Station Support Department

10 CFR 50.90
10 CFR 50.12

PECO NUCLEAR

A Unit of PECO Energy

PECO Energy Company
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

May 15, 2000

Docket No. 50-352

License No. NPF-39

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

Subject: Limerick Generating Station, Unit 1
Technical Specifications Change Request No. 00-02-1
Changes to Reactor Pressure Vessel Pressure-Temperature Limits

References:

1. Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) "Oconee Nuclear Station, Units 1, 2, and 3 RE: Exemption From the Requirements of 10 CFR Part 50, Section 50.60(a) (TAC NOS. MA5473, MA5474, and MA5475)" dated July 29, 1999

2. Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) Amendment No. 307 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 dated October 1, 1999.

3. Letter from J. P. Dimmette, Jr. (Commonwealth Edison) to U.S. Nuclear Regulatory Commission, Quad Cities Nuclear Power Station, Units 1 and 2, Request for an Amendment to Technical Specifications and Request for Exemption from 10CFR50.60, dated November 12, 1999.

4. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Commonwealth Edison), "Quad Cities - Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G (TAC Nos. MA7140 and MA7141)," dated February 4, 2000.

5. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Commonwealth Edison), "Quad Cities - Issuance of Amendments - Revised Pressure-Temperature Limits (TAC Nos. MA7138 and MA7139)," dated February 4, 2000.

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Dear Sir/Madam:

PECO Energy Company (PECO Energy) is submitting Technical Specifications Change Request No. 00-02-1, in accordance with 10 CFR 50.90, requesting a change to Appendix A of Facility Operating License No. NPF-39 for Limerick Generating Station (LGS), Unit 1. The proposed change is to LGS Unit 1 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 1 to a maximum of 32 Effective Full Power Years (EFPY). The use of 32 EFPY conservatively bounds Unit 1 which is currently at approximately 11.5 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 of ASME B&PV Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." ASME Code Case N-588 allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. The code case was approved for use by the ASME on December 12, 1997. 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 these Code Cases results in a reduction in allowable temperatures, for a given pressure, than would have been required without the use of the Code Cases. 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 these Code Cases have not yet received USNRC approval for generic usage, Technical Specification amendments using these Code Cases were recently approved by the NRC for Duke Energy, Oconee Nuclear Station (Reference 2) and for Commonwealth Edison, Quad Cities Nuclear Power Station (Reference 5).

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 Cases N-588 and N-640. Similar exemptions were granted to Duke Energy, Oconee Nuclear Station (Reference 1) and Commonwealth Edison, Quad Cities Nuclear Power Station (Reference 4).

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 Cases N-640 and N-588.

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 Commonwealth Edison (Reference 3).

6. Attachment 6 to this letter provides General Electric (GE) Report GE-NE-B11-00836-00-01, "*Pressure-Temperature Curves for PECO Energy Company, Limerick Generating Station, Unit 1*," Rev. 0, dated April, 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. A *non-proprietary* version of the report is in preparation and will be submitted upon completion.

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 1, TS are valid until 12 EFPY which currently corresponds to a date of October 1, 2000. If the P-T curves for LGS, Unit 1, are not revised by this date, the P-T limitations will no longer be valid and LGS, Unit 1, will be required to shut down. Therefore, we request approval of this amendment prior to October 1, 2000, to ensure the validity of the P-T limitations and to support continued operation of LGS, Unit 1. If approved, we request that the changes become effective within 30 days of issuance, but no later than October 1, 2000.

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

Sincerely,

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. W. Langenbach, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company, the Applicant herein; that he has read the enclosed Technical Specifications Change Request No. 00-02-1, "Changes to Reactor Pressure Vessel Pressure-Temperature Limits," for Limerick Generating Station, Unit 1, Facility Operating License No. NPF-39, 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.

James W Langenbach

Vice President

Subscribed and sworn to

before me this 15th day

of May, 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 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 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 1, 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 1 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.

<u>TS Bases Table B3/4.4.6-1:</u>

- Change Heat C7677-1 ΔRT_{NDT} from +69°F to +35°F
- Change Weld AB ΔRT_{NDT} from +114°F to +58°F

The current values (+69°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 top head flange RT_{NDT} from 0°F to +10°F
- Change the vessel flange RT_{NDT} from -30°F to -20°F
- Change the FW nozzle RT_{NDT} from -10°F to -20°F
- Change the limiting non-beltline weld RT_{NDT} from 0°F to -12°F
- Change the EOL RT_{NDT} for the LPCI nozzle from +42°F to +41°F

These new RPV material property values are documented in General Electric Report GE-NE-B11-00836-00-01.

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 12 EFPY for LGS, Unit 1. 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 1 which is currently at approximately 11.5 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 50°F for LGS, Unit 1 (at 12 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-588 and 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-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." ASME Code Case N-588 allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. The code case was approved for use by the ASME on December 12, 1997. 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 these Code Cases results in a reduction in allowable temperatures, for a given pressure, than would have been required without the use of the Code Cases. 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 these Code Cases have not yet received USNRC approval for generic usage, Technical Specification amendments using these Code Cases were recently approved by the NRC for Duke Energy, Oconee Nuclear Station (Reference 2) and for Commonwealth Edison, Quad Cities Nuclear Power Station (Reference 5).

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 Cases N-588 and N-640 and is provided in Attachment 4.

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-B11-00836-00-01 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 Cases N-588 and N-640.

ASME Code Case N-588

The current ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G approach requires the consideration of an axially oriented flaw in circumferential welds for the purpose of calculating pressure-temperature limits. Postulating the ASME Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the reactor pressure vessel (RPV) thickness, and is much longer than the width of the vessel circumferential welds. The fabrication of reactor pressure vessels (RPVs) for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Subsequent non-destructive examinations were conducted which confirmed that the welds met the pre-service inspection criteria. Experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual RPV welds, confirm that any remaining flaws are small, laminar in nature, and do not cross traverse to the weld bead orientation. Because of this, any defects potentially introduced during the fabrication process and not detected during the subsequent non-destructive examinations should only be oriented along the direction of weld fabrication. For circumferential welds, this indicates a postulated defect with a circumferential orientation.

Using ASME Code Case N-588 to determine P-T limits in conjunction with ASME B&PV Code, Section XI, Appendix G, provides appropriate and conservative procedures to determine limiting maximum postulated defects and to consider those defects in the determination of the P-T limits. The application of this code case maintains the margin of safety for circumferential welds equivalent to that originally contemplated for plates/forgings and axial welds.

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.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed changes to the Limerick Generating Station (LGS), Unit 1 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 1 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. The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   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 Cases N-640 and N-588. 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 Cases N-640 and N-588. 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 1 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 1 Technical Specifications, and have concluded that they do not involve an unreviewed safety question, they do not involve a Significant Hazards Consideration, and they will not endanger the health and safety of the public.

ATTACHMENT 2

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 2000

AFFECTED PAGES
(Mark-ups)

UNIT 1

3/4 4-18
3/4 4-20
B 3/4 4-5
B 3/4 4-7

ATTACHMENT TO
ECR No. LG 00-00194 REV. 0
PAGE 20 OF

Refer to Position # 12

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) curve A ~~and 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: 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 80°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 curve A ~~and A'~~, B, or C as applicable, at least once per 30 minutes.

DELETE

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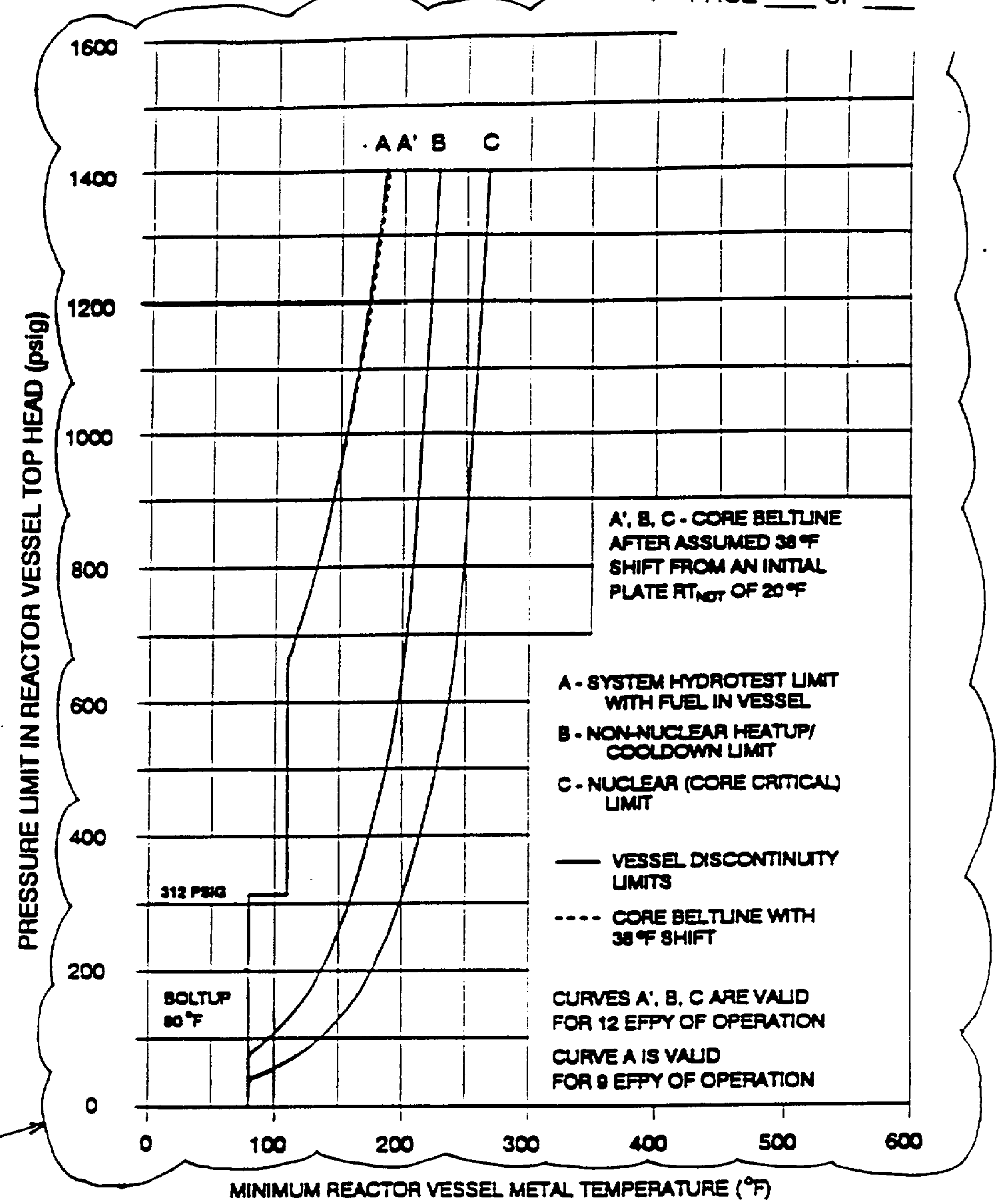

REVISE AS SHOWN ON ATTACHMENT

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

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ATTACHMENT TO
ECR No. LG 00-00194 REV. 0
PAGE 23 OF ___

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 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." REPLACE

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 curves Figure 3.4.6.1-1, ~~curve A~~ include a shift in RT_{NDT} for conditions at ~~8~~ 32 EFPY. The A[1], B and C ~~limit curves are predicted to be bounding for all areas of the RPV until 12~~ 32 EFPY. ~~when the beltline material's RT_{NDT} will shift due to neutron fluence and the beltline curves will intersect the non-beltline discontinuity curves.~~ IN ADDITION, AN INTERMEDIATE A CURVE HAS BEEN PROVIDED FOR 22 EFPY. ADD

DELETE

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 flux wires, 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 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*

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	C 7677-1	.11	.5	+20	~~+69~~ +35	NA	+89
Weld	AB (Field Weld)	640892/ J424B27AE	.09	1.0	-60	~~+114~~ +58	NA	+54

NOTES: * Based on 110% of original rated 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)
Shell Ring	SA 533, Gr. B, CL. 1	C7711-1	+20
Bottom Head Dome	"	C7973-1	+12
Bottom Head Torus	"	C7973-1	+12
Top Head Dome	"	A6834-1	+10
Top Head Torus	"	B1993-1	+10
Top Head Flange	SA-508, CL. 2	123B195-289	~~0~~ +10
Vessel Flange	"	2V1924-302	~~-30~~ -20
Feedwater Nozzle	"	Q2Q22W-412	~~-10~~ -20
Weld	Non-Beltline	All	~~0~~ -12
LPCI Nozzle***	SA-508, CL. 2	Q2Q25W	-6
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. E

Note: *** 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 ~~+42~~°F. +41

FEB 12 1996

ATTACHMENT TO
ECR No. LG 00-00194 REV. 0
PAGE 24 OF

ATTACHMENT 3

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 2000

AFFECTED PAGES
(Camera-ready)

<u>UNIT 1</u>

3/4 4-18
3/4 4-20
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 80°F when reactor vessel head bolting studs are under tension.

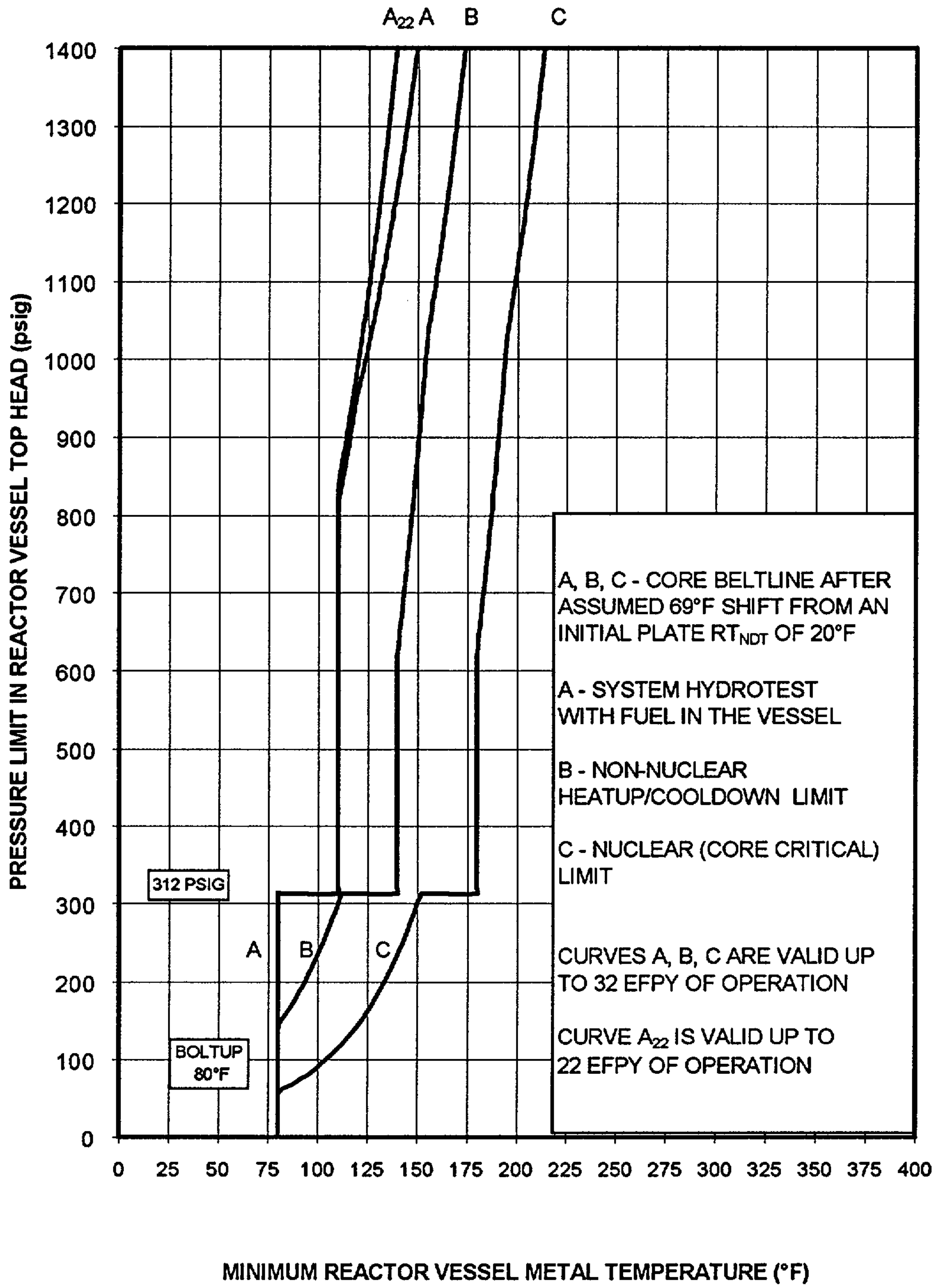
APPLICABILITY: At all times.

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 curve A, B, or C as applicable, at least once per 30 minutes.



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 curves, Figure 3.4.6.1-1, include a shift in RT_{NDT} for conditions at 32 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY. In addition, an intermediate A curve has been provided for 22 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 flux wires, 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*

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	C 7677-1	.11	.5	+20	+35	NA	+89
Weld	AB (Field Weld)	640892/ J424B27AE	.09	1.0	-60	+58	NA	+54

NOTES: * Based on 110% of original rated 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)
Shell Ring	SA 533, Gr. B, CL. 1	C7711-1	+20
Bottom Head Dome	"	C7973-1	+12
Bottom Head Torus	"	C7973-1	+12
Top Head Dome	"	A6834-1	+10
Top Head Torus	"	B1993-1	+10
Top Head Flange	SA-508, CL. 2	123B195-289	+10
Vessel Flange	"	2V1924-302	-20
Feedwater Nozzle	"	Q2Q22W-412	-20
Weld	Non-Beltline	All	-12
LPCI Nozzle***	SA-508, CL. 2	Q2Q25W	-6
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. Exp. at +10°F

Note: *** 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 +41°F.

ATTACHMENT 4

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

Information Supporting a Request for Exemption
from the Requirements of 10 CFR 50.60(a) - 6 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," and ASME B&PV Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, 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

10 CFR 50.12(a) Requirements

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. The requested exemption is authorized by law. 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. The requested exemption does not present an undue risk to the public health and safety. The revised pressure-temperature (P-T) limits being proposed for Limerick Generating Station, Unit 1, 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 1, 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 1, 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 1, will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

Justification for Use of Code Case N-588

10 CFR 50.12(a) Requirements:

The requested exemption to allow use of ASME Code Case N-588 to determine stress intensity factors for postulated flaws and postulated flaw orientation for circumferential welds 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 satisfied:

1. The requested exemption is authorized by law: 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. The requested exemption does not present an undue risk to the public health and safety: 10 CFR 50, Appendix G, requires that Article G-2120 of ASME B&PV Code, Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels (RPV) for the vessel pressure-temperature limits. These limits are determined for normal operation and pressure/leak test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the RPV material and normal (i.e., perpendicular in the plane of the material) to the direction of maximum stress. ASME B&PV Code, Section XI, Appendix G, also provides methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure-temperature limits.

   Code Case N-588 provides benefits, in terms of calculating P-T limits, by revising the Article G-2120 reference flaw orientation for circumferential welds in reactor pressure vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for defects that could physically exist within the geometry of the weldment. The current ASME Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P-T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the reactor pressure vessel wall thickness, which is much longer than the width of circumferential welds. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that defects be postulated in plates/forgings and axial welds.

   The fabrication of reactor pressure vessels for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during pre-service inspection, in-service non-destructive examinations and data taken from destructive examination of actual reactor pressure vessel welds, confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent non-destructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

   ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing reactor pressure vessel P-T limits per ASME B&PV Code, Section XI, Appendix G procedures. The procedures allowed by ASME Code Case N-588 are conservative and provide a margin of safety in the development of reactor pressure vessel pressure-temperature operating and pressure test limits, which will prevent non-ductile fracture of the reactor pressure vessel.

The proposed P-T limits include restrictions on allowable operating conditions and equipment operability requirements 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, "Pressure/Temperature Limits." Therefore, this requested exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by 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 exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) - demonstrates that 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.

   10CFR50.12(a)(2)(ii): The underlying purpose of 10 CFR 50, Appendix G and ASME B&PV Code, Section XI, Appendix G, is to satisfy the underlying requirement that:

   1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that 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 curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

   Application of ASME Code Case N-588 when determining P-T operating and test limit curves per ASME B&PV Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P-T limits. This application of the code case maintains the margin of safety originally contemplated when ASME B&PV Code, Section XI, Appendix G was developed.

   Therefore, use of ASME Code Case N-588, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

10CFR50.12(a)(2)(iii): The Reactor Pressure Vessel pressure-temperature operating window is defined by the P-T operating and test curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of with these P-T curves without the relief provided by ASME Code Case N-588 would unnecessarily restrict the pressure-temperature operating window for Limerick Generating Station, Unit 1. 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-588 in the development the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

10CFR50.12(a)(2)(v): The requested exemption provides only temporary relief from the applicable regulation and Limerick Generating Station, Unit 1, has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME Code Case N-588 for use by the nuclear industry.

ASME Code Case N-588, Conclusion for Exemption Acceptability: Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME B&PV Code, Section XI, Appendix G was developed and imposes restrictions on P-T operating limits beyond those originally contemplated.

This proposed alternative is acceptable because the 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-588 for Limerick Generating Station, Unit 1, 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 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 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 Commonwealth Edison (Reference 3) - 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.

<u>Overall Plant Safety is Improved</u>

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.

<u>Impact on PWRs:</u>

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.

<u>Impact on BWRs:</u>

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.

<u>Reactor Vessel Fracture Margins</u>

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.

Impact of the Change on P-T Curves

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.

Summary and Conclusions

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.

References

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.

4. McCabe, D.E., "Assessment of Metallurgical Effects that Impact Pressure Vessel Safe Margin Issues", Oak Ridge Report ORNL/NRC/LTR-94/26, October 1994.

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

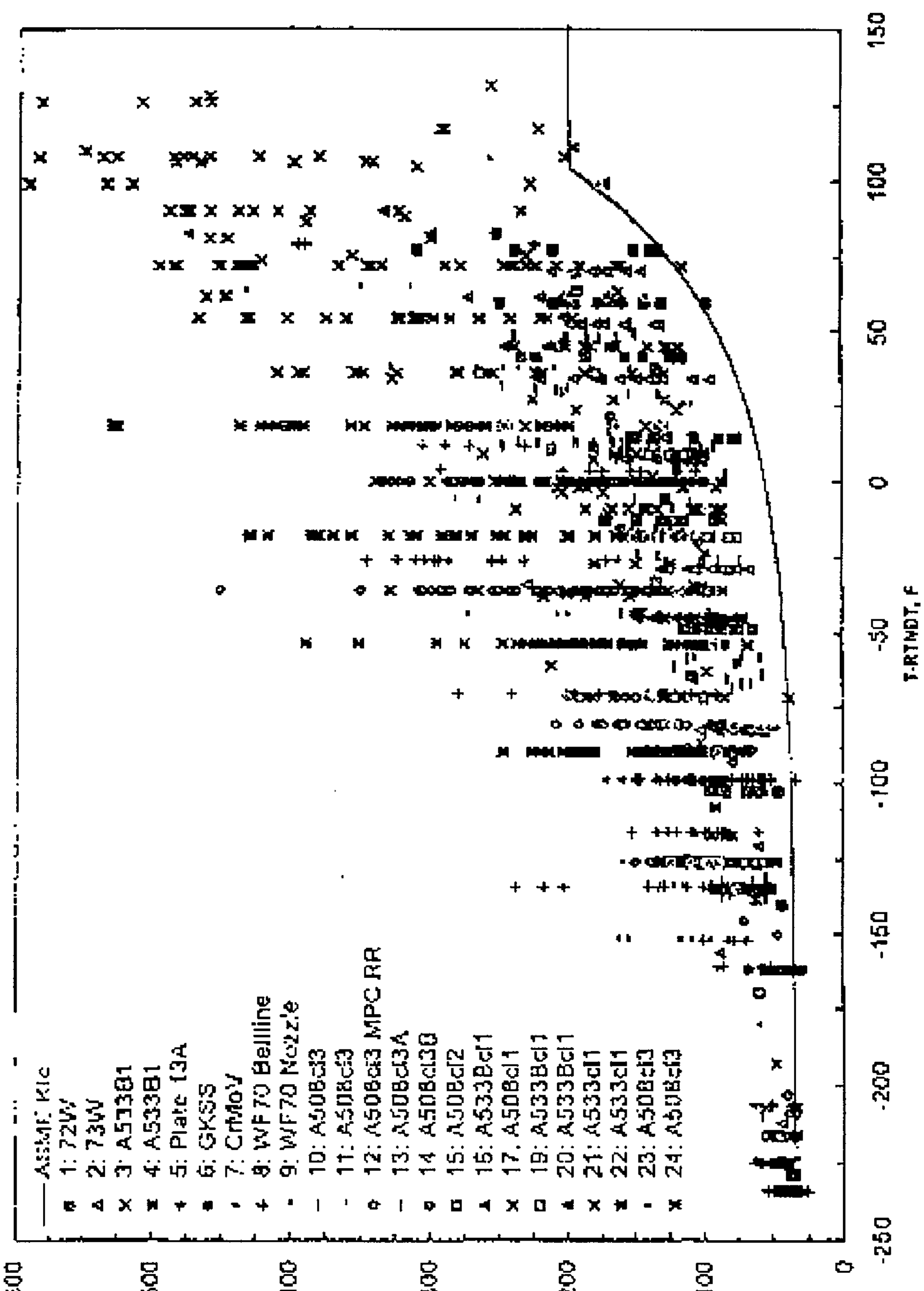
20 Degree/hour Cooldown of Axial Flaw

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]

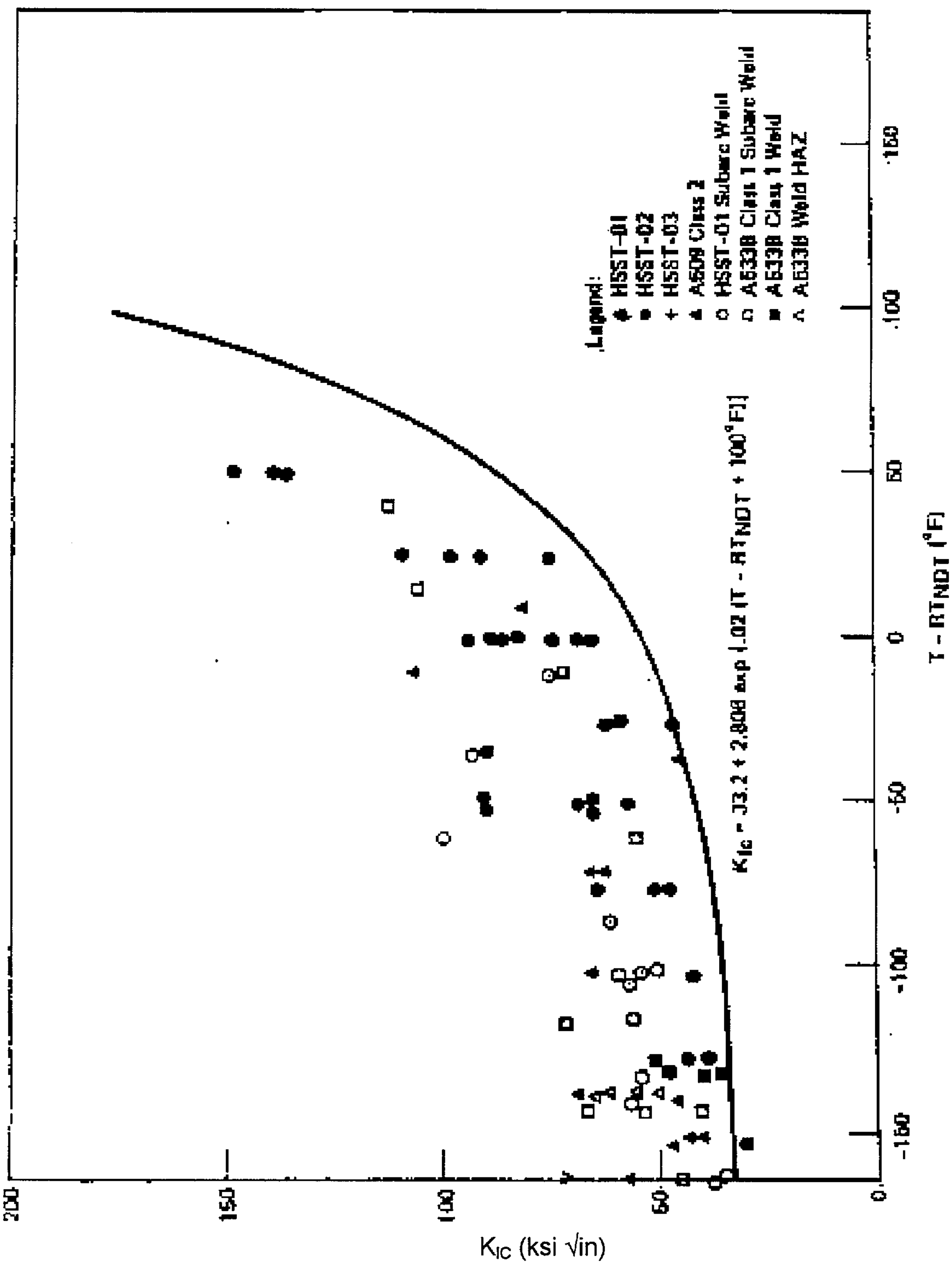


Figure 2. Original K_{IC} Reference Toughness Curve, with Supporting Data [2]

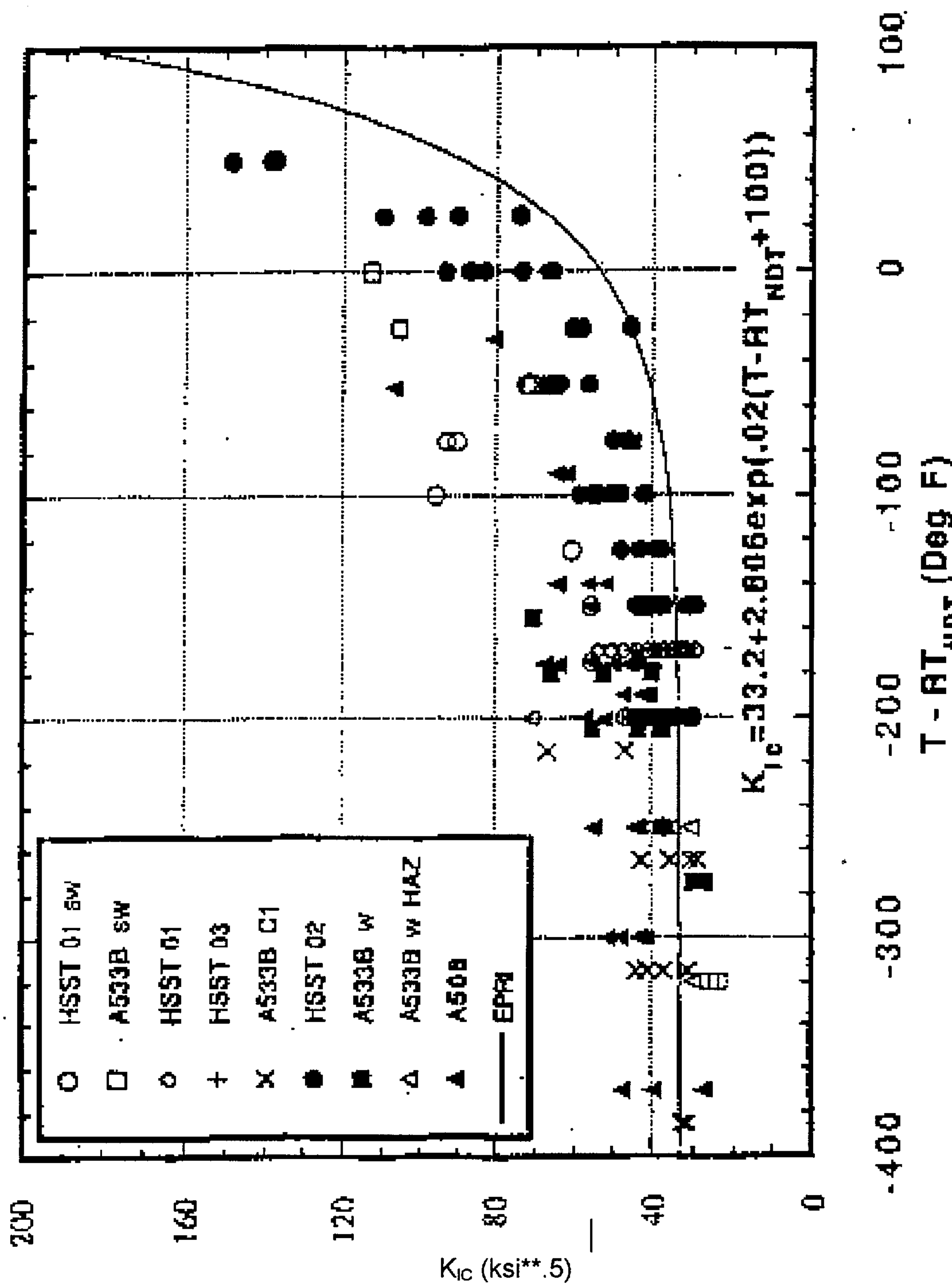


Figure 3. K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

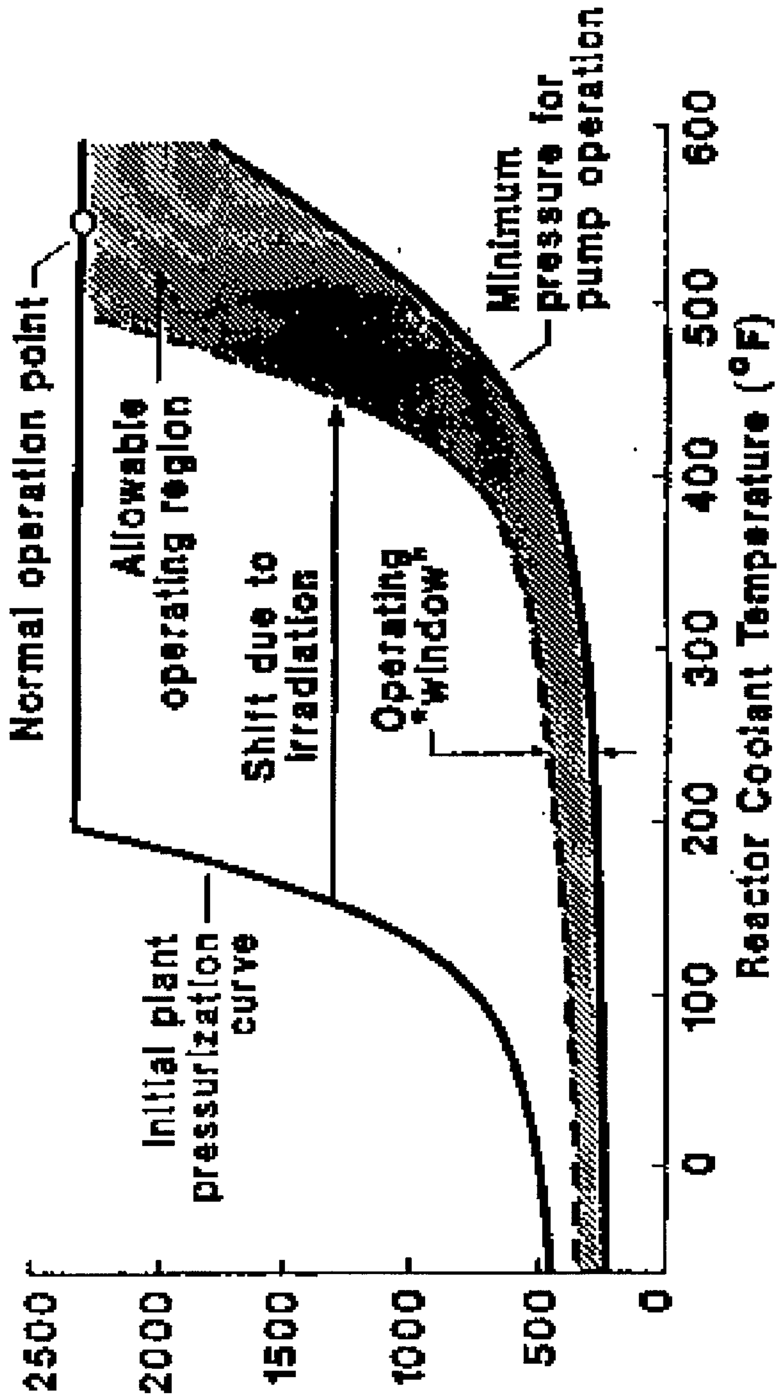


Figure 4. Operating Window From P-T Limit Curves [4]

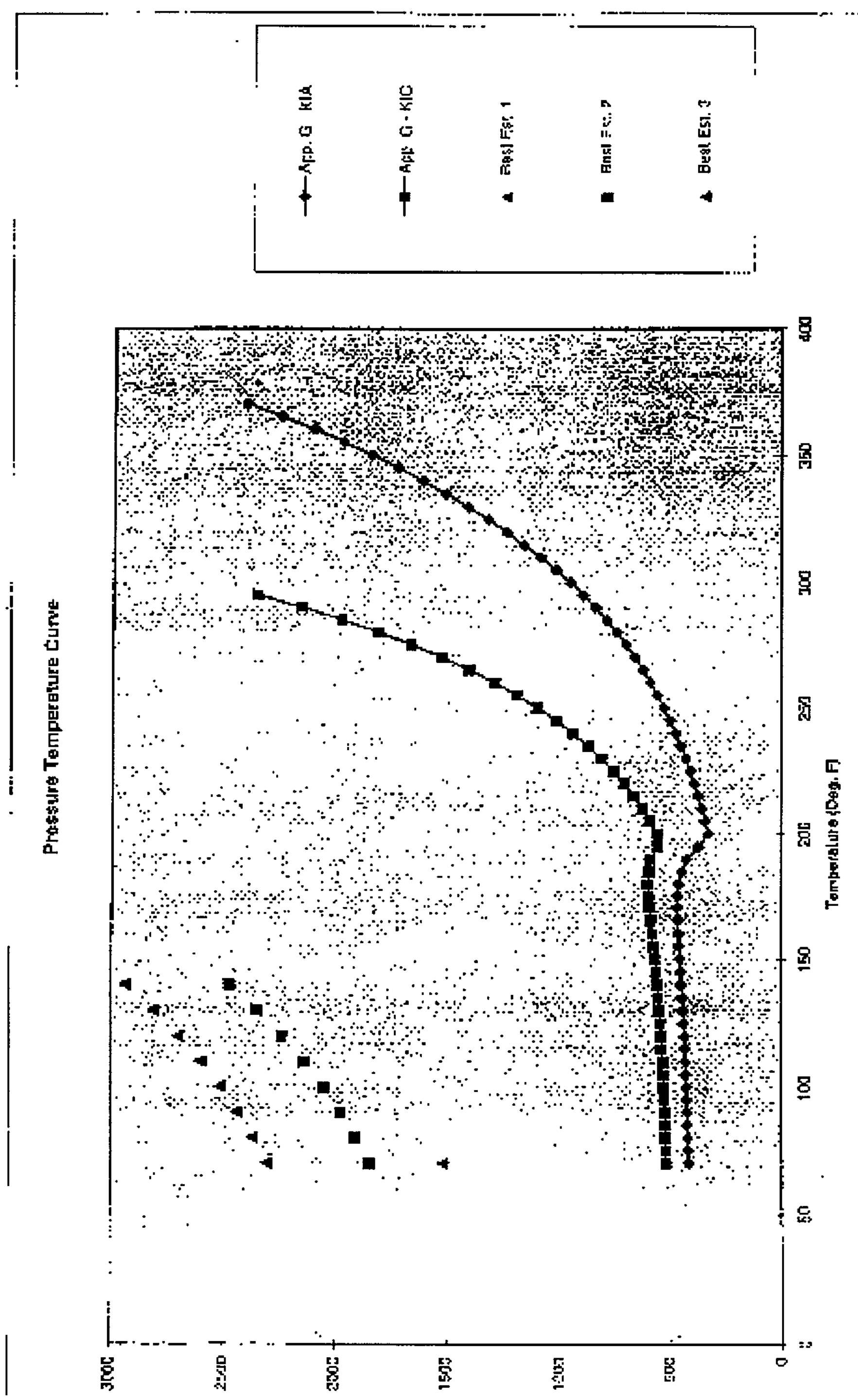


Figure 5. P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

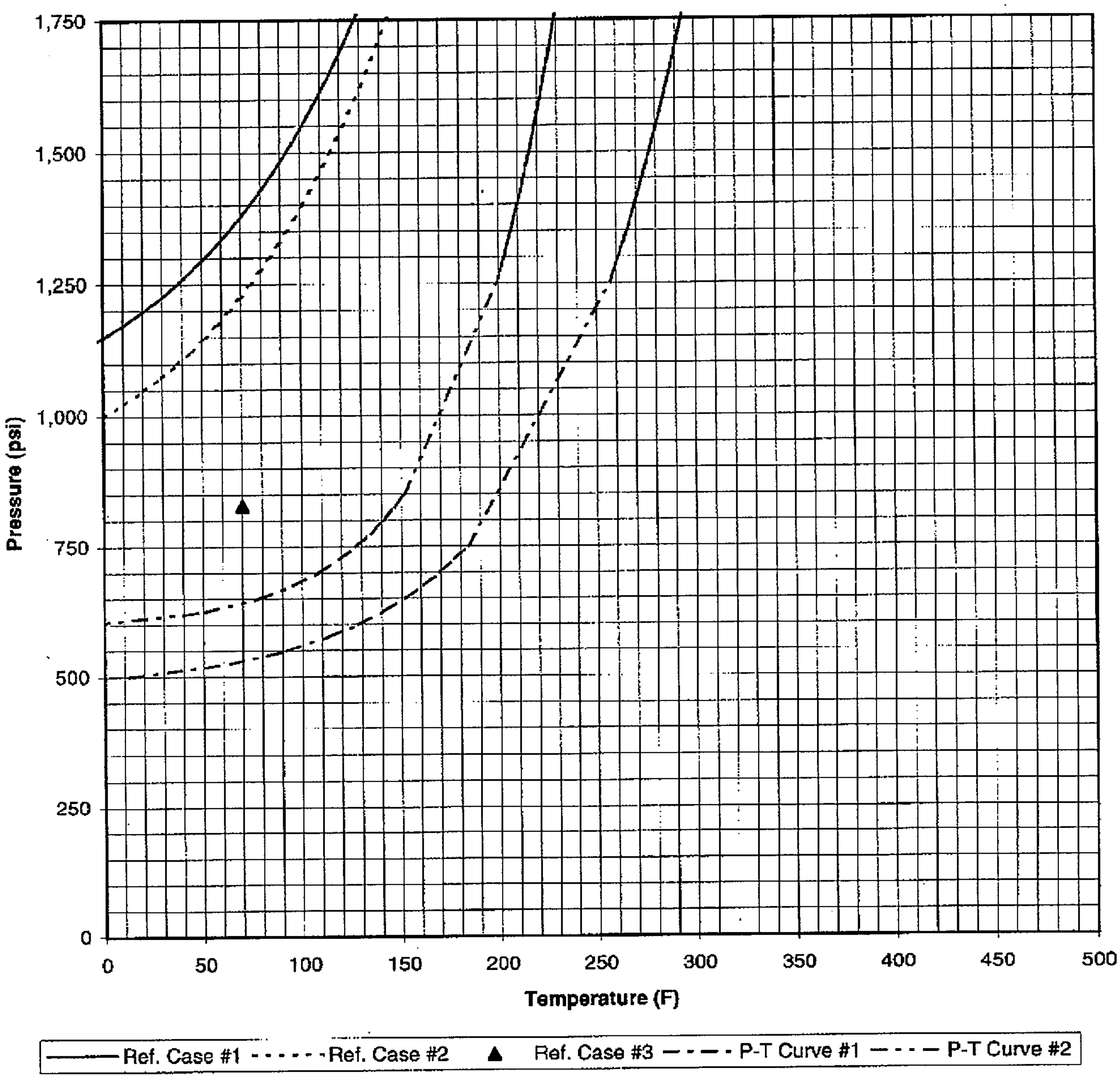


Figure 6. P-T Limit Curves Illustrating Deterministic Safety Factors
for a BWR Reactor Vessel

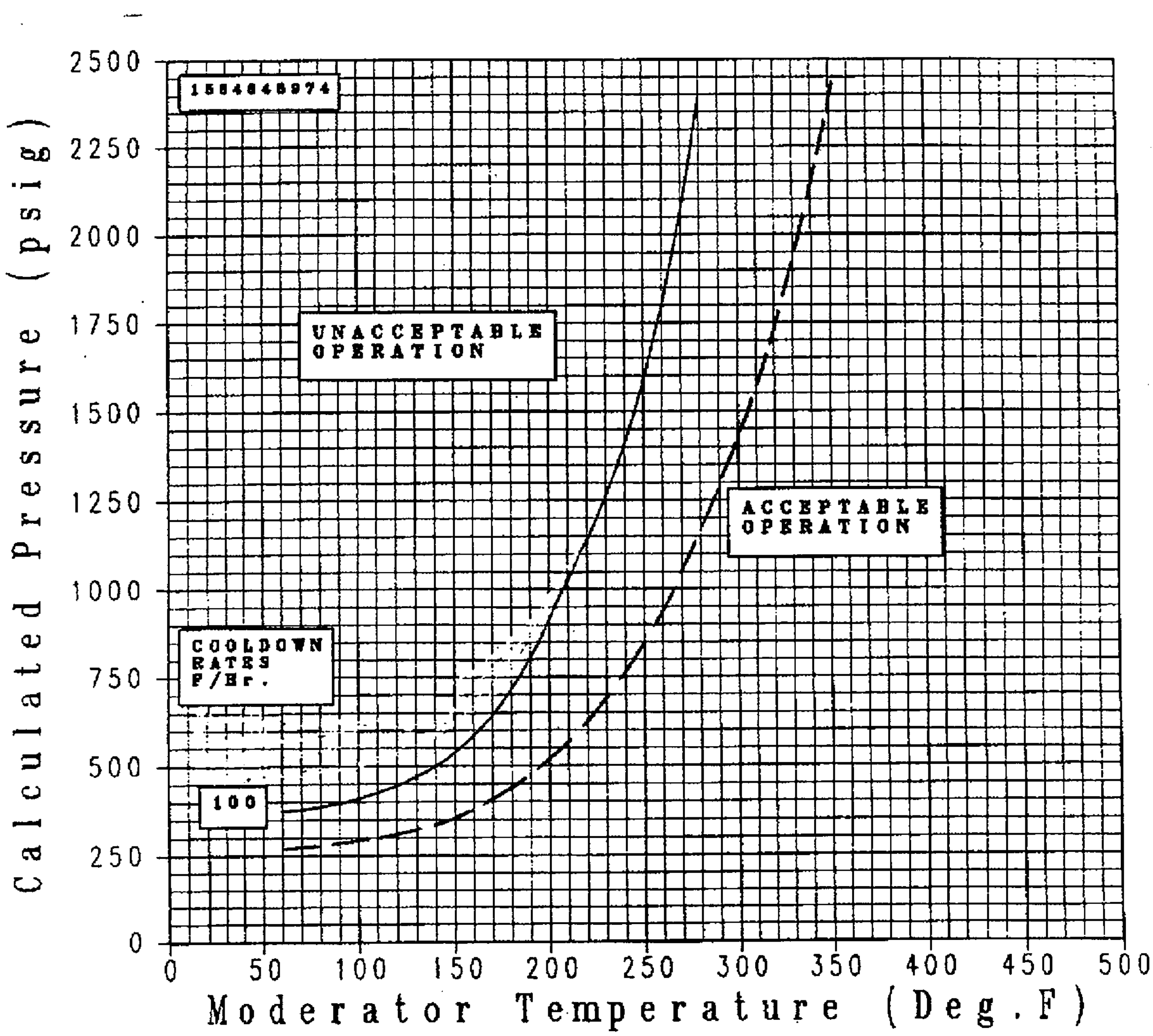


Figure 7. Comparison of Cool-Down Curves for the Existing and Proposed Methods - PWR [Dashed Curve = Existing (K_{Ia}) and Solid Curve = Proposed (K_{Ic})]

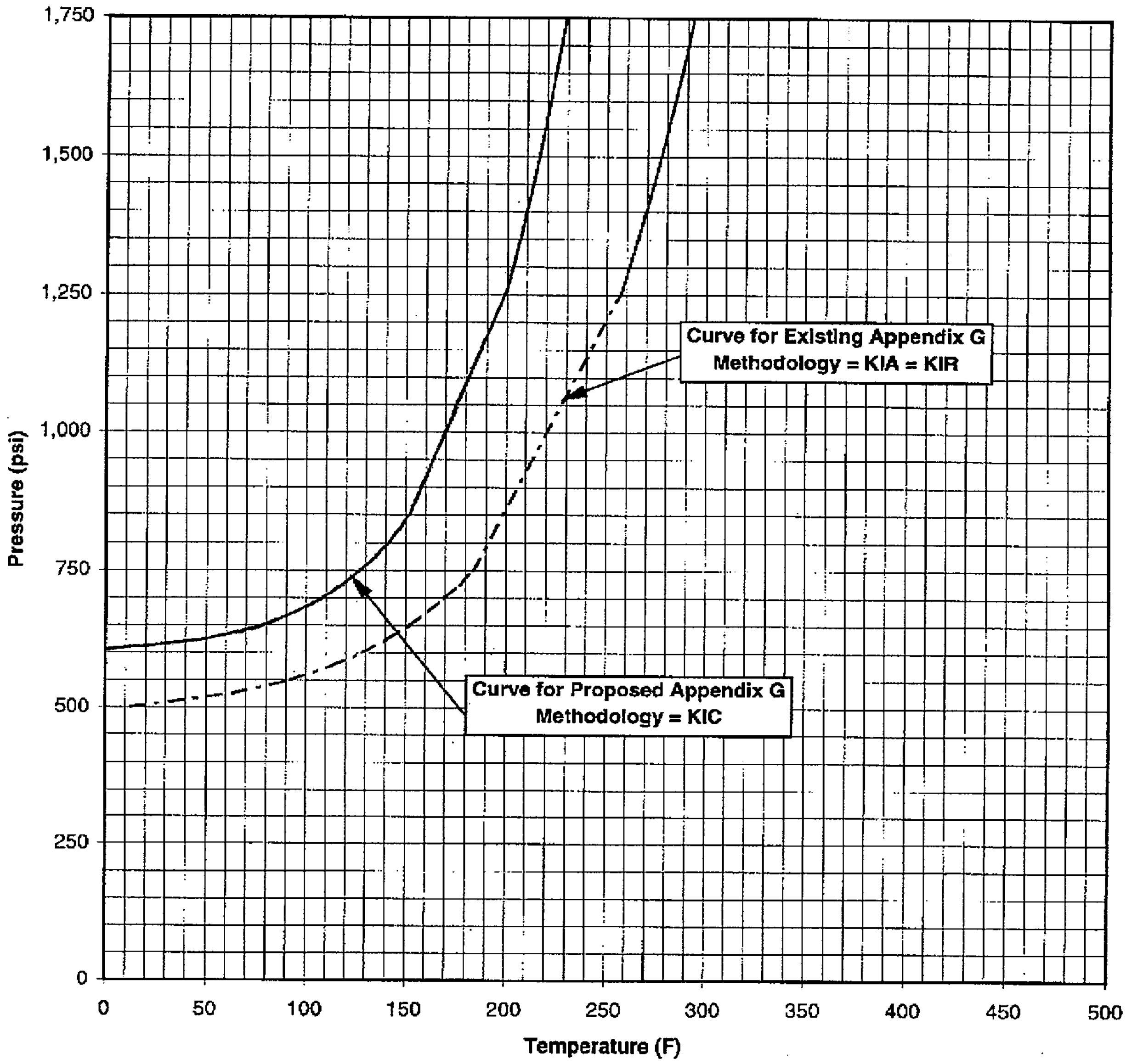


Figure 8. Comparison of Hydrotest P-T Curves for the Existing and Proposed Methods - BWR [Dashed Curve=Existing (K_{Ia}) and Solid Curve=Proposed (K_{Ic})]

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

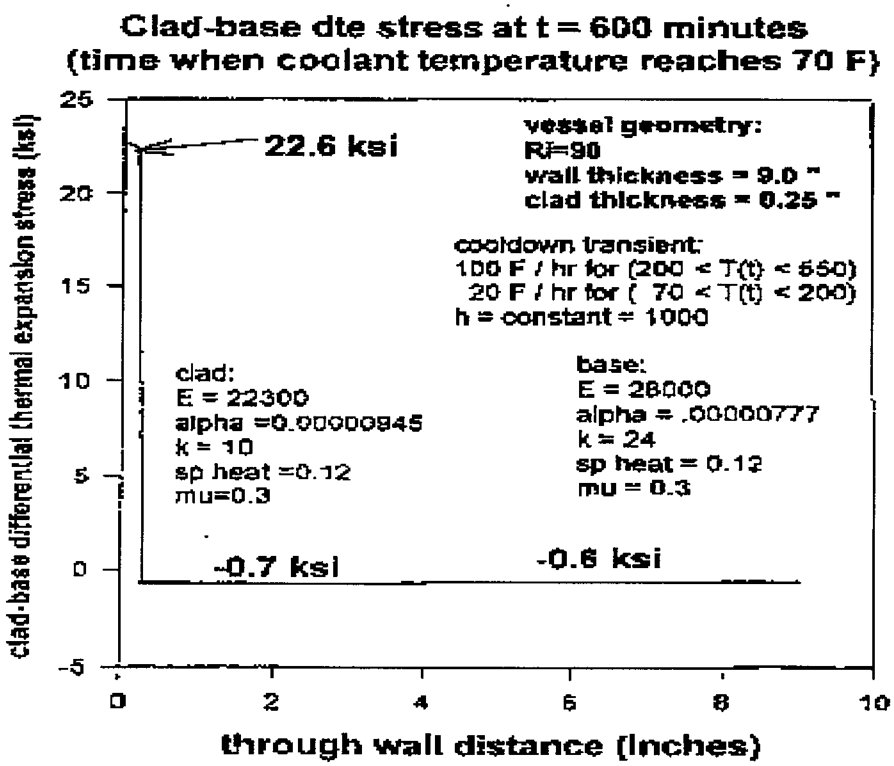


Figure A-1: Clad-base metal stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

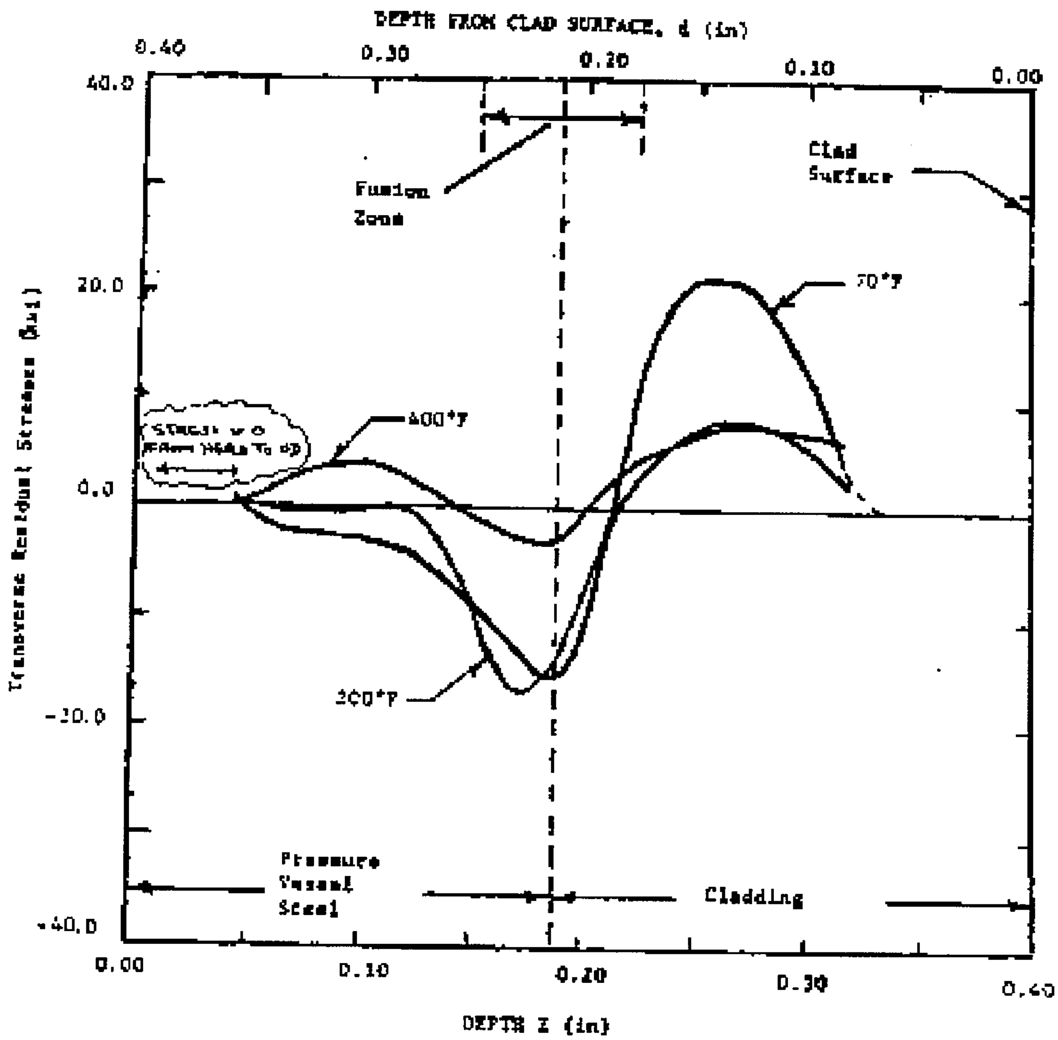


Figure A-2: Residual Stresses Transverse to Direction of Welding

ATTACHMENT 6

LIMERICK GENERATING STATION
UNIT 1

DOCKET NO.
50-352

LICENSE NO.
NPF-39

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NO. 00-02-1

May 15, 2000

"Changes to Reactor Pressure Vessel Pressure-Temperature Limits"

General Electric Company Report
GE-NE-B11-00836-00-01
(proprietary information; affidavit enclosed)

General Electric Company

AFFIDAVIT

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

(1) I am Technical Account 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-B11-00836-00-01, *Pressure-Temperature Curves for PECO Energy, Limerick Unit 1,* Revision 0, Class III (GE Proprietary Information), dated April 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 26TH day of APRIL 2000.

David J. Robare
David J. Robare
General Electric Company

Subscribed and sworn before me this 26th day of April 2000.

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003

Vicky D Schroer
Notary Public, State of California