components not requiring a fracture toughness evaluation are listed below:

Nozzle or Appurtenance Identification	Nozzle or Appurtenance	Material	Reference	Remarks
Nozzle N10	Core Differential Pressure & Liquid Poison – Penetration ≤ 2.5"	SB-166	1, 16 & 18	Nozzles or appurtenances ≤ 2.5" or made from Alloy 600 require no fracture toughness evaluation.
Nozzle N11	Instrumentation < 2.5"	SB-166	1 & 9	Nozzles or appurtenances ≤ 2.5" or made from Alloy 600 require no fracture toughness evaluation.
Nozzle N12	Instrumentation < 2.5"	SB-166	1 & 9	Nozzles or appurtenances ≤ 2.5" or made from Alloy 600 require no fracture toughness evaluation.
Nozzle N13	High and Low Pressure Seal Leak Detection * - Penetration < 2.5" Flange	SB-166	1, 9 & 18	Not a pressure boundary component; therefore, requires no fracture toughness evaluation.
Nozzle N15	Drain- Penetration ≤ 2.5" – Bottom Head	SA-508-Class 1	1 & 9	The discontinuity of the CRD nozzle listed in Table A-1 bounds this discontinuity; therefore, no further fracture toughness evaluation is required.
Nozzle N16	Instrumentation < 2.5"	SB-166	1 & 9	Nozzles or appurtenances ≤ 2.5" or made from Alloy 600 require no fracture toughness evaluation.
	Top Head Lifting Lugs	SA-533 GR. B, Class 1	1 & 17	Not a pressure boundary component and loads only occur on this component when the reactor is shutdown during an outage. Therefore, no fracture toughness evaluation is required.

* The high/low pressure leak detector, and the seal leak detector are the same nozzle, these nozzles are the closure flange leak detection nozzles.

APPENDIX A REFERENCES:

1. RPV Outline or Subassembly
(a.) Chicago Bridge and Iron Co. (CBIN) Drawing # 1, Revision 10, "General Plan," (GE-NE VPF# 2886-72) - Limerick Units 1 and 2
(b.) Chicago Bridge and Iron Co. (CBIN) Drawing # R14, Revision 7, "Vessel Nozzle and Outside Bracket As-Built Dimensions," (GE-NE VPF# 2886-318) - Limerick Unit 2
(c.) Chicago Bridge and Iron Co. (CBIN) Drawing # R15, Revision 5, "Inside Brackets and Refueling Bellows Support As-Built Dimensions," (GE-NE VPF# 2886-319) - Limerick Unit 2

2. Limerick Generating Station Units 1 and 2 Updated Final Safety Analysis Report, Figure 5.3-7, "Beltline Plate and Weld Seam Locations (Unit 2)"

3. Stress Report, "Nozzle N17 LPCI Nozzle Limerick Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-718-1)

4. Stress Report, "Section T9 Thermal Analysis of Recirculation Outlet Nozzle of 251" BWR Vessel Contract 69-5401/02", (GE-NE VPF#2886-714-1)

5. Stress Report, "Section E8 Special Stress Report 251" BWR Vessel Recirculation Inlet Nozzle", (GE-NE VPF#2886-736-1)

6. Stress Report, "Revision 'C' Steam Outlet Nozzle Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-715-1)

7. Stress Report, "Feedwater Nozzle N-4 Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-709-1)

8. Stress Report, "Core Spray Nozzle N-5 Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-712-1)

9. Stress Report, "Revision 'B' Miscellaneous Nozzles Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-716-1)

10. Stress Report, "CRDHSR Nozzle with Safe End and Thermal Sleeve Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122 (for information only)", (GE-NE VPF#2886-493-1)

11. Stress Report, "Capped CRD-HSR Nozzle Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-711-1)

12. Stress Report, "CRD Penetrations Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122", (GE-NE VPF#2886-710-1)

13. Stress Report, “Stabilizer Bracket & Steam-Water Interface Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122”, (GE-NE VPF#2886-721-1)

14. Stress Report, “Vessel Skirt Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122”, (GE-NE VPF#2886-707-1)

15. Stress Report, “Shroud Support Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122”, (GE-NE VPF#2886-708-1)

16. Stress Report, “Nozzle N-10 Core Differential Pressure and Liquid Control Nozzle Limerick Reactor Vessel CBI Contract 69-5401/02”, (GE-NE VPF#2886-719-1)

17. Stress Report, “Section S-20 Stress Analysis Miscellaneous Brackets Limerick I & II Reactor Vessels CBI Contract 69-5401/02 GE Purchase Order 205-AC122”, (GE-NE VPF#2886-722-1)

18. QA Records & RPV CMTR's: Limerick Unit 2 - (QA Records & RPV CMTR's Limerick Unit 2 GE PO# 205-AC122, Manufactured by CBIN), CBI Contract No. 69-5401

APPENDIX B

PRESSURE TEMPERATURE CURVE DATA TABULATION

TABLE B-1. Limerick Unit 2 P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-4, 5-5, 5-6 AND 5-8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	32 EFPY BELTLINE CURVE B (°F)
0	68.0	70.0	70.0	68.0	70.0	70.0
10	68.0	70.0	70.0	68.0	70.0	70.0
20	68.0	70.0	70.0	68.0	70.0	70.0
30	68.0	70.0	70.0	68.0	70.0	70.0
40	68.0	70.0	70.0	68.0	70.0	70.0
50	68.0	70.0	70.0	68.0	70.0	70.0
60	68.0	70.0	70.0	68.0	70.0	70.0
70	68.0	70.0	70.0	68.0	70.0	70.0
80	68.0	70.0	70.0	68.0	70.0	70.0
90	68.0	70.0	70.0	68.0	70.0	70.0
100	68.0	70.0	70.0	68.0	70.0	70.0
110	68.0	70.0	70.0	68.0	70.0	70.0
120	68.0	70.0	70.0	68.0	70.0	70.0
130	68.0	70.0	70.0	68.0	73.2	70.0
140	68.0	70.0	70.0	68.0	76.4	72.4
150	68.0	70.0	70.0	68.0	79.2	75.2
160	68.0	70.0	70.0	68.0	81.9	77.9
170	68.0	70.0	70.0	68.0	84.5	80.5
180	68.0	70.0	70.0	68.0	86.9	82.9
190	68.0	70.0	70.0	68.0	89.2	85.2
200	68.0	70.0	70.0	68.0	91.3	87.3
210	68.0	70.0	70.0	68.0	93.3	89.3
220	68.0	70.0	70.0	68.0	95.3	91.3
230	68.0	70.0	70.0	68.0	97.1	93.1
240	68.0	70.0	70.0	68.0	98.9	94.9
250	68.0	70.0	70.0	68.0	100.6	96.6
260	68.0	70.0	70.0	68.0	102.2	98.2

TABLE B-1. Limerick Unit 2 P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-4, 5-5, 5-6 AND 5-8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	32 EFPY BELTLINE CURVE B (°F)
270	68.0	70.0	70.0	68.0	103.8	99.8
280	68.0	70.0	70.0	68.0	105.3	101.3
290	68.0	70.0	70.0	68.0	106.8	102.8
300	68.0	70.0	70.0	68.0	108.2	104.2
310	68.0	70.0	70.0	68.0	109.5	105.5
312.5	68.0	70.0	70.0	68.0	109.9	105.9
312.5	68.0	100.0	100.0	68.0	130.0	130.0
320	68.0	100.0	100.0	68.0	130.0	130.0
330	68.0	100.0	100.0	68.0	130.0	130.0
340	68.0	100.0	100.0	68.0	130.0	130.0
350	68.0	100.0	100.0	68.0	130.0	130.0
360	68.0	100.0	100.0	68.0	130.0	130.0
370	68.0	100.0	100.0	68.0	130.0	130.0
380	68.0	100.0	100.0	68.0	130.0	130.0
390	68.0	100.0	100.0	68.0	130.0	130.0
400	68.0	100.0	100.0	68.0	130.0	130.0
410	68.0	100.0	100.0	68.0	130.0	130.0
420	68.0	100.0	100.0	68.0	130.0	130.0
430	68.0	100.0	100.0	68.0	130.0	130.0
440	68.0	100.0	100.0	68.0	130.0	130.0
450	68.0	100.0	100.0	68.0	130.0	130.0
460	68.0	100.0	100.0	68.0	130.0	130.0
470	68.0	100.0	100.0	68.0	130.0	130.6
480	68.0	100.0	100.0	68.0	130.0	132.5
490	68.0	100.0	100.0	68.0	130.0	134.3
500	68.0	100.0	100.0	68.0	130.0	136.1
510	68.0	100.0	100.0	68.0	130.4	137.8
520	68.0	100.0	100.0	68.2	131.2	139.4
530	68.0	100.0	100.0	70.2	132.0	141.0

TABLE B-1. Limerick Unit 2 P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-4, 5-5, 5-6 AND 5-8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	32 EFPY BELTLINE CURVE B (°F)
540	68.0	100.0	100.0	72.1	132.8	142.5
550	68.0	100.0	100.0	73.9	133.6	144.0
560	68.0	100.0	100.0	75.7	134.4	145.4
570	68.0	100.0	100.0	77.4	135.1	146.8
580	68.0	100.0	100.0	79.0	135.9	148.2
590	68.0	100.0	100.0	80.6	136.6	149.5
600	68.0	100.0	100.0	82.2	137.1	150.8
610	68.0	100.0	101.6	83.7	137.6	152.1
620	68.0	100.0	104.1	85.1	138.0	153.3
630	68.0	100.0	106.5	86.5	138.4	154.5
640	68.0	100.0	108.8	87.9	138.8	155.7
650	68.0	100.0	110.9	89.2	139.2	156.8
660	68.0	100.0	113.0	90.5	139.7	157.9
670	68.0	100.0	115.0	91.8	140.1	159.0
680	68.0	100.0	116.9	93.1	140.5	160.1
690	68.0	100.0	118.8	94.3	140.9	161.1
700	69.2	100.0	120.6	95.4	141.3	162.2
710	70.7	100.0	122.3	96.6	141.7	163.2
720	72.1	100.4	123.9	97.7	142.1	164.1
730	73.5	101.3	125.6	98.8	142.5	165.1
740	74.8	102.1	127.1	99.9	142.9	166.1
750	76.1	103.0	128.6	101.0	143.2	167.0
760	77.4	103.8	130.1	102.0	143.6	167.9
770	78.6	104.6	131.5	103.0	144.0	168.8
780	79.8	105.3	132.9	104.0	144.4	169.7
790	81.0	106.1	134.3	105.0	144.8	170.6
800	82.2	106.9	135.6	105.9	145.1	171.4
810	83.3	107.6	136.9	106.9	145.5	172.2
820	84.4	108.4	138.1	107.8	145.9	173.1

TABLE B-1. Limerick Unit 2 P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-4, 5-5, 5-6 AND 5-8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	32 EFPY BELTLINE CURVE B (°F)
830	85.5	109.1	139.3	108.7	146.2	173.9
840	86.5	109.8	140.5	109.6	146.6	174.7
850	87.6	110.5	141.7	110.4	146.9	175.5
860	88.6	111.2	142.8	111.3	147.3	176.2
870	89.6	111.9	143.9	112.1	147.6	177.0
880	90.5	112.6	145.0	113.0	148.0	177.7
890	91.5	113.3	146.1	113.8	148.3	178.5
900	92.4	113.9	147.1	114.6	148.7	179.2
910	93.4	114.6	148.1	115.4	149.0	179.9
920	94.3	115.2	149.1	116.1	149.4	180.6
930	95.1	115.9	150.1	116.9	149.7	181.3
940	96.0	116.5	151.1	117.7	150.0	182.0
950	96.9	117.1	152.0	118.4	150.4	182.7
960	97.7	117.7	152.9	119.1	150.7	183.4
970	98.6	118.3	153.8	119.9	151.0	184.0
980	99.4	118.9	154.7	120.6	151.4	184.7
990	100.2	119.5	155.6	121.3	151.7	185.3
1000	101.0	120.1	156.5	122.0	152.0	186.0
1010	101.7	120.7	157.3	122.6	152.3	186.6
1020	102.5	121.2	158.2	123.3	152.6	187.2
1030	103.3	121.8	159.0	124.0	153.0	187.8
1040	104.0	122.4	159.8	124.6	153.3	188.4
1050	104.7	122.9	160.6	125.3	153.6	189.0
1060	105.4	123.5	161.4	125.9	153.9	189.6
1070	106.2	124.0	162.1	126.5	154.2	190.2
1080	106.9	124.5	162.9	127.2	154.5	190.8
1090	107.6	125.1	163.6	127.8	154.8	191.3
1100	108.2	125.6	164.4	128.4	155.1	191.9
1110	108.9	126.1	165.1	129.0	155.4	192.5

TABLE B-1. Limerick Unit 2 P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-4, 5-5, 5-6 AND 5-8

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	32 EFPY BELTLINE CURVE B (°F)
1120	109.6	126.6	165.8	129.6	155.7	193.0
1130	110.2	127.1	166.5	130.2	156.0	193.6
1140	110.9	127.6	167.2	130.7	156.3	194.1
1150	111.5	128.1	167.9	131.3	156.6	194.6
1160	112.1	128.6	168.6	131.9	156.9	195.2
1170	112.8	129.1	169.2	132.4	157.2	195.7
1180	113.4	129.6	169.9	133.0	157.5	196.2
1190	114.0	130.1	170.5	133.5	157.7	196.7
1200	114.6	130.5	171.2	134.1	158.0	197.2
1210	115.2	131.0	171.8	134.6	158.3	197.7
1220	115.8	131.5	172.4	135.2	158.6	198.2
1230	116.3	131.9	173.0	135.7	158.9	198.7
1240	116.9	132.4	173.7	136.2	159.2	199.2
1250	117.5	132.8	174.3	136.7	159.4	199.7
1260	118.0	133.3	174.8	137.2	159.7	200.2
1270	118.6	133.7	175.4	137.7	160.0	200.6
1280	119.1	134.2	176.0	138.2	160.2	201.1
1290	119.7	134.6	176.6	138.7	160.5	201.6
1300	120.2	135.0	177.2	139.2	160.8	202.0
1310	120.7	135.5	177.7	139.7	161.1	202.5
1320	121.3	135.9	178.3	140.2	161.3	202.9
1330	121.8	136.3	178.8	140.6	161.6	203.4
1340	122.3	136.7	179.4	141.1	161.8	203.8
1350	122.8	137.1	179.9	141.6	162.1	204.3
1360	123.3	137.6	180.4	142.0	162.4	204.7
1370	123.8	138.0	181.0	142.5	162.6	205.1
1380	124.3	138.4	181.5	142.9	162.9	205.6
1390	124.8	138.8	182.0	143.4	163.1	206.0
1400	125.3	139.2	182.5	143.8	163.4	206.4

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	70.1
60	68.0	70.0	68.0	70.0	79.0
70	68.0	70.0	68.0	70.0	86.2
80	68.0	70.0	68.0	70.0	92.2
90	68.0	70.0	68.0	70.0	97.3
100	68.0	70.0	68.0	70.0	101.8
110	68.0	70.0	68.0	70.0	105.9
120	68.0	70.0	68.0	70.0	109.7
130	68.0	70.0	68.0	73.2	113.2
140	68.0	70.0	68.0	76.4	116.4
150	68.0	70.0	68.0	79.2	119.2
160	68.0	70.0	68.0	81.9	121.9
170	68.0	70.0	68.0	84.5	124.5
180	68.0	70.0	68.0	86.9	126.9
190	68.0	70.0	68.0	89.2	129.2
200	68.0	70.0	68.0	91.3	131.3
210	68.0	70.0	68.0	93.3	133.3
220	68.0	70.0	68.0	95.3	135.3
230	68.0	70.0	68.0	97.1	137.1
240	68.0	70.0	68.0	98.9	138.9
250	68.0	70.0	68.0	100.6	140.6
260	68.0	70.0	68.0	102.2	142.2

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
270	68.0	70.0	68.0	103.8	143.8
280	68.0	70.0	68.0	105.3	145.3
290	68.0	70.0	68.0	106.8	146.8
300	68.0	70.0	68.0	108.2	148.2
310	68.0	70.0	68.0	109.5	149.5
312.5	68.0	70.0	68.0	109.9	149.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.0	170.0
420	68.0	100.0	68.0	130.0	170.0
430	68.0	100.0	68.0	130.0	170.0
440	68.0	100.0	68.0	130.0	170.0
450	68.0	100.0	68.0	130.0	170.0
460	68.0	100.0	68.0	130.0	170.0
470	68.0	100.0	68.0	130.6	170.6
480	68.0	100.0	68.0	132.5	172.5
490	68.0	100.0	68.0	134.3	174.3
500	68.0	100.0	68.0	136.1	176.1
510	68.0	100.0	68.0	137.8	177.8

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
520	68.0	100.0	68.2	139.4	179.4
530	68.0	100.0	70.2	141.0	181.0
540	68.0	100.0	72.1	142.5	182.5
550	68.0	100.0	73.9	144.0	184.0
560	68.0	100.0	75.7	145.4	185.4
570	68.0	100.0	77.4	146.8	186.8
580	68.0	100.0	79.0	148.2	188.2
590	68.0	100.0	80.6	149.5	189.5
600	68.0	100.0	82.2	150.8	190.8
610	68.0	101.6	83.7	152.1	192.1
620	68.0	104.1	85.1	153.3	193.3
630	68.0	106.5	86.5	154.5	194.5
640	68.0	108.8	87.9	155.7	195.7
650	68.0	110.9	89.2	156.8	196.8
660	68.0	113.0	90.5	157.9	197.9
670	68.0	115.0	91.8	159.0	199.0
680	68.0	116.9	93.1	160.1	200.1
690	68.0	118.8	94.3	161.1	201.1
700	69.2	120.6	95.4	162.2	202.2
710	70.7	122.3	96.6	163.2	203.2
720	72.1	123.9	97.7	164.1	204.1
730	73.5	125.6	98.8	165.1	205.1
740	74.8	127.1	99.9	166.1	206.1
750	76.1	128.6	101.0	167.0	207.0
760	77.4	130.1	102.0	167.9	207.9
770	78.6	131.5	103.0	168.8	208.8
780	79.8	132.9	104.0	169.7	209.7

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
790	81.0	134.3	105.0	170.6	210.6
800	82.2	135.6	105.9	171.4	211.4
810	83.3	136.9	106.9	172.2	212.2
820	84.4	138.1	107.8	173.1	213.1
830	85.5	139.3	108.7	173.9	213.9
840	86.5	140.5	109.6	174.7	214.7
850	87.6	141.7	110.4	175.5	215.5
860	88.6	142.8	111.3	176.2	216.2
870	89.6	143.9	112.1	177.0	217.0
880	90.5	145.0	113.0	177.7	217.7
890	91.5	146.1	113.8	178.5	218.5
900	92.4	147.1	114.6	179.2	219.2
910	93.4	148.1	115.4	179.9	219.9
920	94.3	149.1	116.1	180.6	220.6
930	95.1	150.1	116.9	181.3	221.3
940	96.0	151.1	117.7	182.0	222.0
950	96.9	152.0	118.4	182.7	222.7
960	97.7	152.9	119.1	183.4	223.4
970	98.6	153.8	119.9	184.0	224.0
980	99.4	154.7	120.6	184.7	224.7
990	100.2	155.6	121.3	185.3	225.3
1000	101.0	156.5	122.0	186.0	226.0
1010	101.7	157.3	122.6	186.6	226.6
1020	102.5	158.2	123.3	187.2	227.2
1030	103.3	159.0	124.0	187.8	227.8
1040	104.0	159.8	124.6	188.4	228.4
1050	104.7	160.6	125.3	189.0	229.0

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
1060	105.4	161.4	125.9	189.6	229.6
1070	106.2	162.1	126.5	190.2	230.2
1080	106.9	162.9	127.2	190.8	230.8
1090	107.6	163.6	127.8	191.3	231.3
1100	108.2	164.4	128.4	191.9	231.9
1110	108.9	165.1	129.0	192.5	232.5
1120	109.6	165.8	129.6	193.0	233.0
1130	110.2	166.5	130.2	193.6	233.6
1140	110.9	167.2	130.7	194.1	234.1
1150	111.5	167.9	131.3	194.6	234.6
1160	112.1	168.6	131.9	195.2	235.2
1170	112.8	169.2	132.4	195.7	235.7
1180	113.4	169.9	133.0	196.2	236.2
1190	114.0	170.5	133.5	196.7	236.7
1200	114.6	171.2	134.1	197.2	237.2
1210	115.2	171.8	134.6	197.7	237.7
1220	115.8	172.4	135.2	198.2	238.2
1230	116.3	173.0	135.7	198.7	238.7
1240	116.9	173.7	136.2	199.2	239.2
1250	117.5	174.3	136.7	199.7	239.7
1260	118.0	174.8	137.2	200.2	240.2
1270	118.6	175.4	137.7	200.6	240.6
1280	119.1	176.0	138.2	201.1	241.1
1290	119.7	176.6	138.7	201.6	241.6
1300	120.2	177.2	139.2	202.0	242.0
1310	120.7	177.7	139.7	202.5	242.5
1320	121.3	178.3	140.2	202.9	242.9

TABLE B-2. Limerick Unit 2 Composite P-T Curve Values for 32 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-10, 5-11, AND 5-12

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 32 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 32 EFPY CURVE C (°F)
1330	121.8	178.8	140.6	203.4	243.4
1340	122.3	179.4	141.1	203.8	243.8
1350	122.8	179.9	141.6	204.3	244.3
1360	123.3	180.4	142.0	204.7	244.7
1370	123.8	181.0	142.5	205.1	245.1
1380	124.3	181.5	142.9	205.6	245.6
1390	124.8	182.0	143.4	206.0	246.0
1400	125.3	182.5	143.8	206.4	246.4

TABLE B-3. Limerick Unit 2 P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-3, 5-5, 5-6, AND 5-7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	22 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	22 EFPY BELTLINE CURVE B (°F)
0	68.0	70.0	70.0	68.0	70.0	70.0
10	68.0	70.0	70.0	68.0	70.0	70.0
20	68.0	70.0	70.0	68.0	70.0	70.0
30	68.0	70.0	70.0	68.0	70.0	70.0
40	68.0	70.0	70.0	68.0	70.0	70.0
50	68.0	70.0	70.0	68.0	70.0	70.0
60	68.0	70.0	70.0	68.0	70.0	70.0
70	68.0	70.0	70.0	68.0	70.0	70.0
80	68.0	70.0	70.0	68.0	70.0	70.0
90	68.0	70.0	70.0	68.0	70.0	70.0
100	68.0	70.0	70.0	68.0	70.0	70.0
110	68.0	70.0	70.0	68.0	70.0	70.0
120	68.0	70.0	70.0	68.0	70.0	70.0
130	68.0	70.0	70.0	68.0	73.2	70.0
140	68.0	70.0	70.0	68.0	76.4	70.0
150	68.0	70.0	70.0	68.0	79.2	70.0
160	68.0	70.0	70.0	68.0	81.9	70.0
170	68.0	70.0	70.0	68.0	84.5	71.5
180	68.0	70.0	70.0	68.0	86.9	73.9
190	68.0	70.0	70.0	68.0	89.2	76.2
200	68.0	70.0	70.0	68.0	91.3	78.3
210	68.0	70.0	70.0	68.0	93.3	80.3
220	68.0	70.0	70.0	68.0	95.3	82.3
230	68.0	70.0	70.0	68.0	97.1	84.1
240	68.0	70.0	70.0	68.0	98.9	85.9
250	68.0	70.0	70.0	68.0	100.6	87.6
260	68.0	70.0	70.0	68.0	102.2	89.2
270	68.0	70.0	70.0	68.0	103.8	90.8
280	68.0	70.0	70.0	68.0	105.3	92.3

TABLE B-3. Limerick Unit 2 P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-3, 5-5, 5-6, AND 5-7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	22 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	22 EFPY BELTLINE CURVE B (°F)
290	68.0	70.0	70.0	68.0	106.8	93.8
300	68.0	70.0	70.0	68.0	108.2	95.2
310	68.0	70.0	70.0	68.0	109.5	96.5
312.5	68.0	70.0	70.0	68.0	109.9	96.9
312.5	68.0	100.0	100.0	68.0	130.0	130.0
320	68.0	100.0	100.0	68.0	130.0	130.0
330	68.0	100.0	100.0	68.0	130.0	130.0
340	68.0	100.0	100.0	68.0	130.0	130.0
350	68.0	100.0	100.0	68.0	130.0	130.0
360	68.0	100.0	100.0	68.0	130.0	130.0
370	68.0	100.0	100.0	68.0	130.0	130.0
380	68.0	100.0	100.0	68.0	130.0	130.0
390	68.0	100.0	100.0	68.0	130.0	130.0
400	68.0	100.0	100.0	68.0	130.0	130.0
410	68.0	100.0	100.0	68.0	130.0	130.0
420	68.0	100.0	100.0	68.0	130.0	130.0
430	68.0	100.0	100.0	68.0	130.0	130.0
440	68.0	100.0	100.0	68.0	130.0	130.0
450	68.0	100.0	100.0	68.0	130.0	130.0
460	68.0	100.0	100.0	68.0	130.0	130.0
470	68.0	100.0	100.0	68.0	130.0	130.0
480	68.0	100.0	100.0	68.0	130.0	130.0
490	68.0	100.0	100.0	68.0	130.0	130.0
500	68.0	100.0	100.0	68.0	130.0	130.0
510	68.0	100.0	100.0	68.0	130.4	130.0
520	68.0	100.0	100.0	68.2	131.2	131.4
530	68.0	100.0	100.0	70.2	132.0	133.0
540	68.0	100.0	100.0	72.1	132.8	134.5
550	68.0	100.0	100.0	73.9	133.6	136.0

TABLE B-3. Limerick Unit 2 P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-3, 5-5, 5-6, AND 5-7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	22 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	22 EFPY BELTLINE CURVE B (°F)
560	68.0	100.0	100.0	75.7	134.4	137.4
570	68.0	100.0	100.0	77.4	135.1	138.8
580	68.0	100.0	100.0	79.0	135.9	140.2
590	68.0	100.0	100.0	80.6	136.6	141.5
600	68.0	100.0	100.0	82.2	137.1	142.8
610	68.0	100.0	100.0	83.7	137.6	144.1
620	68.0	100.0	100.0	85.1	138.0	145.3
630	68.0	100.0	100.0	86.5	138.4	146.5
640	68.0	100.0	100.8	87.9	138.8	147.7
650	68.0	100.0	102.9	89.2	139.2	148.8
660	68.0	100.0	105.0	90.5	139.7	149.9
670	68.0	100.0	107.0	91.8	140.1	151.0
680	68.0	100.0	108.9	93.1	140.5	152.1
690	68.0	100.0	110.8	94.3	140.9	153.1
700	69.2	100.0	112.6	95.4	141.3	154.2
710	70.7	100.0	114.3	96.6	141.7	155.2
720	72.1	100.4	115.9	97.7	142.1	156.1
730	73.5	101.3	117.6	98.8	142.5	157.1
740	74.8	102.1	119.1	99.9	142.9	158.1
750	76.1	103.0	120.6	101.0	143.2	159.0
760	77.4	103.8	122.1	102.0	143.6	159.9
770	78.6	104.6	123.5	103.0	144.0	160.8
780	79.8	105.3	124.9	104.0	144.4	161.7
790	81.0	106.1	126.3	105.0	144.8	162.6
800	82.2	106.9	127.6	105.9	145.1	163.4
810	83.3	107.6	128.9	106.9	145.5	164.2
820	84.4	108.4	130.1	107.8	145.9	165.1
830	85.5	109.1	131.3	108.7	146.2	165.9
840	86.5	109.8	132.5	109.6	146.6	166.7

TABLE B-3. Limerick Unit 2 P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-3, 5-5, 5-6, AND 5-7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	22 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	22 EFPY BELTLINE CURVE B (°F)
850	87.6	110.5	133.7	110.4	146.9	167.5
860	88.6	111.2	134.8	111.3	147.3	168.2
870	89.6	111.9	135.9	112.1	147.6	169.0
880	90.5	112.6	137.0	113.0	148.0	169.7
890	91.5	113.3	138.1	113.8	148.3	170.5
900	92.4	113.9	139.1	114.6	148.7	171.2
910	93.4	114.6	140.1	115.4	149.0	171.9
920	94.3	115.2	141.1	116.1	149.4	172.6
930	95.1	115.9	142.1	116.9	149.7	173.3
940	96.0	116.5	143.1	117.7	150.0	174.0
950	96.9	117.1	144.0	118.4	150.4	174.7
960	97.7	117.7	144.9	119.1	150.7	175.4
970	98.6	118.3	145.8	119.9	151.0	176.0
980	99.4	118.9	146.7	120.6	151.4	176.7
990	100.2	119.5	147.6	121.3	151.7	177.3
1000	101.0	120.1	148.5	122.0	152.0	178.0
1010	101.7	120.7	149.3	122.6	152.3	178.6
1020	102.5	121.2	150.2	123.3	152.6	179.2
1030	103.3	121.8	151.0	124.0	153.0	179.8
1040	104.0	122.4	151.8	124.6	153.3	180.4
1050	104.7	122.9	152.6	125.3	153.6	181.0
1060	105.4	123.5	153.4	125.9	153.9	181.6
1070	106.2	124.0	154.1	126.5	154.2	182.2
1080	106.9	124.5	154.9	127.2	154.5	182.8
1090	107.6	125.1	155.6	127.8	154.8	183.3
1100	108.2	125.6	156.4	128.4	155.1	183.9
1110	108.9	126.1	157.1	129.0	155.4	184.5
1120	109.6	126.6	157.8	129.6	155.7	185.0
1130	110.2	127.1	158.5	130.2	156.0	185.6

TABLE B-3. Limerick Unit 2 P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURES 5-1, 5-2, 5-3, 5-5, 5-6, AND 5-7

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER VESSEL CURVE A (°F)	22 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER VESSEL CURVE B (°F)	22 EFPY BELTLINE CURVE B (°F)
1140	110.9	127.6	159.2	130.7	156.3	186.1
1150	111.5	128.1	159.9	131.3	156.6	186.6
1160	112.1	128.6	160.6	131.9	156.9	187.2
1170	112.8	129.1	161.2	132.4	157.2	187.7
1180	113.4	129.6	161.9	133.0	157.5	188.2
1190	114.0	130.1	162.5	133.5	157.7	188.7
1200	114.6	130.5	163.2	134.1	158.0	189.2
1210	115.2	131.0	163.8	134.6	158.3	189.7
1220	115.8	131.5	164.4	135.2	158.6	190.2
1230	116.3	131.9	165.0	135.7	158.9	190.7
1240	116.9	132.4	165.7	136.2	159.2	191.2
1250	117.5	132.8	166.3	136.7	159.4	191.7
1260	118.0	133.3	166.8	137.2	159.7	192.2
1270	118.6	133.7	167.4	137.7	160.0	192.6
1280	119.1	134.2	168.0	138.2	160.2	193.1
1290	119.7	134.6	168.6	138.7	160.5	193.6
1300	120.2	135.0	169.2	139.2	160.8	194.0
1310	120.7	135.5	169.7	139.7	161.1	194.5
1320	121.3	135.9	170.3	140.2	161.3	194.9
1330	121.8	136.3	170.8	140.6	161.6	195.4
1340	122.3	136.7	171.4	141.1	161.8	195.8
1350	122.8	137.1	171.9	141.6	162.1	196.3
1360	123.3	137.6	172.4	142.0	162.4	196.7
1370	123.8	138.0	173.0	142.5	162.6	197.1
1380	124.3	138.4	173.5	142.9	162.9	197.6
1390	124.8	138.8	174.0	143.4	163.1	198.0
1400	125.3	139.2	174.5	143.8	163.4	198.4

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
0	68.0	70.0	68.0	70.0	70.0
10	68.0	70.0	68.0	70.0	70.0
20	68.0	70.0	68.0	70.0	70.0
30	68.0	70.0	68.0	70.0	70.0
40	68.0	70.0	68.0	70.0	70.0
50	68.0	70.0	68.0	70.0	70.1
60	68.0	70.0	68.0	70.0	79.0
70	68.0	70.0	68.0	70.0	86.2
80	68.0	70.0	68.0	70.0	92.2
90	68.0	70.0	68.0	70.0	97.3
100	68.0	70.0	68.0	70.0	101.8
110	68.0	70.0	68.0	70.0	105.9
120	68.0	70.0	68.0	70.0	109.7
130	68.0	70.0	68.0	73.2	113.2
140	68.0	70.0	68.0	76.4	116.4
150	68.0	70.0	68.0	79.2	119.2
160	68.0	70.0	68.0	81.9	121.9
170	68.0	70.0	68.0	84.5	124.5
180	68.0	70.0	68.0	86.9	126.9
190	68.0	70.0	68.0	89.2	129.2
200	68.0	70.0	68.0	91.3	131.3
210	68.0	70.0	68.0	93.3	133.3
220	68.0	70.0	68.0	95.3	135.3
230	68.0	70.0	68.0	97.1	137.1

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
240	68.0	70.0	68.0	98.9	138.9
250	68.0	70.0	68.0	100.6	140.6
260	68.0	70.0	68.0	102.2	142.2
270	68.0	70.0	68.0	103.8	143.8
280	68.0	70.0	68.0	105.3	145.3
290	68.0	70.0	68.0	106.8	146.8
300	68.0	70.0	68.0	108.2	148.2
310	68.0	70.0	68.0	109.5	149.5
312.5	68.0	70.0	68.0	109.9	149.9
312.5	68.0	100.0	68.0	130.0	170.0
320	68.0	100.0	68.0	130.0	170.0
330	68.0	100.0	68.0	130.0	170.0
340	68.0	100.0	68.0	130.0	170.0
350	68.0	100.0	68.0	130.0	170.0
360	68.0	100.0	68.0	130.0	170.0
370	68.0	100.0	68.0	130.0	170.0
380	68.0	100.0	68.0	130.0	170.0
390	68.0	100.0	68.0	130.0	170.0
400	68.0	100.0	68.0	130.0	170.0
410	68.0	100.0	68.0	130.0	170.0
420	68.0	100.0	68.0	130.0	170.0
430	68.0	100.0	68.0	130.0	170.0
440	68.0	100.0	68.0	130.0	170.0
450	68.0	100.0	68.0	130.0	170.0
460	68.0	100.0	68.0	130.0	170.0
470	68.0	100.0	68.0	130.0	170.0

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
480	68.0	100.0	68.0	130.0	170.0
490	68.0	100.0	68.0	130.0	170.0
500	68.0	100.0	68.0	130.0	170.0
510	68.0	100.0	68.0	130.4	170.4
520	68.0	100.0	68.2	131.4	171.4
530	68.0	100.0	70.2	133.0	173.0
540	68.0	100.0	72.1	134.5	174.5
550	68.0	100.0	73.9	136.0	176.0
560	68.0	100.0	75.7	137.4	177.4
570	68.0	100.0	77.4	138.8	178.8
580	68.0	100.0	79.0	140.2	180.2
590	68.0	100.0	80.6	141.5	181.5
600	68.0	100.0	82.2	142.8	182.8
610	68.0	100.0	83.7	144.1	184.1
620	68.0	100.0	85.1	145.3	185.3
630	68.0	100.0	86.5	146.5	186.5
640	68.0	100.8	87.9	147.7	187.7
650	68.0	102.9	89.2	148.8	188.8
660	68.0	105.0	90.5	149.9	189.9
670	68.0	107.0	91.8	151.0	191.0
680	68.0	108.9	93.1	152.1	192.1
690	68.0	110.8	94.3	153.1	193.1
700	69.2	112.6	95.4	154.2	194.2
710	70.7	114.3	96.6	155.2	195.2
720	72.1	115.9	97.7	156.1	196.1
730	73.5	117.6	98.8	157.1	197.1

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
740	74.8	119.1	99.9	158.1	198.1
750	76.1	120.6	101.0	159.0	199.0
760	77.4	122.1	102.0	159.9	199.9
770	78.6	123.5	103.0	160.8	200.8
780	79.8	124.9	104.0	161.7	201.7
790	81.0	126.3	105.0	162.6	202.6
800	82.2	127.6	105.9	163.4	203.4
810	83.3	128.9	106.9	164.2	204.2
820	84.4	130.1	107.8	165.1	205.1
830	85.5	131.3	108.7	165.9	205.9
840	86.5	132.5	109.6	166.7	206.7
850	87.6	133.7	110.4	167.5	207.5
860	88.6	134.8	111.3	168.2	208.2
870	89.6	135.9	112.1	169.0	209.0
880	90.5	137.0	113.0	169.7	209.7
890	91.5	138.1	113.8	170.5	210.5
900	92.4	139.1	114.6	171.2	211.2
910	93.4	140.1	115.4	171.9	211.9
920	94.3	141.1	116.1	172.6	212.6
930	95.1	142.1	116.9	173.3	213.3
940	96.0	143.1	117.7	174.0	214.0
950	96.9	144.0	118.4	174.7	214.7
960	97.7	144.9	119.1	175.4	215.4
970	98.6	145.8	119.9	176.0	216.0
980	99.4	146.7	120.6	176.7	216.7
990	100.2	147.6	121.3	177.3	217.3

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
1000	101.0	148.5	122.0	178.0	218.0
1010	101.7	149.3	122.6	178.6	218.6
1020	102.5	150.2	123.3	179.2	219.2
1030	103.3	151.0	124.0	179.8	219.8
1040	104.0	151.8	124.6	180.4	220.4
1050	104.7	152.6	125.3	181.0	221.0
1060	105.4	153.4	125.9	181.6	221.6
1070	106.2	154.1	126.5	182.2	222.2
1080	106.9	154.9	127.2	182.8	222.8
1090	107.6	155.6	127.8	183.3	223.3
1100	108.2	156.4	128.4	183.9	223.9
1110	108.9	157.1	129.0	184.5	224.5
1120	109.6	157.8	129.6	185.0	225.0
1130	110.2	158.5	130.2	185.6	225.6
1140	110.9	159.2	130.7	186.1	226.1
1150	111.5	159.9	131.3	186.6	226.6
1160	112.1	160.6	131.9	187.2	227.2
1170	112.8	161.2	132.4	187.7	227.7
1180	113.4	161.9	133.0	188.2	228.2
1190	114.0	162.5	133.5	188.7	228.7
1200	114.6	163.2	134.1	189.2	229.2
1210	115.2	163.8	134.6	189.7	229.7
1220	115.8	164.4	135.2	190.2	230.2
1230	116.3	165.0	135.7	190.7	230.7
1240	116.9	165.7	136.2	191.2	231.2
1250	117.5	166.3	136.7	191.7	231.7

TABLE B-4. Limerick Unit 2 Composite P-T Curve Values for 22 EFPY

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-9

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	UPPER RPV & BELTLINE AT 22 EFPY CURVE B (°F)	NON-BELTLINE AND BELTLINE AT 22 EFPY CURVE C (°F)
1260	118.0	166.8	137.2	192.2	232.2
1270	118.6	167.4	137.7	192.6	232.6
1280	119.1	168.0	138.2	193.1	233.1
1290	119.7	168.6	138.7	193.6	233.6
1300	120.2	169.2	139.2	194.0	234.0
1310	120.7	169.7	139.7	194.5	234.5
1320	121.3	170.3	140.2	194.9	234.9
1330	121.8	170.8	140.6	195.4	235.4
1340	122.3	171.4	141.1	195.8	235.8
1350	122.8	171.9	141.6	196.3	236.3
1360	123.3	172.4	142.0	196.7	236.7
1370	123.8	173.0	142.5	197.1	237.1
1380	124.3	173.5	142.9	197.6	237.6
1390	124.8	174.0	143.4	198.0	238.0
1400	125.3	174.5	143.8	198.4	238.4

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
0	70.0	70.0	70.0	70.0
10	70.0	70.0	70.0	70.0
20	70.0	70.0	70.0	70.0
30	70.0	70.0	70.0	70.0
40	70.0	70.0	70.0	70.0
50	70.0	70.0	70.0	70.1
60	70.0	70.0	70.0	79.0
70	70.0	70.0	70.0	86.2
80	70.0	70.0	70.0	92.2
90	70.0	70.0	70.0	97.3
100	70.0	70.0	70.0	101.8
110	70.0	70.0	70.0	105.9
120	70.0	70.0	70.0	109.7
130	70.0	70.0	73.2	113.2
140	70.0	70.0	76.4	116.4
150	70.0	70.0	79.2	119.2
160	70.0	70.0	81.9	121.9
170	70.0	70.0	84.5	124.5
180	70.0	70.0	86.9	126.9
190	70.0	70.0	89.2	129.2
200	70.0	70.0	91.3	131.3
210	70.0	70.0	93.3	133.3
220	70.0	70.0	95.3	135.3
230	70.0	70.0	97.1	137.1
240	70.0	70.0	98.9	138.9
250	70.0	70.0	100.6	140.6
260	70.0	70.0	102.2	142.2

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
270	70.0	70.0	103.8	143.8
280	70.0	70.0	105.3	145.3
290	70.0	70.0	106.8	146.8
300	70.0	70.0	108.2	148.2
310	70.0	70.0	109.5	149.5
312.5	70.0	70.0	109.9	149.9
312.5	100.0	100.0	130.0	170.0
320	100.0	100.0	130.0	170.0
330	100.0	100.0	130.0	170.0
340	100.0	100.0	130.0	170.0
350	100.0	100.0	130.0	170.0
360	100.0	100.0	130.0	170.0
370	100.0	100.0	130.0	170.0
380	100.0	100.0	130.0	170.0
390	100.0	100.0	130.0	170.0
400	100.0	100.0	130.0	170.0
410	100.0	100.0	130.0	170.0
420	100.0	100.0	130.0	170.0
430	100.0	100.0	130.0	170.0
440	100.0	100.0	130.0	170.0
450	100.0	100.0	130.0	170.0
460	100.0	100.0	130.0	170.0
470	100.0	100.0	130.6	170.6
480	100.0	100.0	132.5	172.5
490	100.0	100.0	134.3	174.3
500	100.0	100.0	136.1	176.1
510	100.0	100.0	137.8	177.8

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
520	100.0	100.0	139.4	179.4
530	100.0	100.0	141.0	181.0
540	100.0	100.0	142.5	182.5
550	100.0	100.0	144.0	184.0
560	100.0	100.0	145.4	185.4
570	100.0	100.0	146.8	186.8
580	100.0	100.0	148.2	188.2
590	100.0	100.0	149.5	189.5
600	100.0	100.0	150.8	190.8
610	100.0	101.6	152.1	192.1
620	100.0	104.1	153.3	193.3
630	100.0	106.5	154.5	194.5
640	100.8	108.8	155.7	195.7
650	102.9	110.9	156.8	196.8
660	105.0	113.0	157.9	197.9
670	107.0	115.0	159.0	199.0
680	108.9	116.9	160.1	200.1
690	110.8	118.8	161.1	201.1
700	112.6	120.6	162.2	202.2
710	114.3	122.3	163.2	203.2
720	115.9	123.9	164.1	204.1
730	117.6	125.6	165.1	205.1
740	119.1	127.1	166.1	206.1
750	120.6	128.6	167.0	207.0
760	122.1	130.1	167.9	207.9
770	123.5	131.5	168.8	208.8
780	124.9	132.9	169.7	209.7

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
790	126.3	134.3	170.6	210.6
800	127.6	135.6	171.4	211.4
810	128.9	136.9	172.2	212.2
820	130.1	138.1	173.1	213.1
830	131.3	139.3	173.9	213.9
840	132.5	140.5	174.7	214.7
850	133.7	141.7	175.5	215.5
860	134.8	142.8	176.2	216.2
870	135.9	143.9	177.0	217.0
880	137.0	145.0	177.7	217.7
890	138.1	146.1	178.5	218.5
900	139.1	147.1	179.2	219.2
910	140.1	148.1	179.9	219.9
920	141.1	149.1	180.6	220.6
930	142.1	150.1	181.3	221.3
940	143.1	151.1	182.0	222.0
950	144.0	152.0	182.7	222.7
960	144.9	152.9	183.4	223.4
970	145.8	153.8	184.0	224.0
980	146.7	154.7	184.7	224.7
990	147.6	155.6	185.3	225.3
1000	148.5	156.5	186.0	226.0
1010	149.3	157.3	186.6	226.6
1020	150.2	158.2	187.2	227.2
1030	151.0	159.0	187.8	227.8
1040	151.8	159.8	188.4	228.4
1050	152.6	160.6	189.0	229.0

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
1060	153.4	161.4	189.6	229.6
1070	154.1	162.1	190.2	230.2
1080	154.9	162.9	190.8	230.8
1090	155.6	163.6	191.3	231.3
1100	156.4	164.4	191.9	231.9
1110	157.1	165.1	192.5	232.5
1120	157.8	165.8	193.0	233.0
1130	158.5	166.5	193.6	233.6
1140	159.2	167.2	194.1	234.1
1150	159.9	167.9	194.6	234.6
1160	160.6	168.6	195.2	235.2
1170	161.2	169.2	195.7	235.7
1180	161.9	169.9	196.2	236.2
1190	162.5	170.5	196.7	236.7
1200	163.2	171.2	197.2	237.2
1210	163.8	171.8	197.7	237.7
1220	164.4	172.4	198.2	238.2
1230	165.0	173.0	198.7	238.7
1240	165.7	173.7	199.2	239.2
1250	166.3	174.3	199.7	239.7
1260	166.8	174.8	200.2	240.2
1270	167.4	175.4	200.6	240.6
1280	168.0	176.0	201.1	241.1
1290	168.6	176.6	201.6	241.6
1300	169.2	177.2	202.0	242.0
1310	169.7	177.7	202.5	242.5
1320	170.3	178.3	202.9	242.9

TABLE B-5. Limerick Unit 2 P-T Curve Values for Limiting 32 EFPY (A,B,C) and Limiting 22 EFPY (A Curve)

Required Coolant Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 5-13

PRESSURE (PSIG)	22 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE A (°F)	32 EFPY LIMITING CURVE B (°F)	32 EFPY LIMITING CURVE C (°F)
1330	170.8	178.8	203.4	243.4
1340	171.4	179.4	203.8	243.8
1350	171.9	179.9	204.3	244.3
1360	172.4	180.4	204.7	244.7
1370	173.0	181.0	205.1	245.1
1380	173.5	181.5	205.6	245.6
1390	174.0	182.0	206.0	246.0
1400	174.5	182.5	206.4	246.4

APPENDIX C

OPERATING AND TEMPERATURE MONITORING REQUIREMENTS

C.1 NON-BELTLINE MONITORING DURING PRESSURE TESTS

It is likely that, during leak and hydrostatic pressure testing, the bottom head temperature may be significantly cooler than the beltline. This condition can occur in the bottom head when the recirculation pumps are operating at low speed, or are off, and injection through the control rod drives is used to pressurize the vessel. By using a bottom head curve, the required test temperature at the bottom head could be lower than the required test temperature at the beltline, avoiding the necessity of heating the bottom head to the same requirements of the vessel beltline.

One condition on monitoring the bottom head separately is that it must be demonstrated that the vessel beltline temperature can be accurately monitored during pressure testing. An experiment has been conducted at a BWR-4 that showed that thermocouples on the vessel near the feedwater nozzles, or temperature measurements of water in the recirculation loops provide good estimates of the beltline temperature during pressure testing. Thermocouples on the RPV flange to shell junction outside surface should be used to monitor compliance with upper vessel curve. Thermocouples on the bottom head outside surface should be used to monitor compliance with bottom head curves. A description of these measurements is given in GE SIL 430, attached in Appendix D. First, however, it should be determined whether there are significant temperature differences between the beltline region and the bottom head region.

C.2 DETERMINING WHICH CURVE TO FOLLOW

The following subsections outline the criteria needed for determining which curve is governing during different situations. The application of the P-T curves and some of the assumptions inherent in the curves to plant operation is dependent on the proper monitoring of vessel temperatures.

C.2.1 Curve A: Pressure Test

Curve A should be used during pressure tests at times when the coolant temperature is changing by ≤20°F per hour. If the coolant is experiencing a higher heating or cooling rate in preparation for or following a pressure test, Curve B applies.

C.2.2 Curve B: Non-Nuclear Heatup/Cooldown

Curve B should be used whenever Curve A or Curve C do not apply. In other words, the operator must follow this curve during times when the coolant is heating or cooling faster than 20°F per hour during a hydrotest and when the core is not critical.

C.2.3 Curve C: Core Critical Operation

The operator must comply with this curve whenever the core is critical. An exception to this principle is for low-level physics tests; Curve B must be followed during these situations.

C.3 REACTOR OPERATION VERSUS OPERATING LIMITS

For most reactor operating conditions, coolant pressure and temperature are at saturation conditions, which are well into the acceptable operating area (to the right of the P-T curves). The operations where P-T curve compliance is typically monitored closely are planned events, such as vessel boltup, leakage testing and startup/shutdown operations, where operator actions can directly influence vessel pressures and temperatures.

The most severe unplanned transients relative to the P-T curves are those which result from SCRAMs, which sometimes include recirculation pump trips. Depending on operator responses following pump trip, there can be cases where stratification of colder water in the bottom head occurs while the vessel pressure is still relatively high. Experience with such events has shown that operator action is necessary to avoid P-T curve exceedance, but there is adequate time for operators to respond.

In summary, there are several operating conditions where careful monitoring of P-T conditions against the curves is needed:

- Head flange boltup
- Leakage test (Curve A compliance)
- Startup (coolant temperature change of less than or equal to 100°F in one hour period heatup)
- Shutdown (coolant temperature change of less than or equal to 100°F in one hour period cooldown)
- Recirculation pump trip, bottom head stratification (Curve B compliance)

APPENDIX D

GE SIL 430

September 27, 1985 SIL No. 430

REACTOR PRESSURE VESSEL TEMPERATURE MONITORING

Recently, several BWR owners with plants in initial startup have had questions concerning primary and alternate reactor pressure vessel (RPV) temperature monitoring measurements for complying with RPV brittle fracture and thermal stress requirements. As such, the purpose of this Service Information Letter is to provide a summary of RPV temperature monitoring measurements, their primary and alternate uses and their limitations (See the attached table). Of basic concern is temperature monitoring to comply with brittle fracture temperature limits and for vessel thermal stresses during RPV heatup and cooldown. General Electric recommends that BWR owners/operators review this table against their current practices and evaluate any inconsistencies.

TABLE OF RPV TEMPERATURE MONITORING MEASUREMENTS (Typical)

Measurement	Use	Limitations
Steam dome saturation temperature as determined from main steam instrument line pressure	Primary measurement above 212°F for Tech Spec 100°F/hr heatup and cooldown rate.	Must convert saturated steam pressure to temperature.
Recirc suction fine coolant temperature.	Primary measurement below 212°F for Tech Spec 100°F/hr heatup and cooldown rate.	Must have recirc flow. Must comply with SIL 251 to avoid vessel stratification.
	Alternate measurement above 212°F.	When above 212°F need to allow for temperature variations (up to 10-15°F lower than steam dome saturation temperature) caused primarily by FW flow variations.

TABLE OF RPV TEMPERATURE MONITORING MEASUREMENTS (CONTINUED)

(Typical)

Measurement	Use	Limitations
	Alternate measurement for RPV drain line temperature (can use to comply with delta T limit between steam dome saturation temperature and bottom head drain line temperature).	
RHR heat exchanger inlet coolant temperature	Alternate measurement for Tech Spec 100°F/hr cooldown rate when in shutdown cooling mode.	Must have previously correlated RHR inlet coolant temperature versus RPV coolant temperature.
RPV drain line coolant temperature	Primary measurement to comply with Tech Spec delta T limit between steam dome saturated temp and drain line coolant temperature.	Must have drain line flow. Otherwise, lower than actual temperature and higher delta T's will be indicated Delta T limit is 100°F for BWR/6s and 145°F for earlier BWRs.
	Primary measurement to comply with Tech Spec brittle fracture limits during cooldown.	Must have drain line flow. Use to verify compliance with Tech Spec minimum metal temperature/reactor pressure curves (using drain line temperature to represent bottom head metal temperature).
	Alternate information only measurement for bottom head inside/ outside metal surface	Must compensate for outside metal temperature lag during heatup/cooldown. Should have drain line flow.

temperatures.

TABLE OF RPV TEMPERATURE MONITORING MEASUREMENTS (CONTINUED)

(Typical)

Measurement	Use	Limitations
Closure head flanges outside surface T/Cs	Primary measurement for BWR/6s to comply with Tech Spec brittle fracture metal temperature limit for head boltup.	Use for metal (not coolant) temperature. Install temporary T/Cs for alternate measurement, if required.
	One of two primary measurements for BWR/6s for hydro test.	
RPV flange-to-shell junction outside surface T/Cs	Primary measurement for BWRs earlier than 6s to comply with Tech Spec brittle fracture metal temperature limit for head boltup.	Use for metal (not coolant) temperature. Response faster than closure head flange T/Cs.
	One of two primary measurements for BWRs earlier than 6s for hydro test. Preferred in lieu of closure head flange T/Cs if available.	Use RPV closure head flange outside surface as alternate measurement.
RPV shell outside surface T/Cs	Information only.	Slow to respond to RPV coolant changes. Not available on BWR/6s.
Top head outside surface T/Cs	Information only.	Very slow to respond to RPV coolant changes. Not available on BWR/6s.

TABLE OF RPV TEMPERATURE MONITORING MEASUREMENTS (CONTINUED)

(Typical)

Measurement	Use	Limitations
Bottom head outside surface T/Cs	1 of 2 primary measurements to comply with Tech Spec brittle fracture metal temperature limit for hydro test.	Should verify that vessel stratification is not present for vessel hydro. (see SIL No. 251).
	Primary measurement to comply with Tech Spec brittle fracture metal temperature limits during heatup.	Use during heatup to verify compliance with Tech Spec metal temperature/reactor pressure curves.

Note: RPV vendor specified metal T limits for vessel heatup and cooldown should be checked during initial plant startup tests when initial RPV vessel heatup and cooldown tests are run.

Product Reference: B21 Nuclear Boiler

Prepared By: A.C. Tsang

Approved for Issue:
B.H. Eldridge, Mgr.
Service Information
and Analysis

Issued By:
D.L. Allred, Manager
Customer Service Information

Notice:

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ATTACHMENT 8

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"
Supplemental Information Supporting Changes

GE Nuclear Energy Letter B13-00836-00-LT08
dated October 31, 2000

GE Nuclear Energy

Engineering and Licensing Consulting Services
175 Curtner Avenue M/C 747
San Jose, CA 95125
(408) 925-5945

B13-00836-00-LT08
October 31, 2000

TO: **David Sun**

FROM: **Lori J. Tilly** Lori Tilly

SUBJECT: **Limerick Generating Station Unit 2**
Response to Request for Additional Information

REFERENCE:
1) **Letter from D. W. Diefenderfer to G. M. Zaimis, " Limerick Generating Station Unit 1 Belt Line Dosimeter," GE-NEBO, San Jose, CA, July 15, 1987.**
2) **L.J. Tilly, GE-NE-B11-00836-00-02, Rev. 0, "Pressure-Temperature Curves for PECO Energy Company Limerick Unit 2", GENE (San Jose), July 2000**
3) **Lurie, D. and Moore, R.H., "Applying Statistics", U.S. Nuclear Regulatory Commission, NUREG-1475, February 1994**
4) **Letter #BJB-9710R1 from Betty J. Branlund to Bob McCall (PECO Energy), "Limerick Generating Station Unit 1 and 2 First Cycle Beltline Dosimeter Missing Data", GENE (San Jose), November 20, 1997 (retransmitted as Letter #BJB-0010 on July 12, 2000)**

Executive Summary

The Reference 2 report provides pressure-temperature (P-T) curves using updated methodology which are based upon the same fluence as the currently licensed P-T curves. PECO Energy has requested information in order to better understand the determination of fluence since Limerick Unit 2 does not have dosimetry data or a fluence calculation using current methodology. The report [2] states that the updated methodology contained in Code Cases N-588 and N-640 was used. It should be noted that Code Case N-588 as described in [2] has no impact because Limerick Unit 2 is plate limited with respect to the Adjusted Reference Temperature. The calculations for M_m are contained in the ASME Code endorsed by the NRC (1995 Edition with 1996 Addenda).

The revised Table 1 demonstrates that the flux used to determine the predicted capsule fluence for Limerick Units 1 and 2 (1.6×10^9 n/cm^2-s) contains conservatism of 18% above the measured average of 19 data points with a 95/95 confidence factor of 2.8σ (1.36×10^9 n/cm^2-s).

Table 2 provides the surveillance capsule and first cycle dosimeter azimuthal locations for the plants referenced in Table 1.

Table 3 and Figure 1 provide the corresponding obstructions between the capsules and the reactor core. In the case of all of the plants referenced in Table 1, the only obstruction is the 10 inch Schedule 30 Jet Pump Riser Pipe.

Table 4 provides the location of the welds in the Limerick Unit 2 reactor pressure vessel as clarification of the information contained in Table 4-3 of [2].

Introduction and Background

A letter was provided to PECO Energy in 1997 [4] discussing the baseline fluence evaluation for both Limerick Units. This information was discussed in a telecon which included the NRC, PECO Energy and GENE on 7/12/00. For convenience, the original text of the 1997 letter is included within this letter (denoted by *Beginning of Original Text* and *End of Original Text*) in addition to information for Limerick Unit 2 that is similar to the responses to the NRC requested information for Limerick Unit 1 and other items requested by PECO Energy. The specific information requested follows:

(a) azimuthal location of the surveillance capsules and first cycle dosimeters contained in Table 1 of this letter (which is also part of the original letter),
(b) definition of any obstructions between the reactor core and the surveillance capsules or dosimeters, and
(c) definition of the location of the welds listed in Table 4-3 of [2].

PECO Energy has also requested a normality test of the data presented in Table 1 and the application of a correction factor due to the small sample size. Therefore, Table 1 in this letter differs from the 1997 letter with respect to the Measured Data Average Plus (factor) Sigma as discussed in the text of this letter. Additionally, PECO Energy requested clarification as to why initial RT_{NDT} values changed during the P-T curve evaluation.

The information contained in the original Table 1 has been refined to include 95/95 confidence as defined by NUREG-1475 [3], and as a result of methods used to determine the normality of the data set as discussed below. This caused the flux value of 1.1 x 10^9 n/cm^2-s for "Measured Data Average Plus 2σ" to be revised to 1.36 x 10^9 n/cm^2-s, which represents "Measured Data Average Plus **2.8**σ". Specific calculated values are defined below using natural log, with the final flux value transformed to n/cm^2-s. Figure 2 provides descriptive statistics of the data contained in Table 1 and demonstrates the normality of the data set.

Natural Log Mean of Table 1 Data	20.2896
Natural Log Standard Deviation of Table 1 Data	0.2642
Natural Log Mean + 2.8σ where 2.8 derived from [3]	21.0294
Mean + 2.8σ (transformed to flux)	1.36 x 10^9 n/cm^2-s

In performing the task to create the new P-T curves, initial RT_{NDT} information changed from that previously reported in the FSAR due to a detailed review of the Certified Material Test Report (CMTR) data.

(Beginning of Original Text)
In 1987 the Reference 1 report addressed the concern of the missing first cycle flux wire dosimeters for Limerick Unit 1. The purpose of this letter is to update the 1987 letter, include the concern that first cycle flux wires are unavailable for Limerick Units 1 and 2, and provide further information regarding the baseline fluence evaluation. This letter addresses the influence of the unavailable flux wire data on the existing P -T curves particularly in light of the power rerate condition. Finally, compliance with 10CFR50 Appendix H and ASTM E185 including the concern of saturation of the flux wires as discussed in ASTM E185-82 is considered.

Summary

The wire test results would be used to confirm that the predicted peak ID surface fluence (1.7 x 10^{18} n/cm^2) used to develop the pre-rerate P-T curves was conservative (note that a fluence of 1.7 x 10^{18} n/cm^2 corresponds to a flux of 1.7 x 10^9 n/cm^2-sec). GE has reviewed the flux wire data collected to-date for numerous BWRs with a 251" vessel ID and 764 fuel bundles and determined that a flux of 1.7 x 10^9 n/cm^2-sec is conservative. In addition, sufficient data from similar plants exists to address the concern of flux wire saturation. Therefore, the station can be confident that the P-T curves developed using the predicted flux of 1.7 x 10^9 n/cm^2-sec for rated power and 1.9 x 10^9 n/cm^2-sec for rerated power are conservative.

Baseline Predicted Flux and Fluence

The baseline flux was determined using a generic one dimensional Sn calculation in 1983 for BWR/4&5s with a 251" vessel ID and 764 fuel bundles. The calculated peak vessel ID surface flux at energies greater than 1MeV is 8.5 x 10^8n/cm^2-sec. A safety factor of 2.0 was conservatively applied to establish a baseline predicted flux and end-of-life (EOL) fluence (flux = 1.7 x 10^9n/cm^2- sec and EOL fluence = 1.7 x 10^{18}n/cm^2). Since the lead factors (ratio of capsule flux to vessel ID surface flux) for plants of this size and fuel configuration range between 0.94 and 1.01, then this flux corresponds to a capsule flux of ~1.6 x 10^9n/cm^2- sec (i.e., 1.7 x 10^9n/cm^2- sec*0.94).

Note that in the Reference 1 letter, the predicted flux for some plants was reported to be 1.3 x 10^9n/cm^2-sec. This flux was determined from the same one dimensional Sn calculation, but using a safety factor of 1.5 instead of 2.0. The flux of 1.1 x 10^9n/cm^2-sec corresponds to the 1/4T flux.

Comparison of Predicted to Measured Data

Table 1 shows nineteen flux measurements taken between 1978 and 1996; ten measurements were from first cycle dosimeters, while the remaining nine were from capsules withdrawn between 6 and 9 EFPY. The population represents BWR 4 and 5 plants with 764 bundles and a thermal power rating of 3293 and 3323 MWth for the BWR/4 and 5, respectively. The average value of that data is 6.7 x 10^8n/cm^2- sec and the standard deviation is 1.9 x 10^8n/cm^2- sec. Therefore, the predicted flux used to develop the Limerick 1 and 2 P-T curves is well above two standard deviations of the measured data (i.e., 6.7 x 10^8n/cm^2- sec + 2(1.9 x 10^8n/cm^2- sec) = 1.1 x 10^9n/cm^2- sec, measured compared to 1.6 x 10^9n/cm^2- sec, predicted capsule flux). There is a factor of approximately 1.5 between the measured data with two standard deviations and the value used to develop the P-T curves. Also, the predicted capsule flux is 34% greater than the largest measured value (i.e., 1.2 x 10^9n/cm^2- sec * 1.34 = 1.6 x 10^9n/cm^2- sec). Therefore, the P-T curves are conservative.

Consideration of Power Rerate

For power rerate the original fluence (1.7 x 10^{18}n/cm^2) was assumed to increase proportional to the increase in power. To be conservative, it was assumed that the power increased from the first day of commercial operation rather than when the power rerate was implemented. Therefore, for the proposed 10% power rerate the fluence was increased by 10% to 1.9 x 10^{18}n/cm^2. Since Limerick Units 1 and 2 are licensed for 5% power rerate (from 3293 MWth to 3458 MWth), the fluence used to develop the P-T curves for power rerate are additionally conservative.

10CFR50 Appendix H and ASTM E185

The material surveillance program for both Limerick Units 1 and 2 are designed to meet the intent of ASTM E185-73. There are no requirements with ASTM E185-73 to include first cycle dosimetry. 10CFR50 Appendix H permits the use of revisions of ASTM E185 up to and including the 1982 version. In ASTM E185-82, although there are no specific requirements for first cycle dosimetry, article 7.3.3 states that separate dosimeter capsules should also be used to monitor radiation conditions independent of the specimen capsules if it is expected that the withdrawal schedule will otherwise result in saturation of the dosimeter activities.

The surveillance capsules for Limerick Units 1 and 2 contain iron and copper wires. The iron wire isotope, Mn-54, has a half-life of 312.5 days. The copper isotope is Co-60, has a half-life of 5.27 years. GE experience with these types of wires, pulled after 10 years of operation has shown that, while the iron wires reached a saturated condition after 4 or 5 years, consistent results with the "non-saturated" copper wires are achieved as long as an accurate daily power history is known. Extending the copper wire dosimetry to a saturated (or nearly saturated) condition is covered in ASTM E261-90, paragraph 9.5, which is referenced from the copper wire dosimetry method, ASTM E523-87. Paragraph 9.5 authorizes the use of the power history in the saturation condition, which will allow measurement of any saturated wire beyond the time of saturation (i.e., ~ 3 years for the iron wire and ~15 years for the copper wire). Based on these methods, saturation is not an issue for Unit 1 and 2 dosimetry. Therefore, there is no requirement nor recommendation for first cycle flux data based on 10CFR50 Appendix H, ASTM E185-73, and ASTM E185-82.

Conclusion

The data from similar plants demonstrate there is sufficient conservatism in the predicted fluence for both rated and rerated power. Conservatism in the fluence assures that the P-T curves are not impacted by the fact that the first cycle flux wire data is unavailable for both Limerick Units 1 and 2.
(End of Original Text)

Additional Information (Similar to July 2000 NRC Request for Additional Information for Limerick Unit 1)

The information provided below for Limerick Unit 2 is similar to the NRC requested additional information for Limerick Unit 1. As stated above, the additional items include: (a) the azimuthal location of the surveillance capsules and first cycle dosimeters for the data reported for the plants contained in Table 1, (b) any obstructions between the reactor core and the capsules and dosimeters for the plants contained in Table 1, and (c) the location of the welds listed in Table 4-3 of [2]. Table 2 contains the information requested in item (a), Table 3 contains the information requested in item (b), and Table 4 contains the information requested in item (c). Figure 1 provides a typical arrangement of the reactor core, surveillance capsule, and obstructions.

PECO Energy also requested an explanation for the change in initial RT_{NDT} for several materials in the Limerick Unit 2 reactor pressure vessel. All changes occurred as a result of a detailed review of Certified Material Test Report (CMTR) data.

Discussion of Revisions to Table 1

At PECO Energy's request, the data provided in Table 1 has been plotted to present the distribution in the data set to demonstrate that it represents a normal distribution, i.e., the P-value is greater than 0.05. In order to demonstrate the normality of this data set, the flux values were transformed to natural log. The statistical methods were applied to the natural log values, deriving a P-value of 0.533. The larger the P-value, the greater the confidence that the null hypothesis is true. Since the P-value, 0.533, is much greater than 0.05, the assumption that the distribution is normal is accepted. Figure 2 demonstrates the attributes of the data.

In addition, due to the small sample size of the data set, a correction factor has been determined and applied. This factor of 2.8 was determined using statistical methods provided in NUREG-1475 [3] methods which include 95/95 confidence. As can be seen in Table 1, the Measured Data Average plus 2.8σ (where 2.8 is the factor from [3]) is greater than the Maximum Measured Value and significantly lower than the Predicted Capsule Flux. The Predicted Capsule Flux with a Safety Factor of 2.0 is used for the P-T curve calculations.

Table 1
Flux Wire Measurements from 251" Vessel ID BWR/4&5's
With 764 Fuel Bundles

Plant ID	Measured Capsule Flux (N/cm^2-s)
First Cycle Measurements at Capsule Location	
P	7.9E+8
J	8.2E+8
Y	1.2E+9
AM	1.0E+9
AW	6.3E+8
AX	4.7E+8
AZ	5.0E+8
AK	4.8E+8
AH	4.9E+8
AY	6.2E+8
Surveillance Capsule Flux Measurements	
P	7.50E+08
J	6.80E+08
Y	5.90E+08
AW	6.60E+08
AT	6.70E+08
AX	4.41E+08
AZ	5.22E+08
AK	6.85E+08
AY	7.49E+08
Minimum	**4.4E+08**
Maximum	**1.2E+09**
Average	**Discussed in Text**
Standard Deviation	**Discussed in Text**

251" Vessel ID BWR/4&5's With 764 Fuel Bundles	**Flux (N/cm^2-s)**
Predicted ID surface Flux with Safety Factor of 2.0	**1.7E+09**
Predicted Capsule Flux (ID surface flux * capsule lead factor) (1.7E+09 * 0.94)	**1.6E+09**
Measured Data Average plus 2.8σ*	**1.36E+09**
Maximum Measured Value	**1.2E+09**

*** 2.8σ derived from NUREG-1475, "Applying Statistics" due to sample size**

Table 2

Surveillance Capsule and First Cycle Dosimeter Locations

Plant Code	Capsule Azimuth (°)	First Cycle Dosimeter Azimuth (°)
P	120	30
J	30	30
Y	30	30
AM	(1)	30
AW	30	30
AT	30	(2)
AX	300	30
AZ	300	30
AK	300	30
AH	(1)	30
AY	30	30

(1) Capsules have not been removed for these plants
(2) Data point not included in statistical evaluation

Table 3

Location of Surveillance Capsules/Dosimeters and Corresponding Obstructions

Plant	BWR Type	ID (inches)	Number of Fuel Bundles	Azimuth of Specimen Holder Brackets (°)	Specimen Holder Bracket Azimuth Corresponds to a Jet Pump Riser (N2 Nozzle) Azimuth?
P	4	251	764	30, 120, 300	Yes
J	4	251	764	30, 120, 300	Yes
Y	4	251	764	30, 120, 300	Yes
AM	4	251	764	30, 120, 300	Yes
AW	4	251	764	30, 120, 300	Yes
AT	4	251	764	30, 120, 300	Yes
AX	5	251	764	30, 120, 300	Yes
AZ	5	251	764	30, 120, 300	Yes
AK	5	251	764	30, 120, 300	Yes
AH	4	251	764	30, 120, 300	Yes
AY	4	251	764	30, 120, 300	Yes

Table 4
Location of Welds in the Limerick Unit 2 Reactor Pressure Vessel[1]

Weld Identifier	Weld Direction	Weld Location
AA[2]	Circumferential	Bottom Head Torus to Shell Ring #1
AB[2]	Circumferential	Shell Ring #1 to Shell Ring #2
AC[2]	Circumferential	Shell Ring #2 to Shell Ring #3
AD[2]	Circumferential	Shell Ring #3 to Shell Ring #4
AE[2]	Circumferential	Shell Ring #4 to Shell Ring #5
AF	Circumferential	Shell Ring #5 to Shell Flange
AG	Circumferential	Head Flange to Top Head Torus
AH	Circumferential	Top Head Torus to Top Head Dollar Plate
AJ	Circumferential	Bottom Head Torus to Bottom Head Dollar Plate
BA, BB, BC	Vertical	Shell Ring #1
BD, BE, BF	Vertical	Shell Ring #2
BG, BH, BJ	Vertical	Shell Ring #3
BK, BM	Vertical	Shell Ring #4
BN, BP	Vertical	Shell Ring #5
DA, DB, DC, DD, DE, DF	Vertical	Bottom Head Torus
DG	N/A	Bottom Head Dollar Plate
DH, DJ, DK, DM, DN, DP	Vertical	Top Head Torus
LPCI	N/A	LPCI Nozzle

(1) as presented in Table 4-3 of GE-NE-B11-00836-00-02, Revision 0, "Pressure Temperature Curves for PECO Energy Company Limerick Unit 2", July 2000, General Electric Company, San Jose, CA
(2) Field Girth Welds

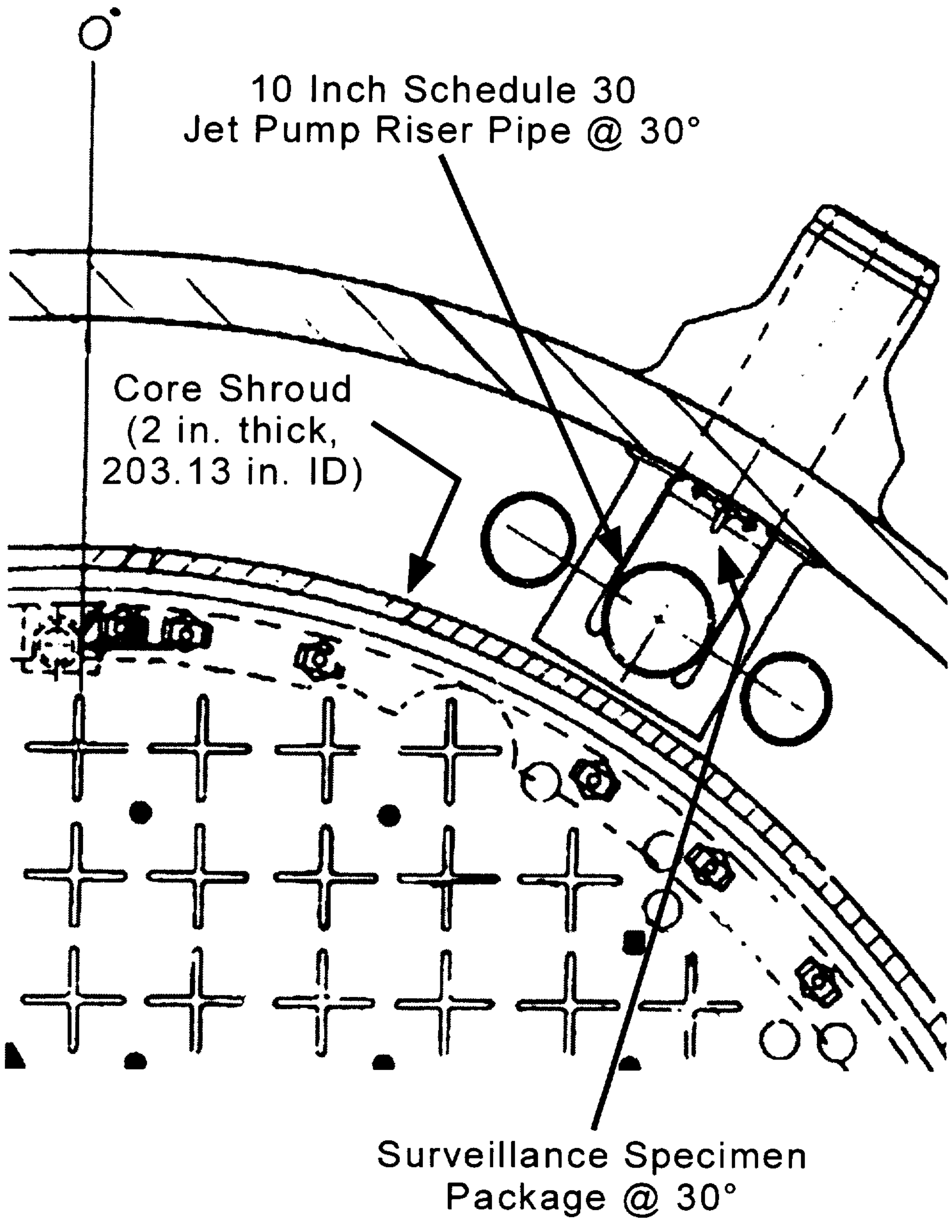
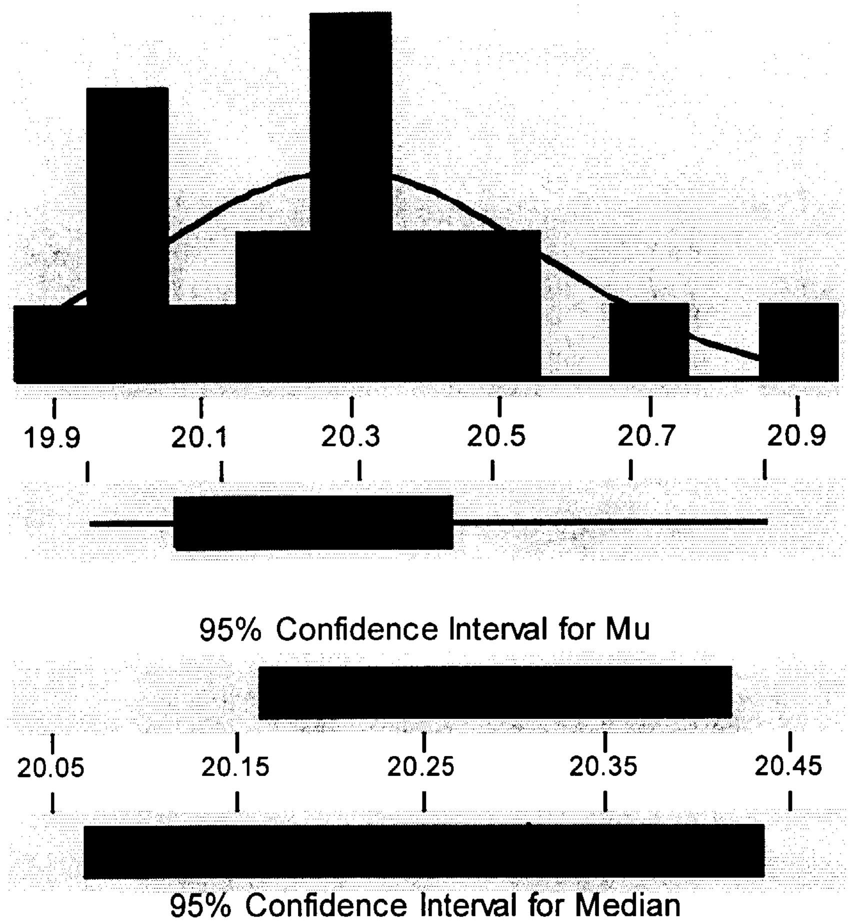


Figure 1

**Typical 251" BWR 4/5 Arrangement at 30° Azimuth
for Plants Contained in Table 1
(also applicable to 120° and 300° azimuths)**

Descriptive Statistics



Variable: ln flux

Anderson-Darling Normality Test

A-Squared:	0.306
P-Value:	0.533
Mean	20.2896
StDev	0.2642
Variance	6.98E-02
Skewness	0.588690
Kurtosis	0.225173
N	19
Minimum	19.9046
1st Quartile	20.0301
Median	20.3078
3rd Quartile	20.4356
Maximum	20.9056

95% Confidence Interval for Mu

20.1622	20.4169

95% Confidence Interval for Sigma

0.1997	0.3908

95% Confidence Interval for Median

20.0673	20.4344

Figure 2
Flux Data Normality Test

ATTACHMENT 6

LIMERICK GENERATING STATION
UNIT 2

DOCKET NO.
50-353

LICENSE NO.
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-2

November 20, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

General Electric Company Report
GE-NE-B11-00836-00-02
(proprietary information; affidavit enclosed)

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Technical Projects Manager, Technical Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report GE-NE-B11-00836-00-02, Revision 0, *Pressure-Temperature Curves for PECO Energy Company, Limerick Unit 2,* Class III (GE Proprietary Information), dated July 2000. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

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

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

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed methods and processes, which GE has developed and applied to pressure-temperature curves for the BWR over a number of years.

The development of the BWR pressure-temperature curves was achieved at a significant cost, on the order of ¾ million dollars, to GE. The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

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

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

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

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 20th day of JULY 2000.

David J. Robare
General Electric Company

Subscribed and sworn before me this 20th day of JULY 2000.

Notary Public, State of California

ANNA HANLIN
Commission # 1184507
Notary Public - California
Santa Clara County
My Comm. Expires Jun 19, 2002