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January 24, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47
River Bend Station Licensing Amendment Request (LAR 2000-26), "Revision to reactor vessel pressure/temperature (P/T or P-T) limits", and application of ASME Code Case N640.

File Nos.: G9.5, G9.42

References: 1) Letter from J. B. Hopkins (USNRC) to M. Reandeau (Clinton Power Station), "Clinton Power Station, Unit 1 - Issuance of Amendment", dated October 31, 2000.

2) Letter from S.N. Bailey (USNRC) to O. D. Kingsley (ComEd), "Quad Cities - Issuance of Amendments - Revised Pressure-Temperature Limit," dated February 4, 2000.

RBEXEC-01-010
RBF1-01-0010
RBG-45633

Gentlemen:

In accordance with 10CFR50.90, Entergy Operations, Inc. (EOI) hereby applies for an amendment of the River Bend Station (RBS) Facility Operating License No.NPF-47. This request consists of changes to the Technical Specifications (TS) to revise the reactor vessel pressure/temperature (P/T or P-T) limits specified in TS 3.4.11, "RCS Pressure and Temperature (P/T) Limits", for reactor heatup, cooldown, and critical operation as well as for inservice leak and hydrostatic tests for the reactor coolant system (RCS). Per the proposed changes, the current RCS P/T limits in TS figure 3.4-11, "Minimum Temperature Required Vs. RCS Pressure", would be replaced with recalculated RCS P/T limits based, in part, on an alternative methodology.

Use of the alternative methodology requires an exemption from the current requirements of 10CFR50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," pursuant to 10 CFR 50.60(b) and 10 CFR 50.12, "Specific Exemptions".

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Pursuant to 10CFR50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) requests authorization to use ASME Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," at River Bend Station (RBS), as documented in the Technical Basis for Revised P-T Limit Curve Methodology (see attachment 3). ASME Code Case N-640 permits the use of alternate reference fracture toughness for reactor vessel materials in determining the pressure-temperature (P-T) limits. These limits are reflected in the RBS Technical Specifications (TS).

The NRC has recently approved similar changes for Quad Cities and Clinton Stations (Reference 1 and 2).

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The proposed change introduces no new commitments. The bases for these determinations are included in the attached submittal.

The information supporting the proposed Technical Specification changes and use of Code Case N640 are provided in attachments to this letter. Attachment 1 provides the description and justification for the proposed changes, a finding of no significant hazards consideration and environmental impact consideration for the proposed changes. Attachment 2 contains marked up TS pages reflecting the proposed changes. Attachment 3 contains Request for use of Code Case N-640. Attachment 4 provides General Electric (GE) Nuclear Energy report GE-NE-B13-02094-00-01, "Pressure-Temperature Curves for Entergy Operations Inc. (EOI) Using the K_{Ic} Methodology". GE-NE-B13-02094-00-01 contains information that is proprietary to GE. Consistent with the proprietary information notice provided in the preface of the report, General Electric requests the information provided by the report be withheld from public disclosure pursuant to 10 CFR 2.790(a)(4).

Entergy Operations requests the NRC approve this amendment request and use of Code Case N-640 on or before September 1, 2001, such that it may be implemented prior Refueling Outage 10, which is scheduled in the Fall of 2001.

If you have any questions regarding this request or require additional information, please contact Mr. Gregory P. Norris of the EOI Corporate Nuclear Safety and Licensing staff at 225-336-6391.

Pursuant to 28 U.S.C.A. Section 1746, I declare under penalty of perjury that the foregoing is true and correct. Executed on January 24, 2001.

Very truly yours,

Randall K. Edington

RKE/RJK/GPN
Attachments (4)

cc:

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ATTACHMENT 1

TO

LETTER NO. RBF1-01-0010

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

LICENSING DOCUMENT INVOLVED

River Bend Station (RBS) Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

DESCRIPTION OF PROPOSED CHANGES

The proposed change contained in this license amendment request is a change to Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits" to replace figure 3.4.11-1 with a revised figure containing recalculated curves based on new methodology. These new curves for specifying the required temperature limits are established to the requirements of 10CFR50, Appendix G and will continue to ensure margin to the brittle fracture temperature. One of the primary effects of the revised curves is to permit reactor vessel inservice hydrostatic and leak tests to be performed at a lower temperature at applicable vessel pressures. The revised P/T limits (as proposed) are based, in part, on application of American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative to Reference Fracture Toughness for Development of P/T Limit Curves for ASME B&PV Code Section XI, Division 1." This code case provides alternative methods to those currently approved by the NRC and recognized per 10 CFR 50.60. The use and acceptability of these alternative methods therefore require an exemption from 10 CFR 50.60 requirements. The request for this exemption is further addressed in Attachment 3.

BASIS FOR PROPOSED CHANGE

Entergy Operations, Inc. contracted with General Electric Company (GE) to recalculate the P/T limit curves for River Bend Station. The methodology used to generate the new P/T limit curves was similar to the methodology previously used to generate the current P/T limit curves of TS Figure 3.4.11-1. However, several improvements or modifications were made to the P/T limit curve methodology to remove excess conservatism associated with the current P/T limits.

One improvement is the application of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1." ASME B&PV Code Case N-640 allows the use of K_{Ic} rather than K_{Ia}, to determine $T\text{-}RT_{NDT}$

A detailed description of the methodology used and the results obtained are contained in Attachment 4 to this letter.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the River Bend Operating License be amended with a change to Technical Specification 3.4.11, "RCS Pressure and

Temperature (P/T) Limits" to replace figure 3.4.11-1 with a revised figure containing recalculated curves based on new methodology.

Entergy Operations, Inc. has evaluated the proposed changes to the Technical Specifications against the above criteria of 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration. The information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for the proposed changes is provided below.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. **Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed changes to the River Bend reactor coolant system (RCS) pressure/temperature (P/T) limits do not modify the boundary, operating pressure, materials or seismic loading of the reactor coolant system. The proposed changes do adjust the P/T limits for radiation effects to ensure that the RPV fracture toughness is consistent with analysis assumptions and NRC regulations. An evaluation has been performed justifying the use of the methodology contained in Code Case N-640 to determine the P-T curve. The proposed P/T limits were determined using this methodology. Thus, the proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. 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.

   Therefore, this change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed changes to the reactor pressure vessel pressure-temperature limits do not affect the assumed accident performance of any structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

   Therefore, this change does <u>not</u> create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

The methodology for determining the RCS P/T limits ensures that the limits provide a margin of safety to the conditions at which brittle fracture may occur. The methodology is based on requirements set forth in Appendix G and Appendix H of 10CFR50, with reference to the requirements and guidance of ASME Section XI, and on guidance provided in Regulatory Guide 1.99, Revision 2. The revised P/T limits are also based on this methodology except as modified by application of the noted Code Case. Although the Code Case constitutes relaxation from the current requirements of 10CFR50 Appendix G, the alternatives allowed by the Code are based on industry experience gained since the inception of the 10CFR50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Case maintains a margin of safety that is consistent with the intent of 10CFR50 Appendix G, i.e., with regard to the margin originally contemplated by 10CFR50 Appendix G for determination of RPV/RCS P/T limits.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

Pursuant to 10CFR51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR 51.22(c)(9) of the regulations. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change.

   There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant.

Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.

ATTACHMENT 2

TO

LETTER NO. RBF1-01-0010

PROPOSED TECHNICAL SPECIFICATION MARK-UPS

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

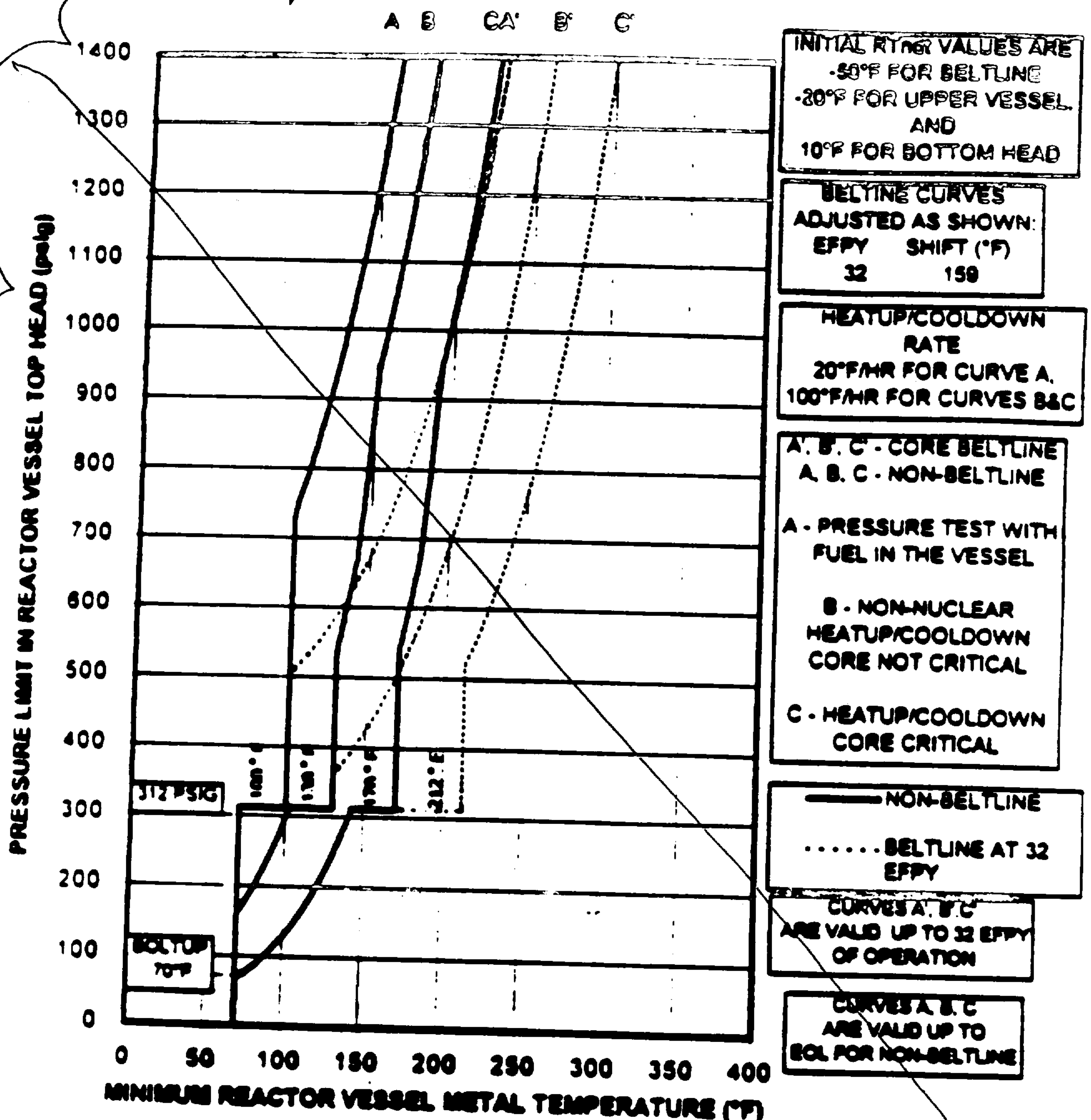

Figure 3 4 11-1 (page 1 of 1)
Minimum Temperature Required vs RCS Pressure

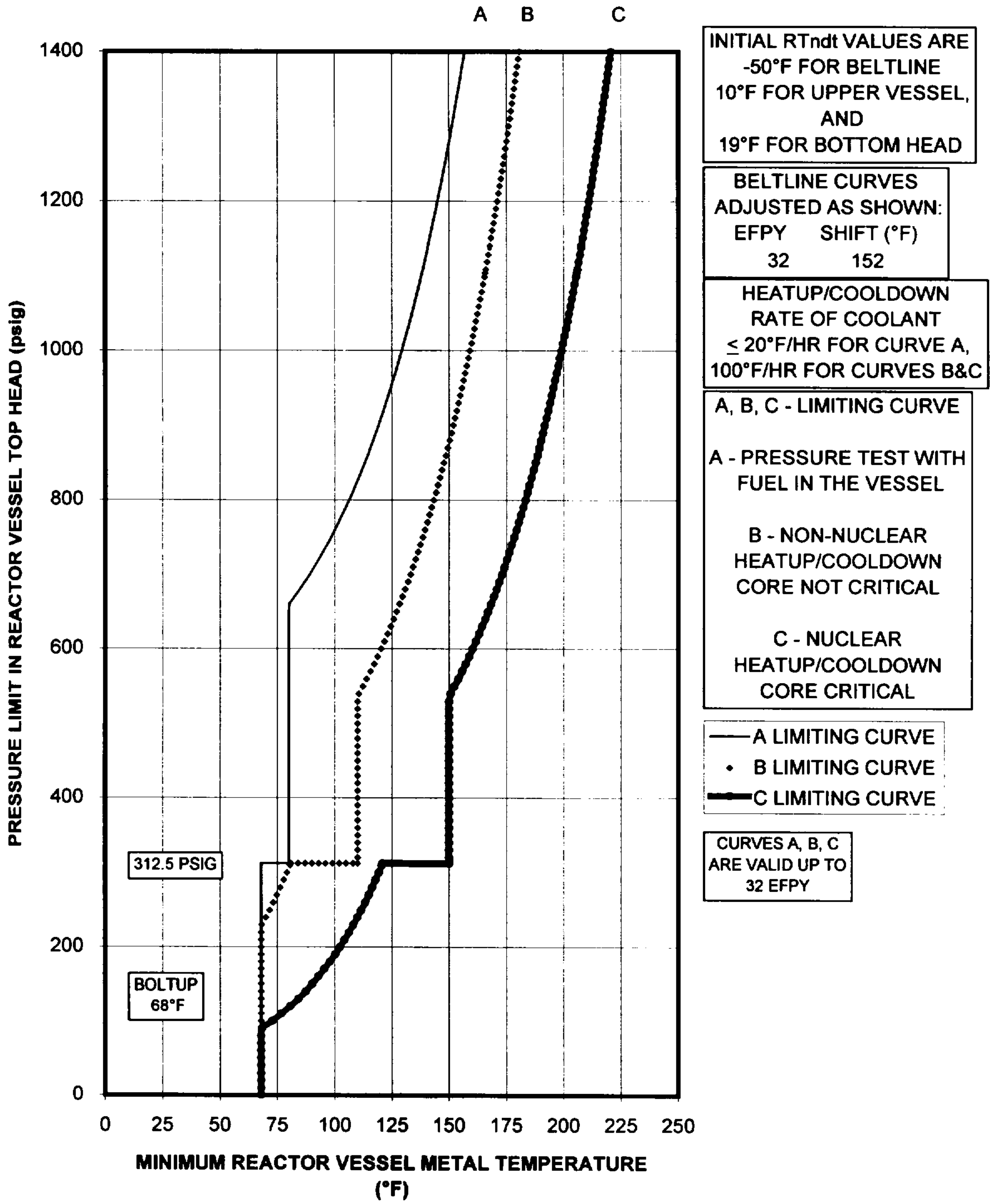

Figure 3.4.11-1 (Page 1 of 1)
Minimum Temperature Required vs. RCS Pressure

ATTACHMENT 3

TO

LETTER NO. RBF1-01-0010

TECHNICAL BASIS FOR REVISED

P-T LIMIT CURVE METHODOLOGY

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

Technical Basis for Revised P-T Limit Curve Methodology

Abstract

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature (P-T) limits, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate safety margins on the fracture toughness.

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 reason for using the conservative K_{Ia} curve was the limited database available at the time WRC-175 (the precursor to Appendix G) was developed. Since then, there has been a substantial increase in the available toughness data so that the excessive conservatism is no longer necessary. 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. The primary impact on the BWR will be a reduction in the pressure test temperature.

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI P-T limit curve methodology and confirms that the safety margin, which exists with the revised methodology, is still large, whether considered deterministically or from the standpoint of risk. The technical basis for the revised methodology is provided in Reference 1.

Introduction

The startup and shutdown process, as well as pressure testing, for an operating nuclear plant is controlled by P-T 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, 25% thickness and 6 times as long as depth
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, such as weld residual stress, were not considered in the original methodology. However, the validation exercises for the proposed methodology did consider the effect of these stresses on safety margins (Reference 5). These studies showed that the safety margins utilized in the computation of applied stress intensity factors ensured adequate safety against non-ductile failure.

There are two lower bound fracture toughness curves available in Section XI: (1) K_{Ia}, which is a lower bound on all static, dynamic and arrest fracture toughness; and (2) 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 P-T limits should be changed from K_{Ia} to K_{Ic}.

Use of K_{Ic} is More Technically Correct

The heatup and cooldown process in a BWR is a very slow one, with the fastest rate allowed being 100°F 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 and B) correspond to quasi-static loadings, with regard to fracture toughness. Therefore, use of the static fracture toughness K_{Ic} lower bound curve 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." Where indications have been found, all have been qualified as buried, or embedded. Such indications are generally benign since they do not communicate with the water environment and extension under operating conditions is negligible.

There are a number of reasons why no surface flaws have been found, 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} (Reference 2) compared to Figure 2, which is the original database (Reference 3). 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.

As can be seen from Figure 1, the vast majority of the data fall well above the K_{Ic} curve. An example set of carefully screened data in the extreme range of lower temperatures is shown in Figure 3 (Reference 4). The data points that fall below the K_{IC} curve are at $T\text{-}RT_{NDT}$ values below the normal operating range for BWR vessels[1] It should also be noted that for irradiated materials, when the RT_{NDT} is adjusted in accordance with Regulatory Guide 1.99, Revision 2, all the K_{IC} data is above the lower bound curve. (Reference 5)

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 (Reference 6) has shown some evidence of early pop-ins for some simulated

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[1] For $T\text{-}RT_{NDT}$ = -150°F and minimum operating temperature of 70°F, the resultant ART of 220°F would be above the requirement for in-situ annealing.

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 appropriate to remove the conservatism associated with this postulated condition and use the ASME Code lower bound K_{Ic} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

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

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, and (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 while still maintaining the required fracture margin.

Reactor Vessel Fracture Margins

It has long been known that the P-T limit curve methodology is very conservative (References 5 and 7). Changing the reference toughness to K_{Ic} will still maintain a very high margin, as illustrated in Figure 4, for a BWR hydrotest. This figure shows three reference cases: (a) Case 1, which is a best estimate curve with no safety factor; (b) Case 2, which is Case 1 plus weld residual stress; and (c) Case 3, which is Case 2 plus cladding induced stresses. For comparison, P-T curves based both the K_{Ia} and K_{IC} equations are shown, which incorporate the standard ASME Section XI safety factors. Even with the proposed change from K_{Ia} to K_{IC}, adequate safety margins are demonstrated by Figure 4. Although there is an approximately 22% reduction in margin (see Table 5 of Reference 1), Figure 4 does demonstrate that sufficient margin still remains.

The impact of the new K_{Ic} based P-T curve on BWR plant operation is negligible. Since the temperature in a BWR follows the saturation curve during all operating conditions including startup and shutdown, the vessel temperature is well in excess of the minimum temperature required by the P-T curve based on K_{Ia} or K_{Ic}. Thus, changing to the P-T curve based on K_{Ic} has no impact on the heatup/cooldown or any other operating conditions.

The only condition that is affected by the new K_{Ic} curve methodology is the pressure test since the core is not critical under the pressure test conditions. Furthermore, analytical studies with reasonable flaw size assumptions show that the typical ASME code margins are still maintained under the new K_{Ic} based P-T curve methodology. Finally the slightly lower temperatures offer several functional advantages:

1. Primarily containment isolation is not required.
2. Lower temperature offers a safer environment for inspection personnel and provides greater reliability of leak detection.

<u>Summary and Conclusions</u>

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI P-T limit curve methodology. The safety margin that exists with the revised methodology is still very large. For the BWR, this change only affects the pressure test curve, when the core is not critical, and therefore there is no measurable effect on plant safety. On the other hand, using the K_{Ic} based P-T curve methodology improves plant safety by eliminating the need for primary containment isolation, which offers a safer temperature environment for inspection personnel, as well as an enhanced ability to detect leaks.

References

1. Bamford, W.H. et al, "Technical Basis for Revised P-T Limit Curve Mehtodology," PVP Volume 407, 2000.

2. 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.

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

4. 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.

5. 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.

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

7. 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.

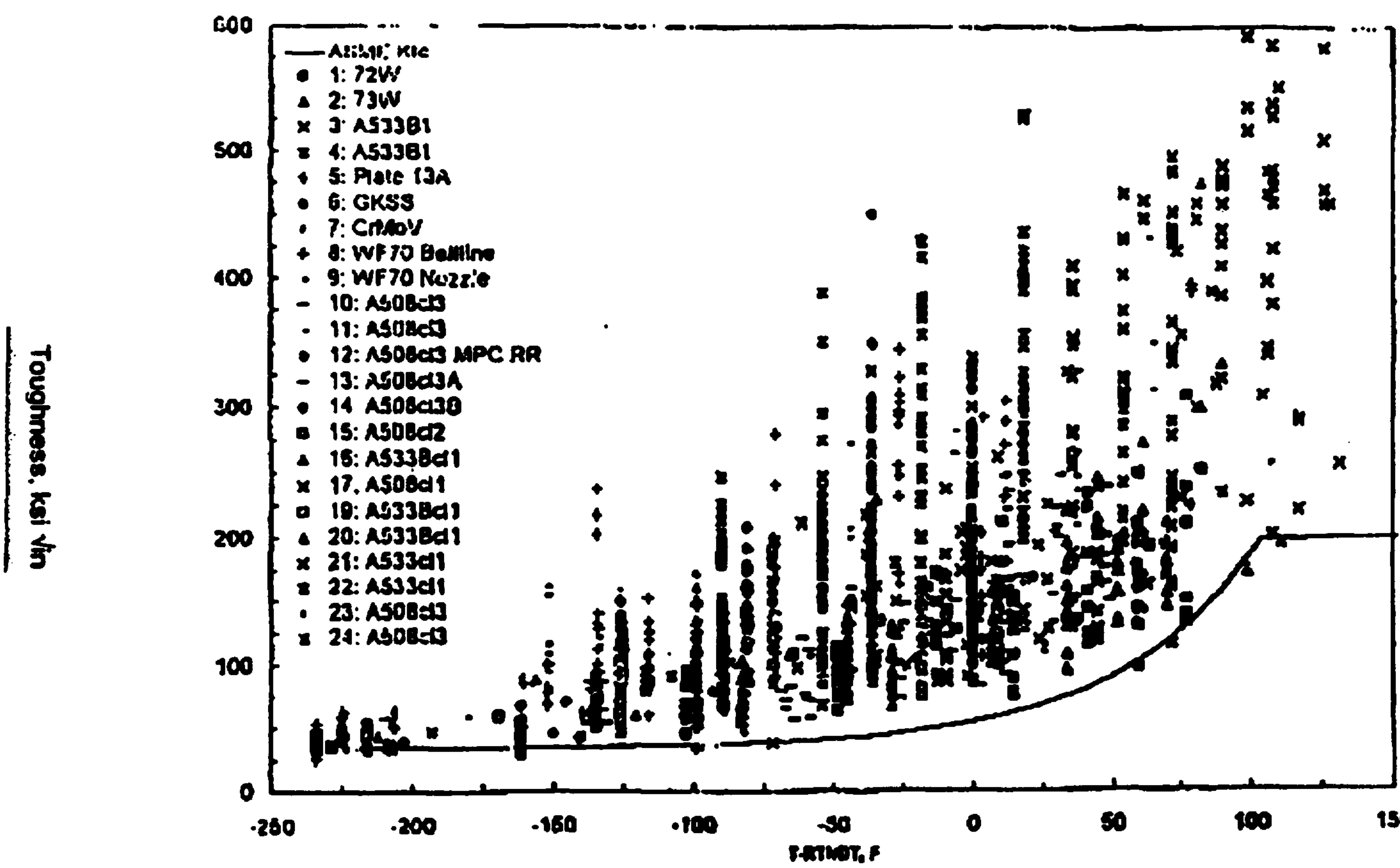

Figure 1: Static Fracture Toughness Data (K_{Jc}) Now Available, Compared to K_{Ic} [2]

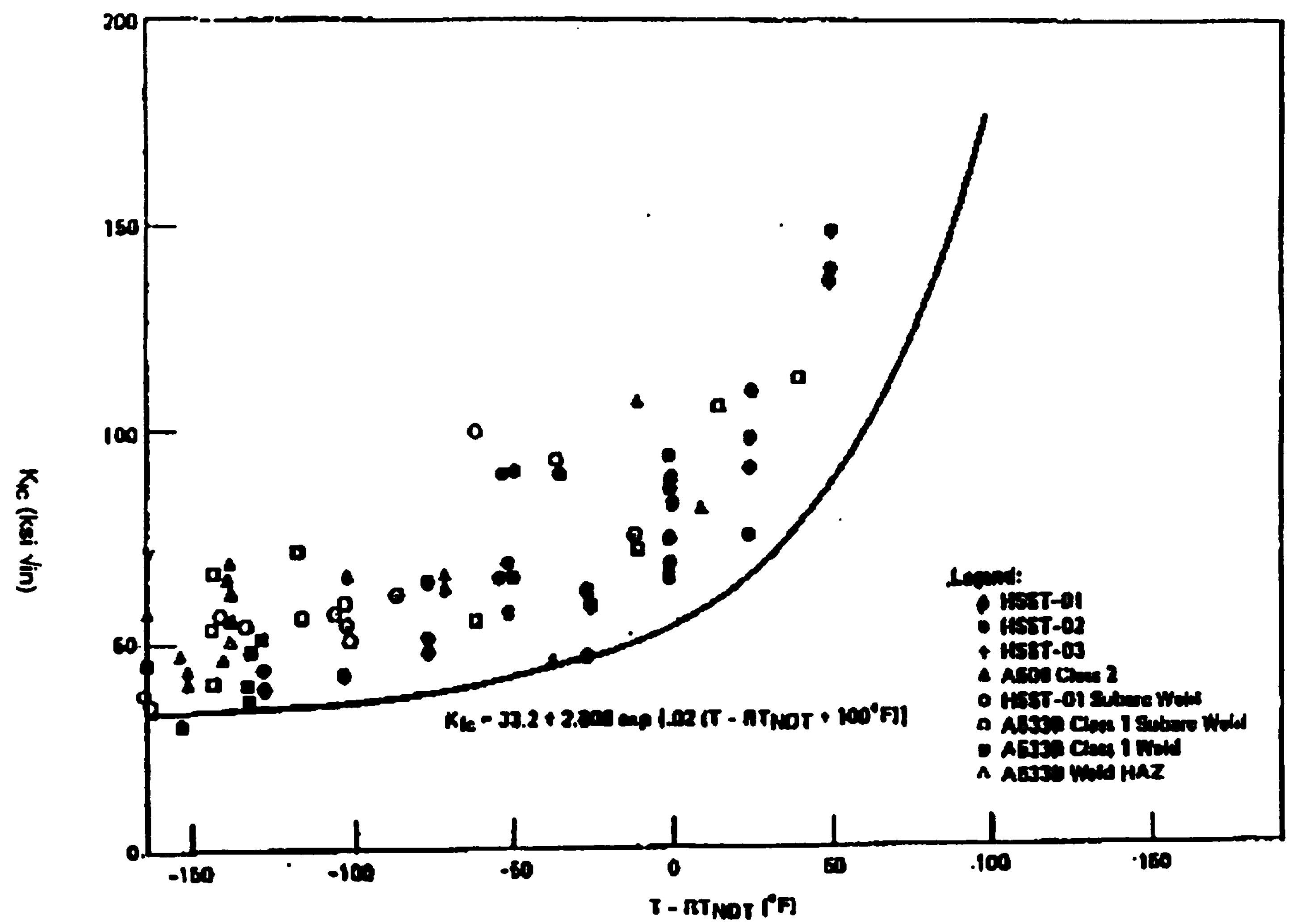

Figure 2: Original K_{IC} Reference Toughness Curve, with Supporting Data [3]

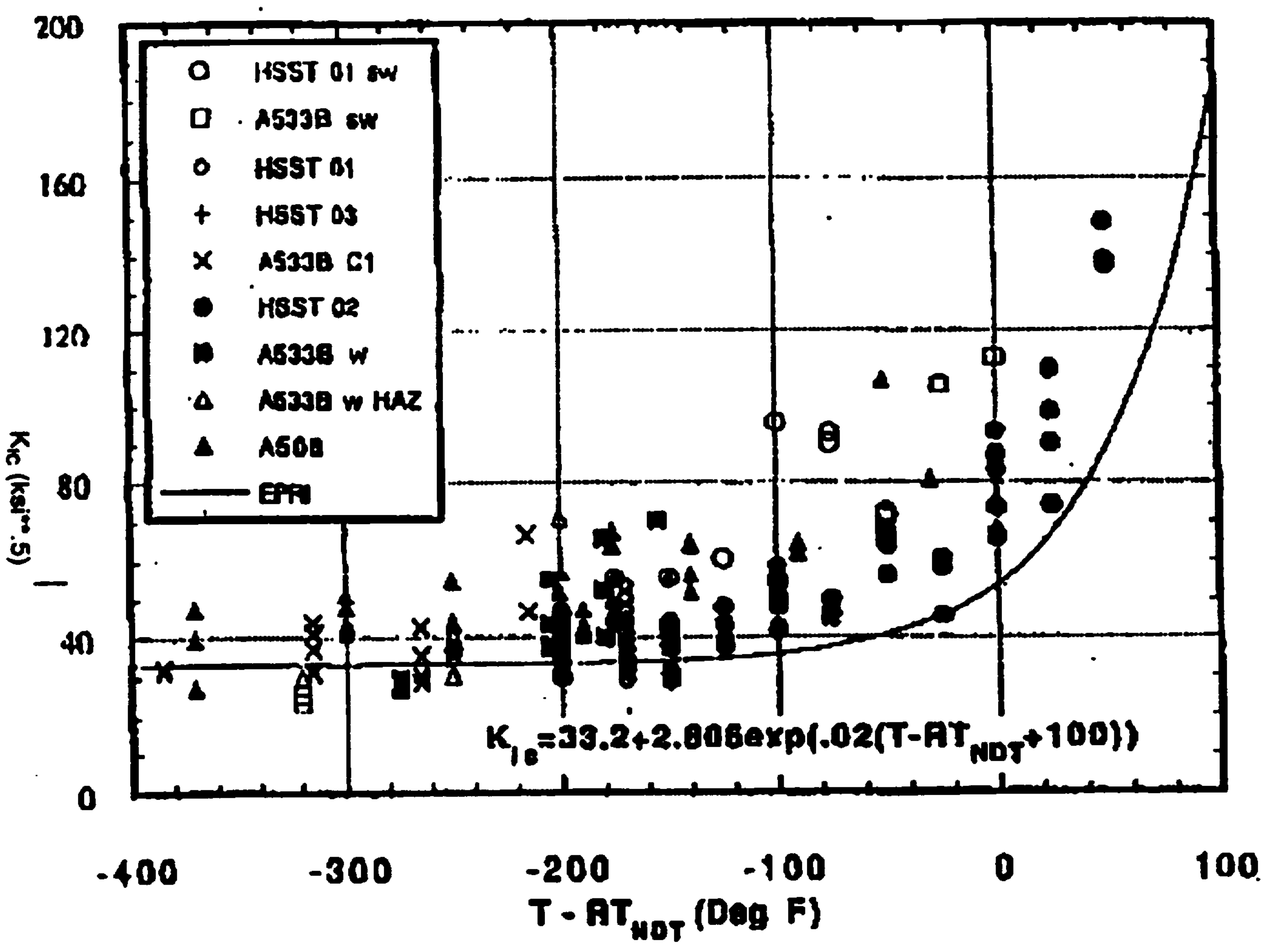

Figure 3: K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [4]

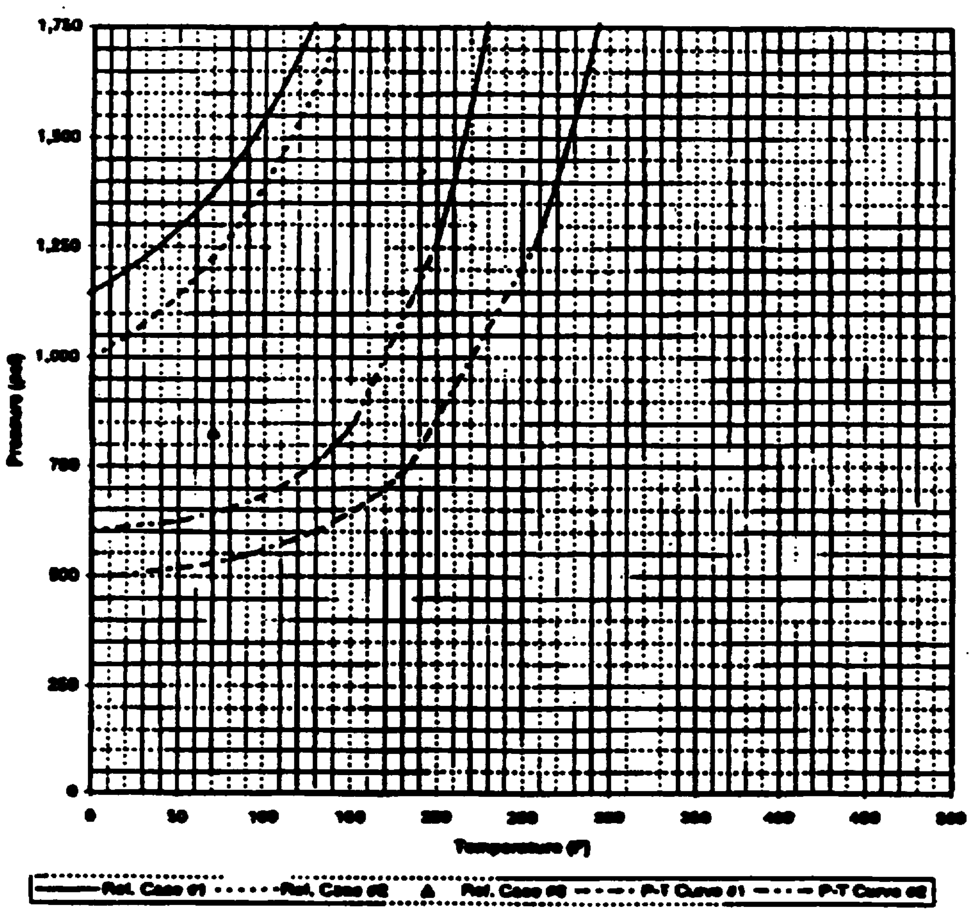

Figure 4: P-T Limit Curves Illustrating Deterministic Safety Factors for a BWR Reactor Vessel [1]