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Pressure-Temperature Limits for 33 EFPY for Oconee Nuclear Station Units 1, 2, and 3

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CONTENTS

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION

This report provides the Oconee Nuclear Station Units 1, 2, and 3 (ONS-1, ONS-2 and ONS-3) reactor coolant Technical Specification Basis pressure-temperature operating limits for thirty-three effective full-power years (EFPY) of reactor operation. These pressure-temperature limits are generated for normal operation heatup, normal operation cooldown, inservice leak and hydrostatic test (ISLH) conditions, and reactor core operations.

2. METHODOLOGY

The pressure-temperature limits for the reactor coolant pressure boundary (RCPB) of Oconee Nuclear Station Units 1, 2, and 3 are established in accordance with the requirements of 10 CFR Part 50, Appendix G[1]. The methods and criteria employed to establish operating pressure and temperature limits are described in topical report BAW-10046A[2] and the alternative rules provided by ASME Code Case N-588 for circumferential flaws in welds and Code Case N-626 for K_{Ic} fracture toughness. These limits are provided to prevent non-ductile failure during any normal operating condition, including anticipated operation occurrences and system hydrostatic tests. The loading conditions of interest include the following:

a. Normal operation, including heatup and cooldown.

b. Inservice leak and hydrostatic test (ISLH).

c. Reactor core operation.

The major components of the RCPB have been analyzed in accordance with 10 CFR Part 50, Appendix G[1]. The closure head region, the reactor vessel (RV) outlet nozzle, and the nozzle belt and beltline regions have been identified as the only regions of the reactor vessel (and consequently the RCPB) that require pressure-temperature limits.

The limit curves for Oconee Nuclear Station Units 1, 2, and 3 are based on the predicted value of the adjusted reference temperature (ART) of the limiting reactor vessel materials at the end of 33 EFPY. The ART's are calculated by adding a radiation-induced ΔRT_{NDT} to the initial RT_{NDT} plus a margin term, using Regulatory Guide 1.99 Revision 2[3] to predict the radiation-induced ΔRT_{NDT} values as a function of the material's copper and nickel content and neutron fluence. Table 1 summarizes the predicted reactor vessel inside surface peak fluence value as well as the fluence values at the ¼ t and ¾ t vessel wall locations (t = wall thickness) of the limiting weld materials for each of the three Oconee Nuclear Station Units at the end of 33 EFPY. The ART values at the ¼ t and ¾ t vessel wall locations are also

provided in Table 1. A value of 60 F is assumed for the RT_{NDT} of the closure head region and the outlet nozzle steel forging, in accordance with BAW-10046A[2].

Unadjusted pressure-temperature (P/T) limits for the closure head, outlet nozzle, nozzle belt, and beltline regions are determined for the heatup and cooldown rates listed in Tables 2 and 3 as well as for steady-state conditions. Differential pressure corrections are applied to the unadjusted P/T limits to account for the pressure differential between the analyzed regions of the reactor vessel and the system pressure sensor locations in the reactor coolant system. These corrections are based on the reactor coolant pump constraints presented in Tables 2 and 3. The maximum allowable pressure as a function of fluid temperature is obtained by a point-by-point comparison of the limiting regions, adjusted for sensor location. The maximum allowable pressure is taken to be the lowest of the calculated allowable pressures (under transient and steady-state conditions) for a given time point. The collection of these data points form the bounding Technical Specification Basis P/T limits. Instrument errors for pressure and temperature are not included in to the Technical Specification Basis P/T limits provided in this document.

Criticality P/T limits are also provided for operation with the core critical. Per 10 CFR Part 50, Appendix G[1], these limits are derived by adding 40 F to the Technical Specification Basis P/T limits above the minimum critical limit temperature.

Table 1. Fluences and ART's at 33 EFPY

Plant	Location of Limiting Beltline Weld	Fluence, n/cm^2	ART, °F
ONS-1	Peak Inside Surface Location	0.956 E19	
	¼ t Limiting Weld Location (SA-1229)	0.522 E19	203.1
	¾ t Limiting Weld Location (WF-25)	0.190 E19	188.0
ONS-2	Peak Inside Surface Location	0.925 E19	
	¼ t Limiting Weld Location (WF-25)	0.538 E19	248.4
	¾ t Limiting Weld Location (WF-25)	0.195 E19	189.6
ONS-3	Peak Inside Surface Location	0.913 E19	
	¼ t Limiting Weld Location (WF-67)	0.532 E19	211.7
	¾ t Limiting Weld Location (WF-67)	0.193 E19	164.5

Table 2. Operational Constraints for Plant Heatup

Plant Heatup Rates	
Reactor Coolant Temperature	Heatup Rate
T < 280 °F	≤ 50 °F/hr
T ≥ 280 °F	≤ 100 °F/hr
Reactor Coolant Pump Constraints during Plant Heatup	
Reactor Coolant Temperature	Loop A / B Pump Constraints
T < 250 °F	2 / 0 or 0 / 2
T ≥ 250 °F	2 / 2

Table 3. Operational Constraints for Plant Cooldown

Plant Cooldown Rates	
Reactor Coolant Temperature	Cooldown Rate
T ≥ 280 °F	≤ 50 °F in any 1/2 hour period
150 °F ≤ T < 280 °F	≤ 25 °F in any 1/2 hour period
T < 150 °F	≤ 10 °F in any 1 hour period
Reactor Coolant Pump Constraints during Plant Cooldown	
Reactor Coolant Temperature	Loop A / B Pump Constraints
T < 250 °F	2 / 0 or 0 / 2
T ≥ 250 °F	2 / 2

3. RESULTS

Technical Specification Basis P/T limits are provided in Figures 1 through 9 for ONS-1, ONS-2, and ONS-3 at 33 EFPY for normal heatup, normal cooldown, and inservice leak and hydrostatic test conditions. Criticality limits for reactor core operations also provided with the normal heatup data. These P/T limits have been developed considering the operational constraints identified in Tables 2 and 3.

Maintaining the reactor coolant system pressure below the upper limits of the pressure-temperature limit curves ensures protection against non-ductile failure. Acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the applicable P/T limit curves. These P/T limit curves have been adjusted based on pressure differentials between points of system pressure measurement and the point in the reactor vessel that establishes the controlling, unadjusted pressure limit. System pressure is measured at Oconee Nuclear Station Units 1, 2, and 3 at the decay heat removal system drop line location when that pressure is less than 600 psig, and at the hot leg taps for higher pressures. The P/T limit curves provided in Figures 1 through 9 account for this switch in the source of pressure measurement, although they do not include margins for instrument error. The reactor is not permitted to be critical until the pressure-temperature combinations are to the right of the criticality limit curve.

The numerical values for the Technical Specification Basis P/T curves provided in Figures 1 through 9 are provided in Tables 4 through 6.

Table 4. Normal Operation Heatup P/T and Core Criticality Limits for 33 EFPY

Normal Operation Heatup P/T Limits					
ONS-1		ONS-2		ONS-3	
Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig
60	562	60	588	60	588
100	562	180	588	180	588
125	590	180	782	180	785
180	590	200	964	190	866
180	751	215	1160	200	968
200	859	225	1329	210	1093
225	1070	240	1656	220	1246
250	1387	245	1725	230	1434
265	1695	250	1751	240	1664
275	1957	270	2039	245	1798
287	2250	281	2250	255	2077
				260	2250

Core Criticality Limits					
ONS-1		ONS-2		ONS-3	
Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig
311	0	297	0	287	0
311	1853	297	1859	287	1860
315	1957	310	2039	295	2077
327	2250	321	2250	300	2250

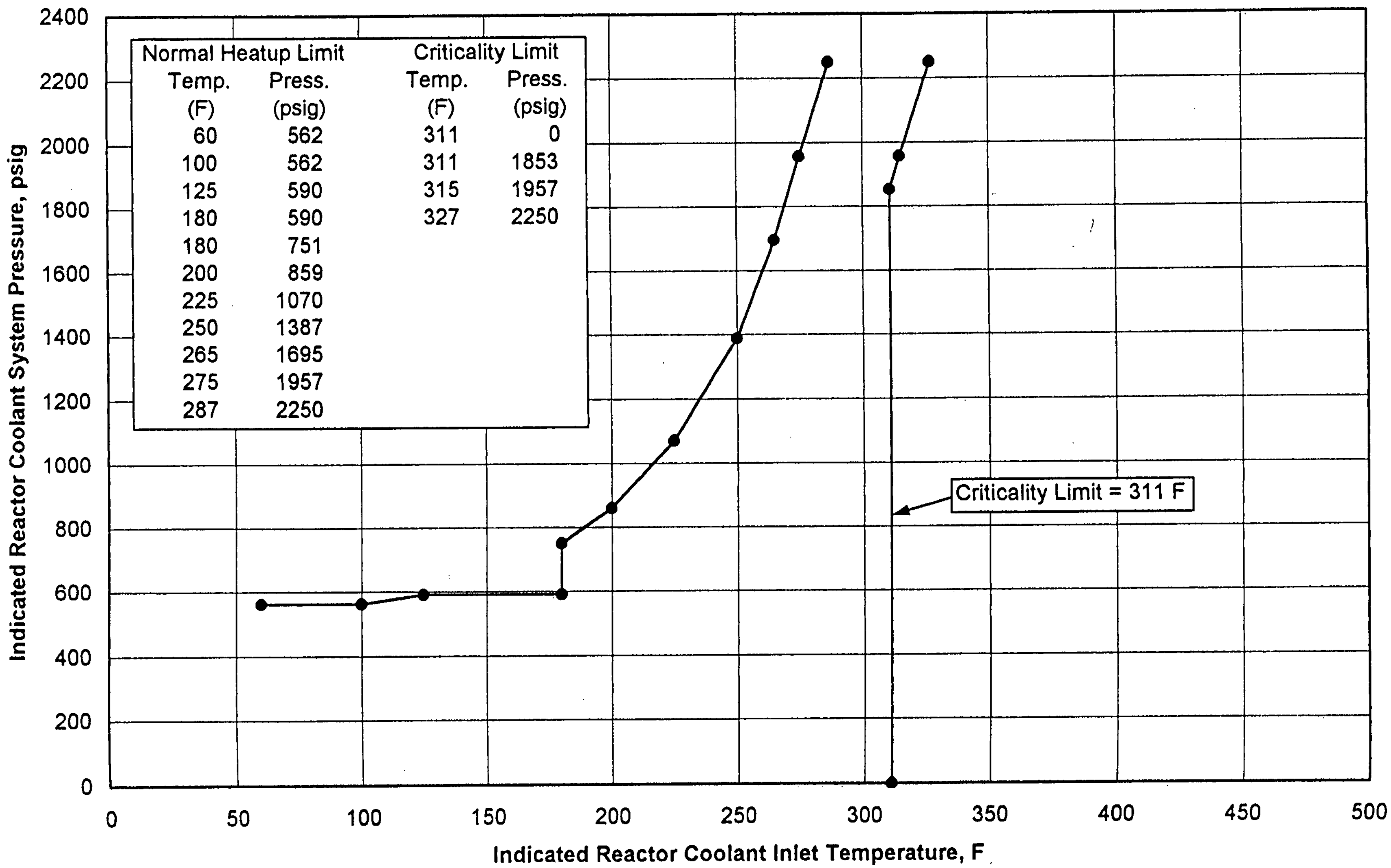
Table 5. Normal Operation Cooldown P/T Limits for 33 EFPY

Normal Operation Cooldown P/T Limits					
ONS-1		ONS-2		ONS-3	
Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig
60	572	60	588	60	588
90	590	180	588	180	588
180	590	180	1315	180	1495
180	786	200	1427	200	1817
200	965	227	1694	212	1988
227	1285	250	1987	227	2250
240	1534	255	2065		
255	1859	264	2250		
268	2250				

Table 6. ISLH Heatup and Cooldown P/T Limits for 33 EFPY

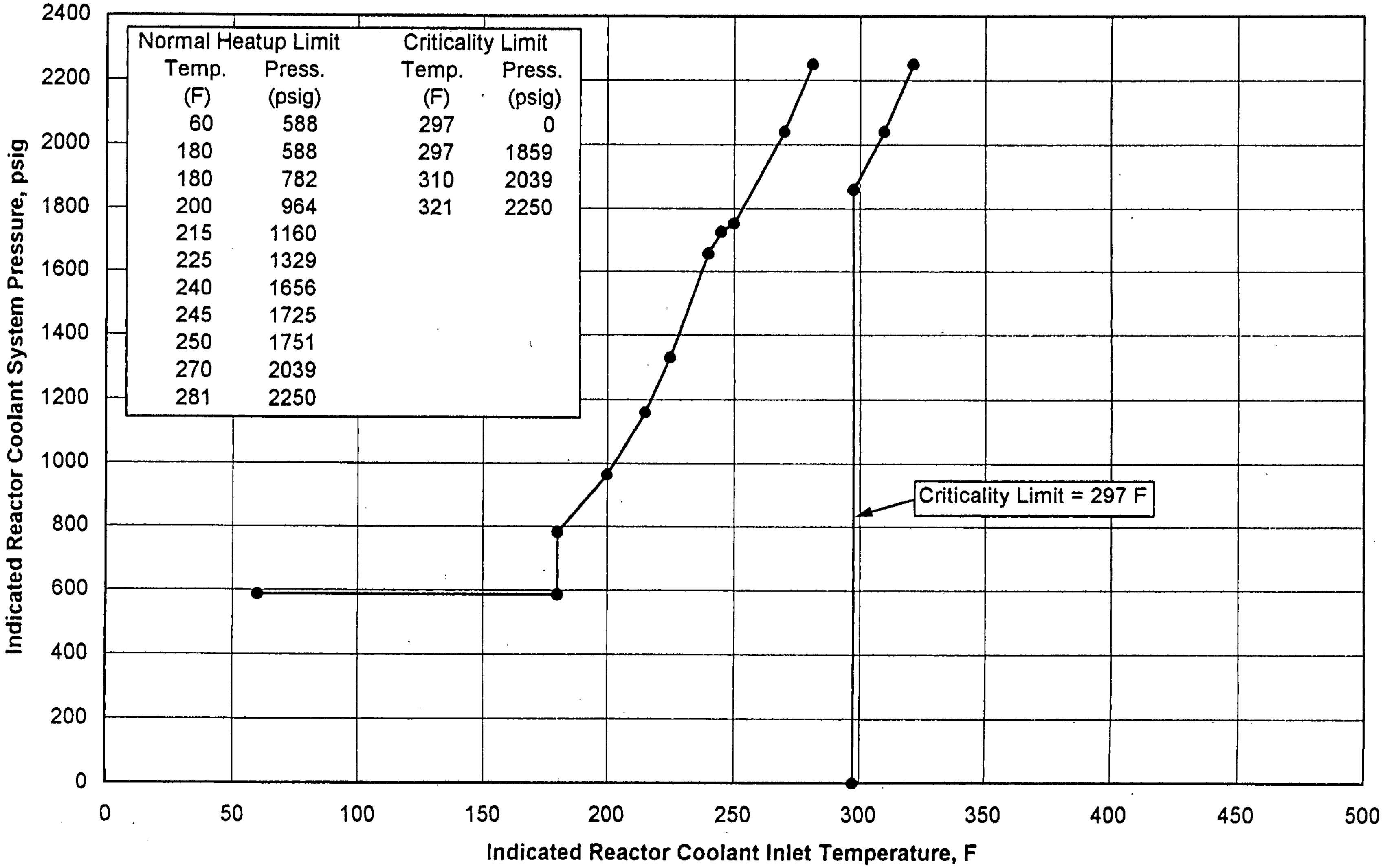
ISLH Heatup and Cooldown P/T Limits					
ONS-1		ONS-2		ONS-3	
Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig	Indicated RCS Inlet Temp., F	Indicated Pressure, psig
60	590	60	588	60	588
150	590	150	588	150	588
150	848	150	875	150	877
170	975	155	896	155	899
185	1059	175	1026	175	1030
200	1172	200	1314	185	1127
220	1385	215	1575	200	1319
235	1611	225	1800	215	1583
250	1887	240	2236	225	1809
271	2500	245	2328	240	2248
		250	2373	247	2500
		257	2500		

Figure 1. Normal Operation Heatup P/T Limits for Oconee Unit 1 at 33 EFPY



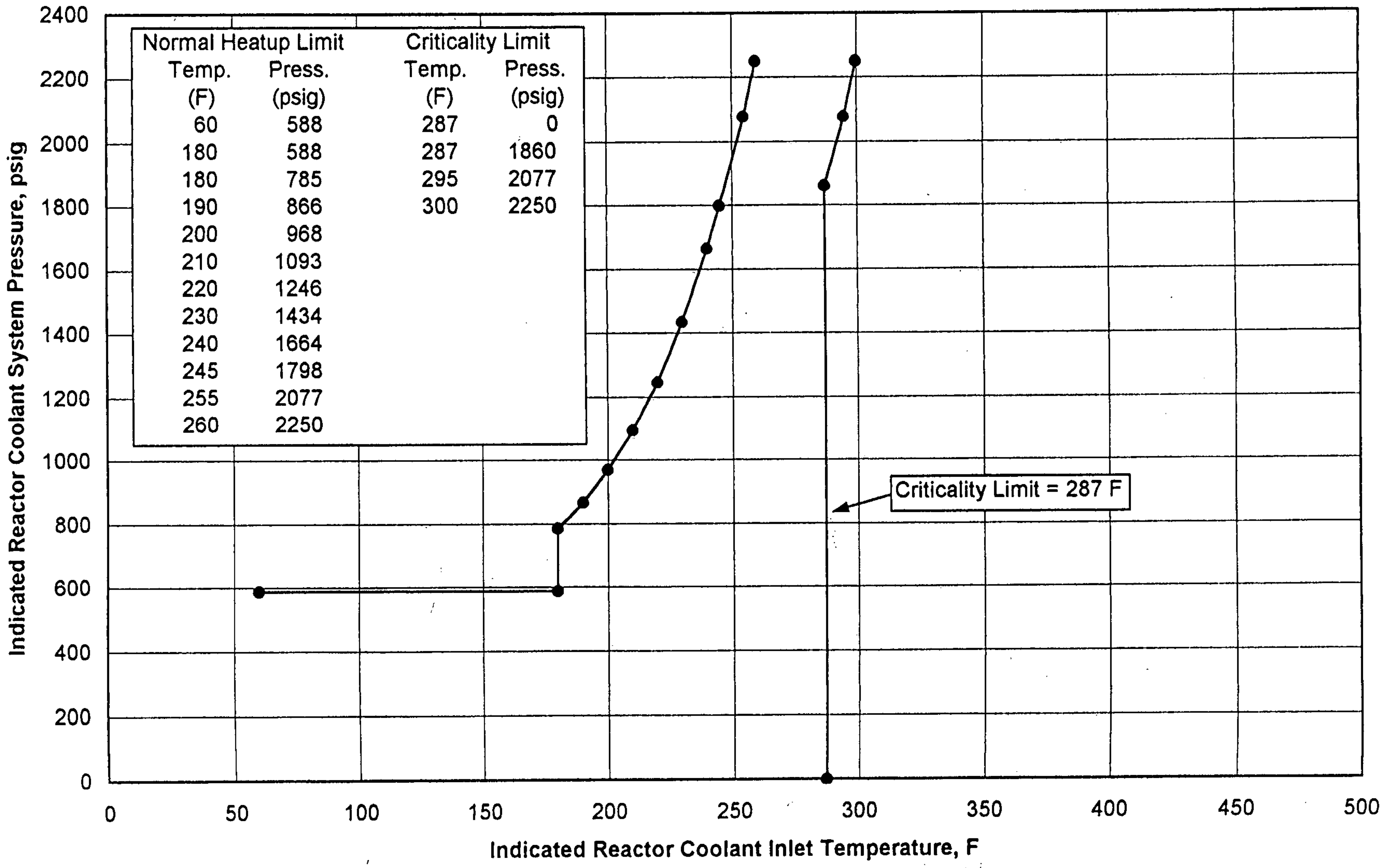
Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	562	311	0
100	562	311	1853
125	590	315	1957
180	590	327	2250
180	751		
200	859		
225	1070		
250	1387		
265	1695		
275	1957		
287	2250		

Figure 2. Normal Operation Heatup P/T Limits for Oconee Unit 2 at 33 EFPY



Normal Heatup Limit		Criticality Limit	
Temp. (F)	Press. (psig)	Temp. (F)	Press. (psig)
60	588	297	0
180	588	297	1859
180	782	310	2039
200	964	321	2250
215	1160		
225	1329		
240	1656		
245	1725		
250	1751		
270	2039		
281	2250		

Figure 3. Normal Operation Heatup P/T Limits for Oconee Unit 3 at 33 EFPY



Normal Heatup Limit Temp. (F)	Normal Heatup Limit Press. (psig)	Criticality Limit Temp. (F)	Criticality Limit Press. (psig)
60	588	287	0
180	588	287	1860
180	785	295	2077
190	866	300	2250
200	968		
210	1093		
220	1246		
230	1434		
240	1664		
245	1798		
255	2077		
260	2250		

Figure 4. Normal Operation Cooldown P/T Limits for Oconee Unit 1 at 33 EFPY

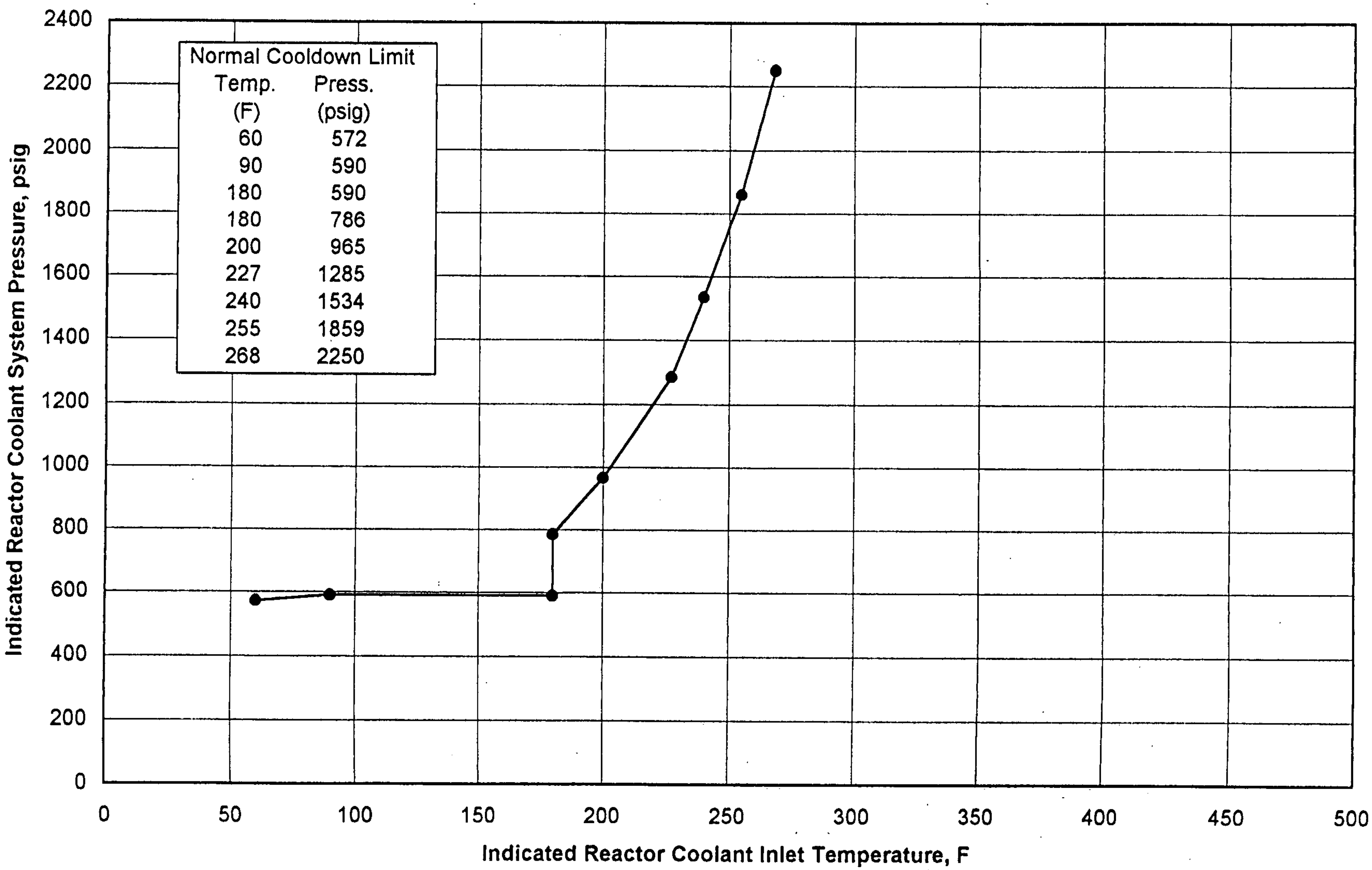


Figure 5. Normal Operation Cooldown P/T Limits for Oconee Unit 2 at 33 EFPY

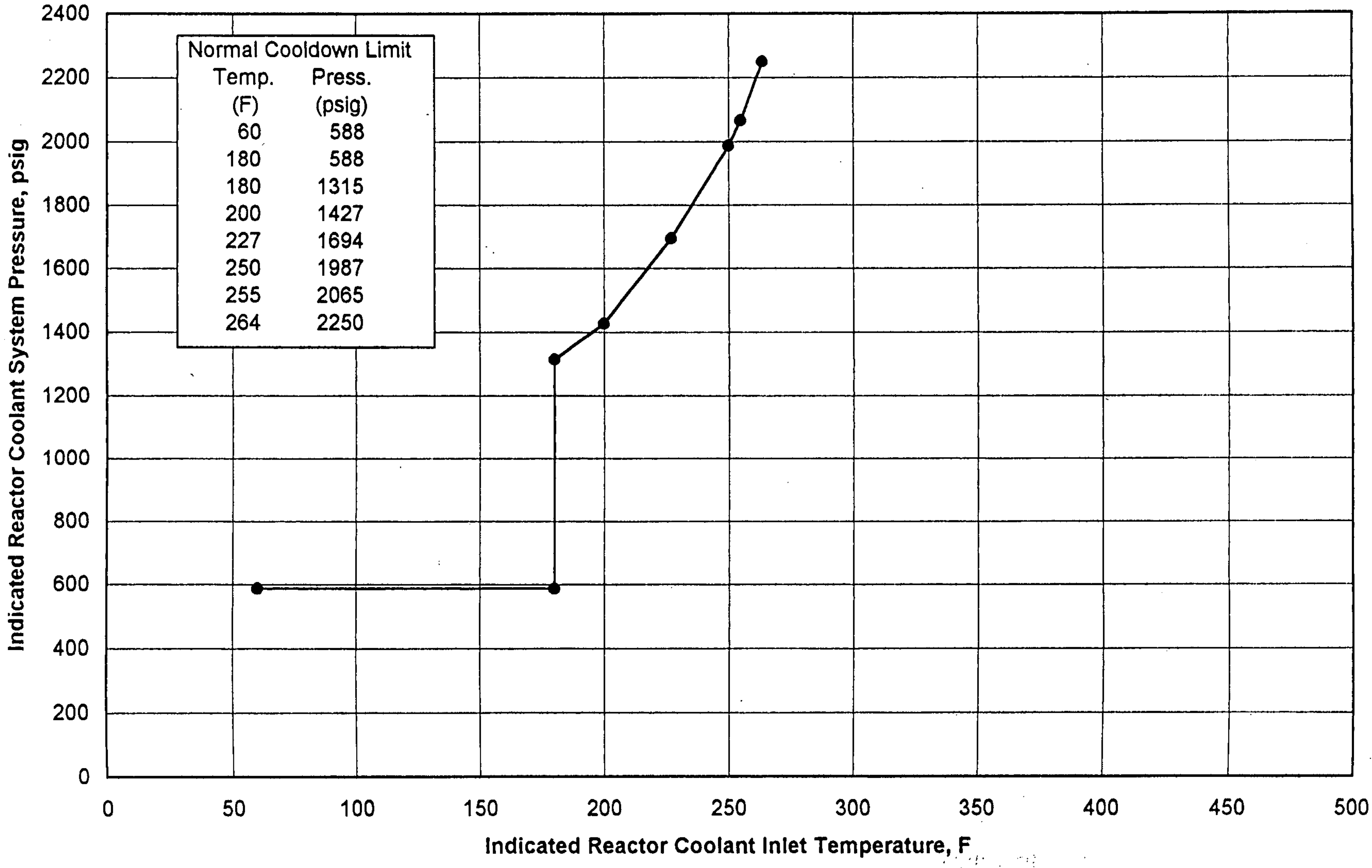


Figure 6. Normal Operation Cooldown P/T Limits for Oconee Unit 3 at 33 EFPY

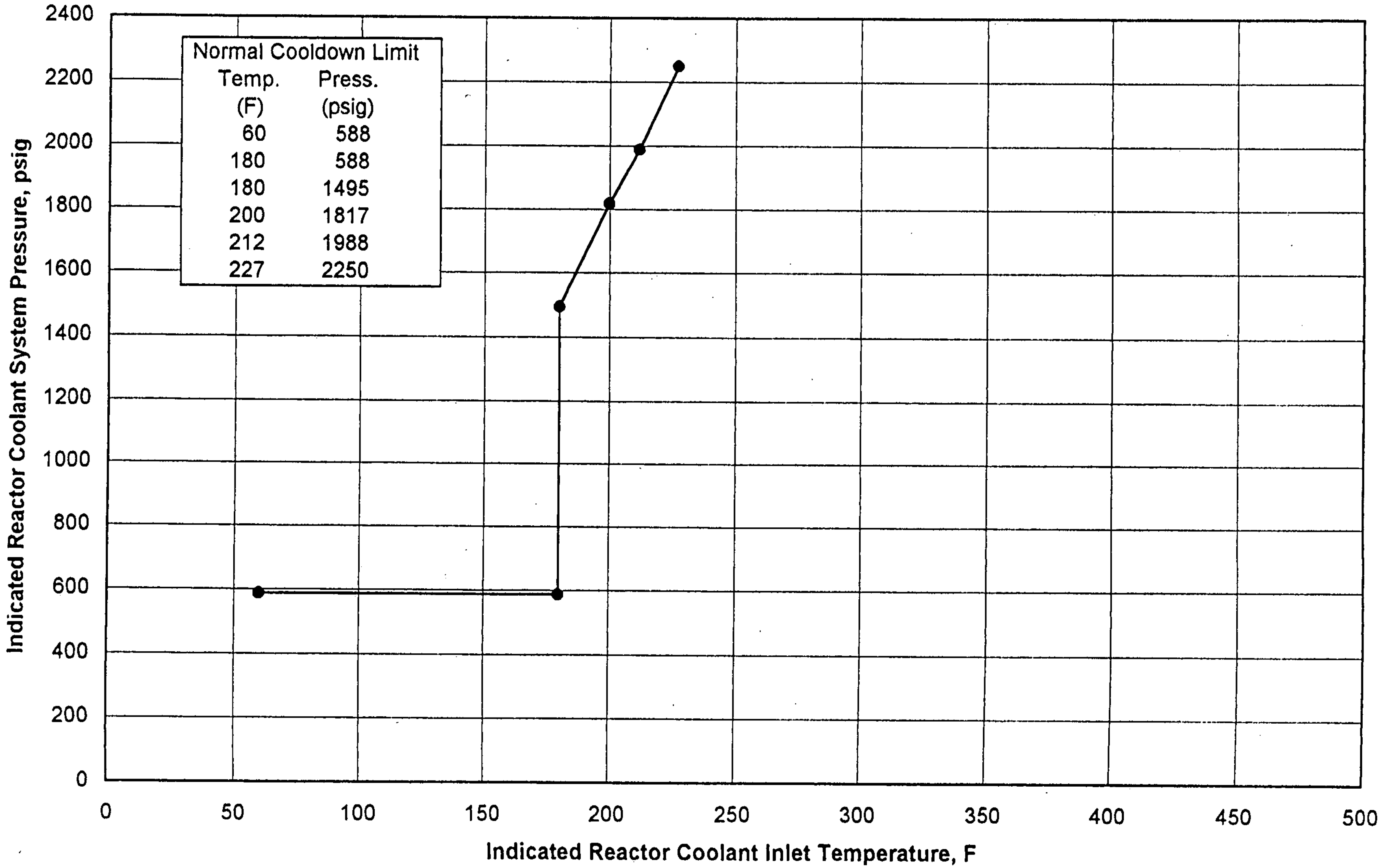


Figure 7. ISLH Heatup and Cooldown P/T Limits for Oconee Unit 1 at 33 EFPY

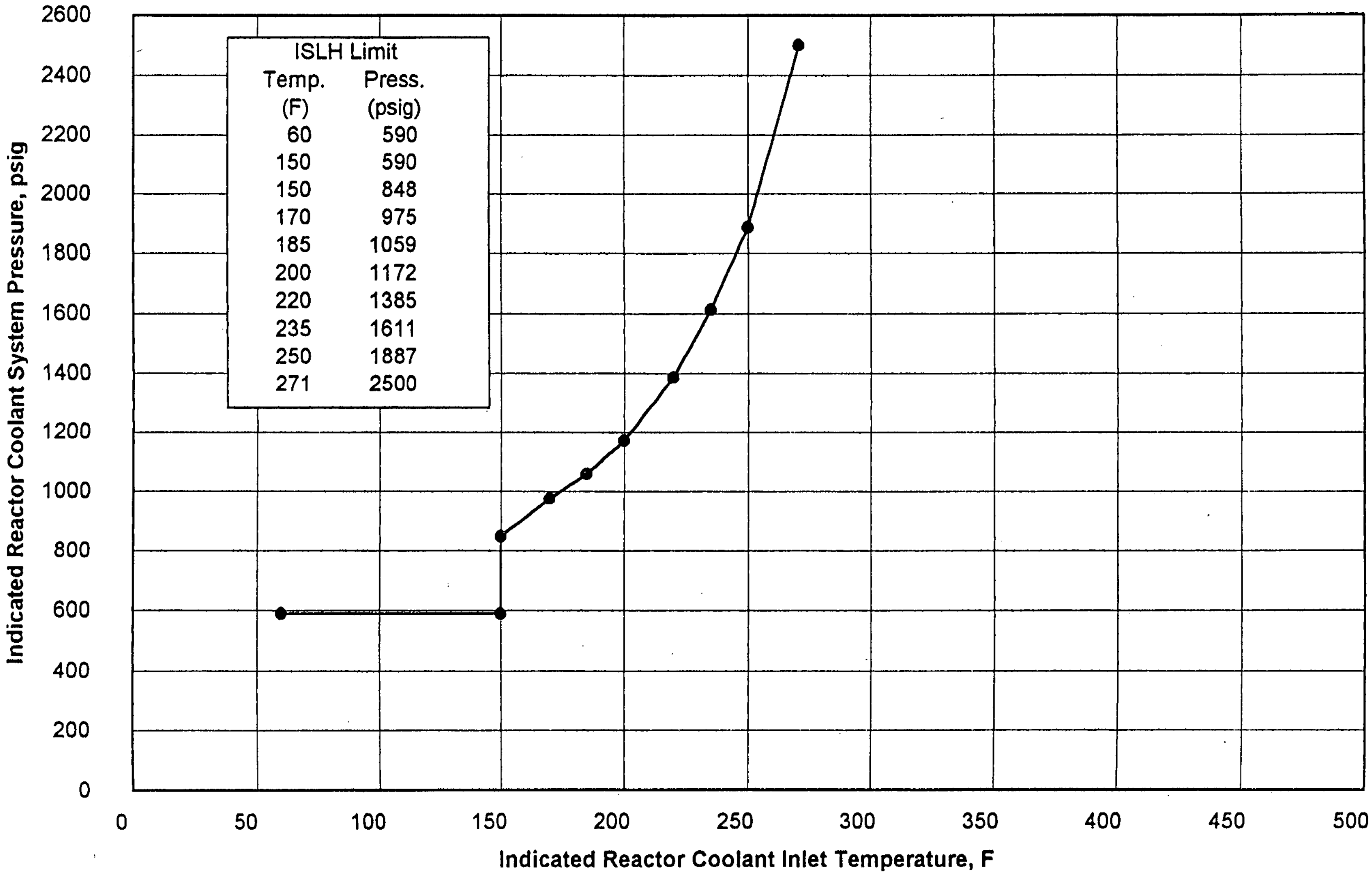


Figure 8. ISLH Heatup and Cooldown P/T Limits for Oconee Unit 2 at 33 EFPY

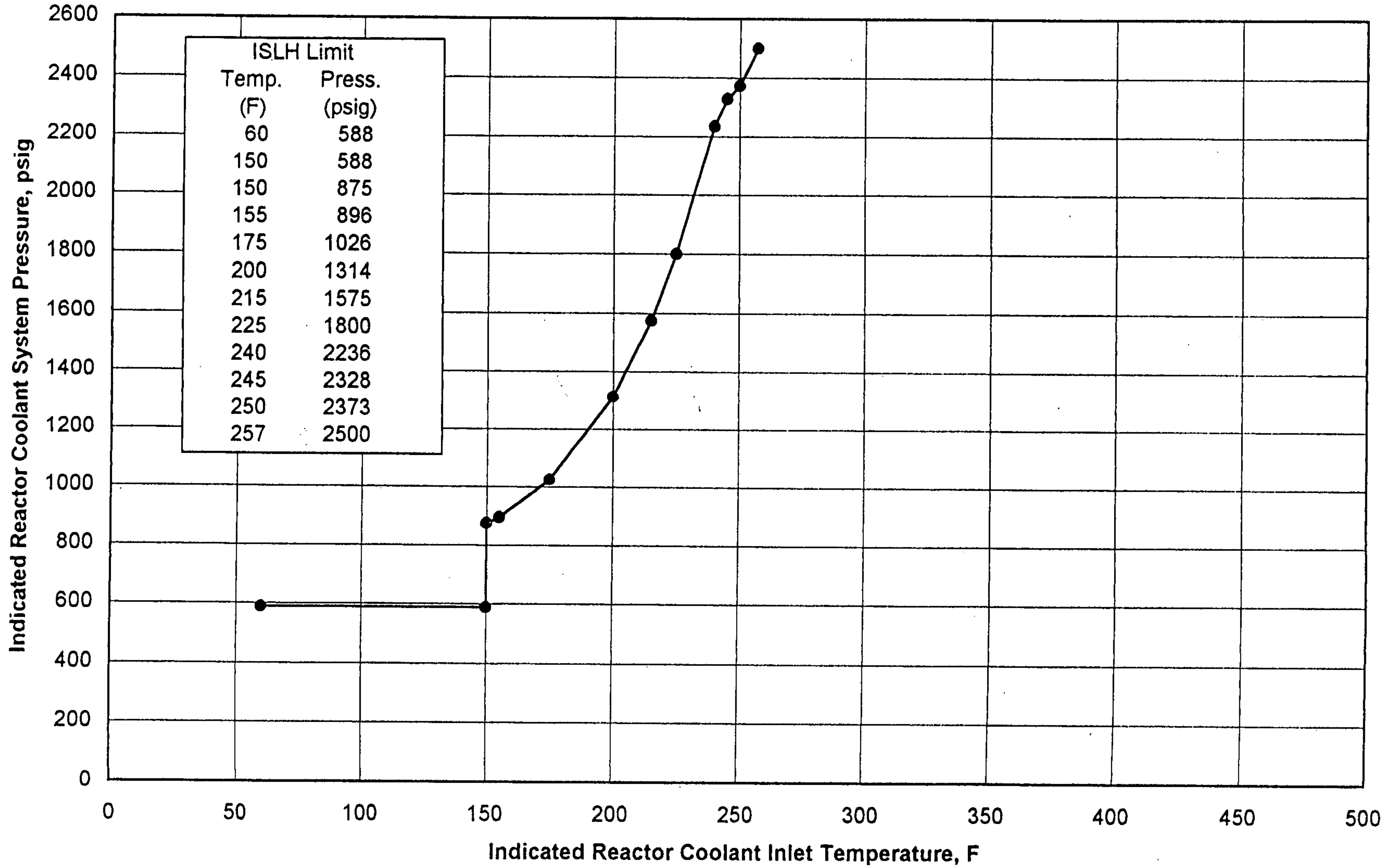
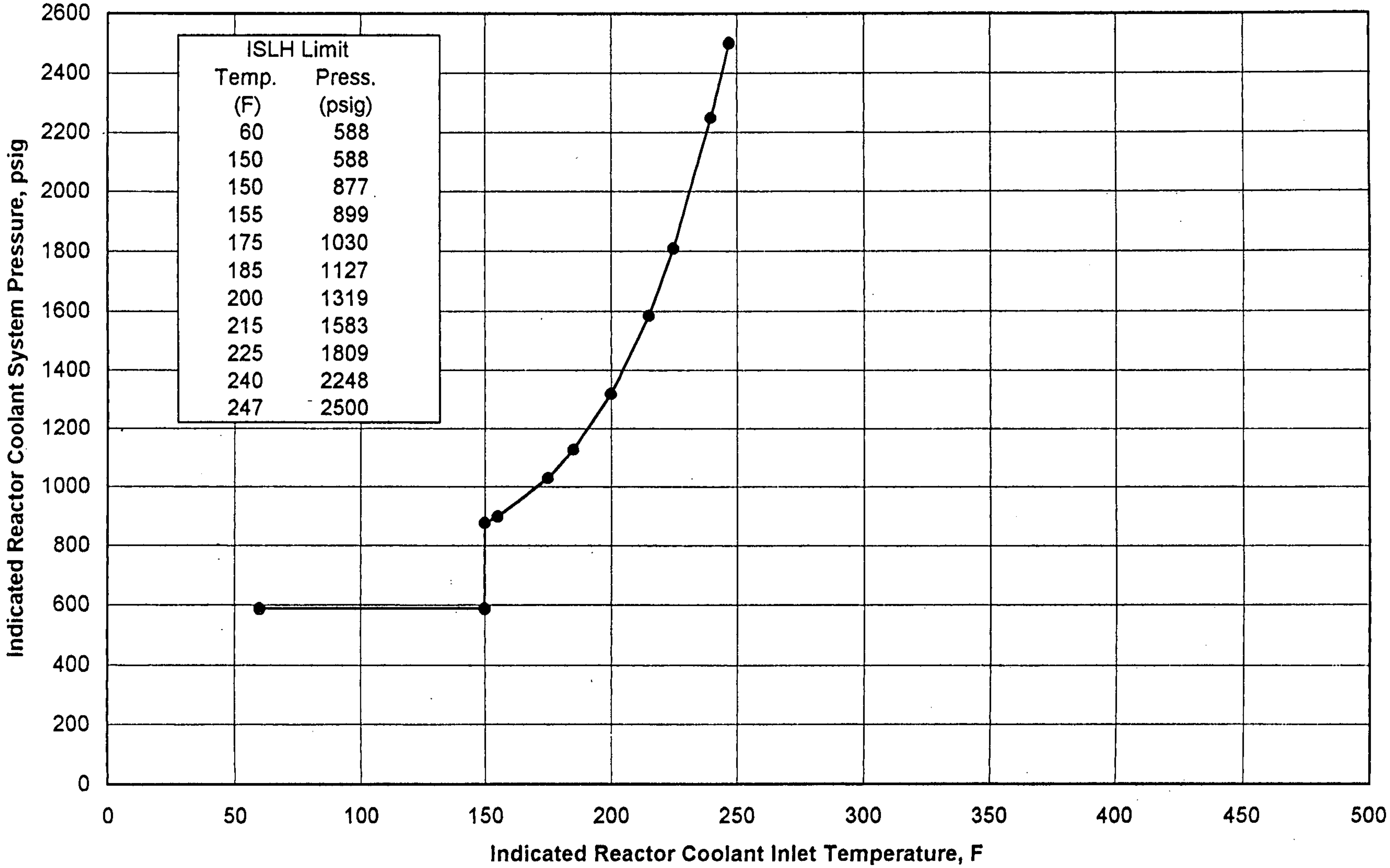


Figure 9. ISLH Heatup and Cooldown P/T Limits for Oconee Unit 3 at 33 EFPY



4. REFERENCES

1. Code of Federal Regulation, Title 10, Part 50, "Fracture Toughness Requirements for Light Water Reactor Pressure Vessels; Final Rule," Appendix G to Part 50 — Fracture Toughness Requirements, Federal Register Vol. 60. No. 243, December 19, 1995.

2. H.W. Behnke, et al, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," BAW-10046A, Rev. 2, BWNT, Lynchburg, Virginia, June 1986.

3. U.S. Nuclear Regulatory Commission, "Radiation Damage to Reactor Vessel Material," Regulatory Guide 1.99, Revision 2, May 1988.

Attachment 3 - Technical Justification

ENCLOSURE 2

Code Case N-626
Alternative Reference Fracture Toughness for Development of P-T Limit Curves for
ASME Section XI, Division I,
September 18, 1998,

Technical Bases for
Revised P-T Limit Curve Methodology
(ASME Code Case N-626), August 6, 1998,

and

ASME XI, Appendix G,
draft changes to incorporate methodology of Code Case N-626,
(ISI-98-04)

Code Case N-626
K_{IC} Curve Methodology

Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1.

Inquiry: May the reference fracture toughness curve K_{IC}, as found in Appendix A of Section XI, be used in lieu of Figure G-2210-1 in Appendix G for development of P-T limit curves?

Reply: It is the opinion of the Committee that the reference fracture toughness K_{IC} of Figure A-4200-1 of Appendix A may be used in lieu of Figure G-2210-1 in Appendix G for the development of P-T limit curves. When this code case is employed, LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the P-T limit curves.

Applicability: Section XI, 1992 edition through 1998 edition.

TECHNICAL BASIS FOR REVISED P-T LIMIT CURVE METHODOLOGY

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Abstract

The startup and shutdown process 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 nine numbers of safety margins; 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 press 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})

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 heat-up and cool-down 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 stress is essentially constant in this case. Both the heat-up and cool-down 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. 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 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.

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 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_{IC} and/or K_{JC}). Therefore, it is time to remove the conservatism associated with this postulated condition and use the ASME Code lower bound K_{IC} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

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

Impact on PWRs:

By opening up the operating window relative to the pump seal requirements, as shown schematically in Figure 3, 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 radio-active release. Finally, challenges to the plant LTOP system and potential problems with reseating the valves would also be reduced.

Impact on BWRs:

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

Reactor Vessel Fracture Likelihood is Very Low

It has long been known that the P-T limit curve methodology is very conservative[3,4]. Changing the reference toughness to K_{IC} will maintain a very high margin, as illustrated in Figure 4, for a pressurized water reactor. This figure shows a series of P-T curves developed for the same plant, but with different assumptions concerning flaw size, safety margin and fracture toughness.

The results shown in Figure 4 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 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. Typical results are shown in Table 1. 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 4. .

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 (cool-down) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs.

The results are shown in Figure 5 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.

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, 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. 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.

4. 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 1 inch flaw for pressure, thermal, residual and cladding loads	1520	3.61
Reference 1 inch flaw for pressure, thermal and residual loads	1845	4.38
Reference 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

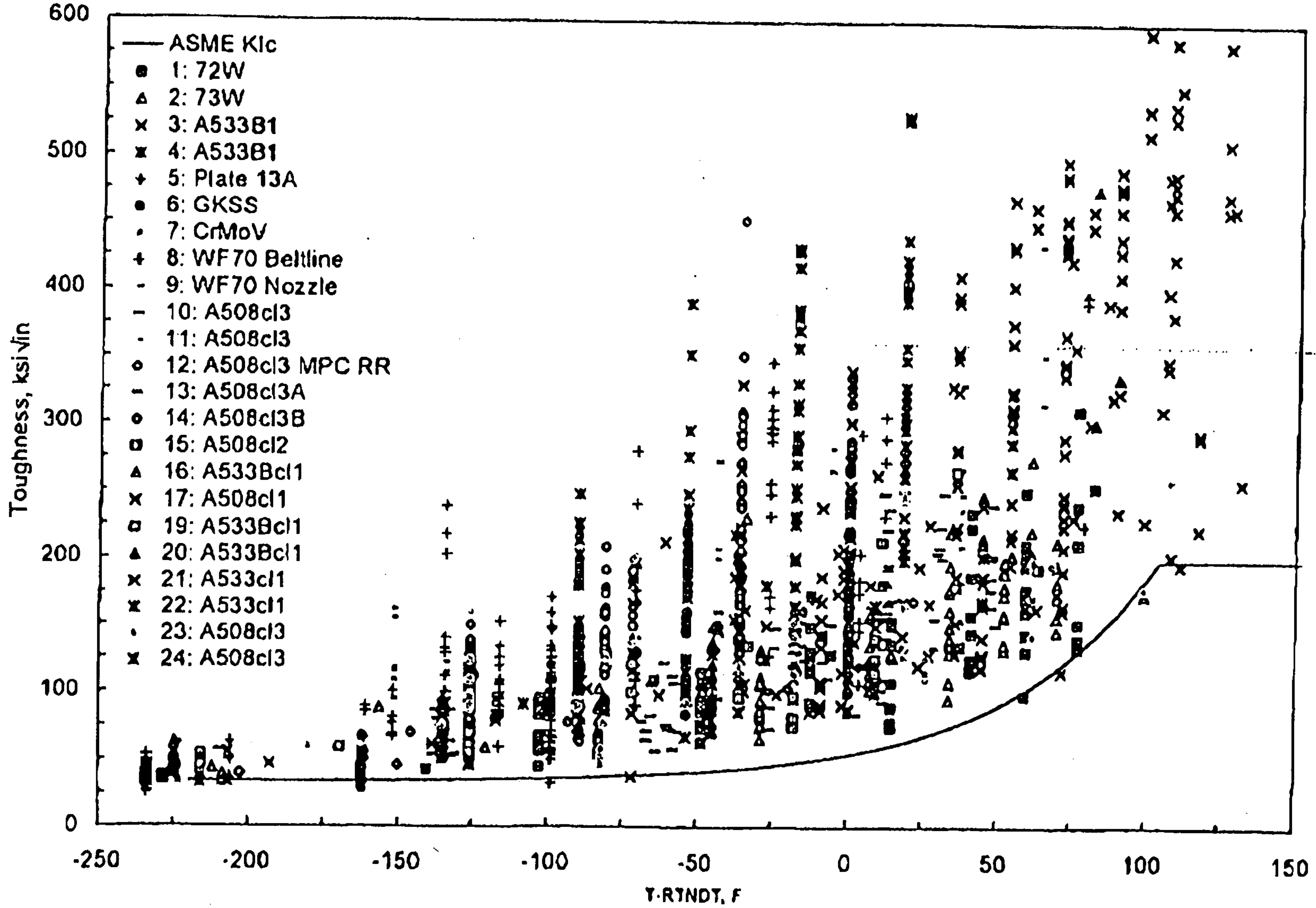

Figure1. Static Fracture Toughness Data (K_{Ic}) Now Available, Compared to K_{Ic} [1]

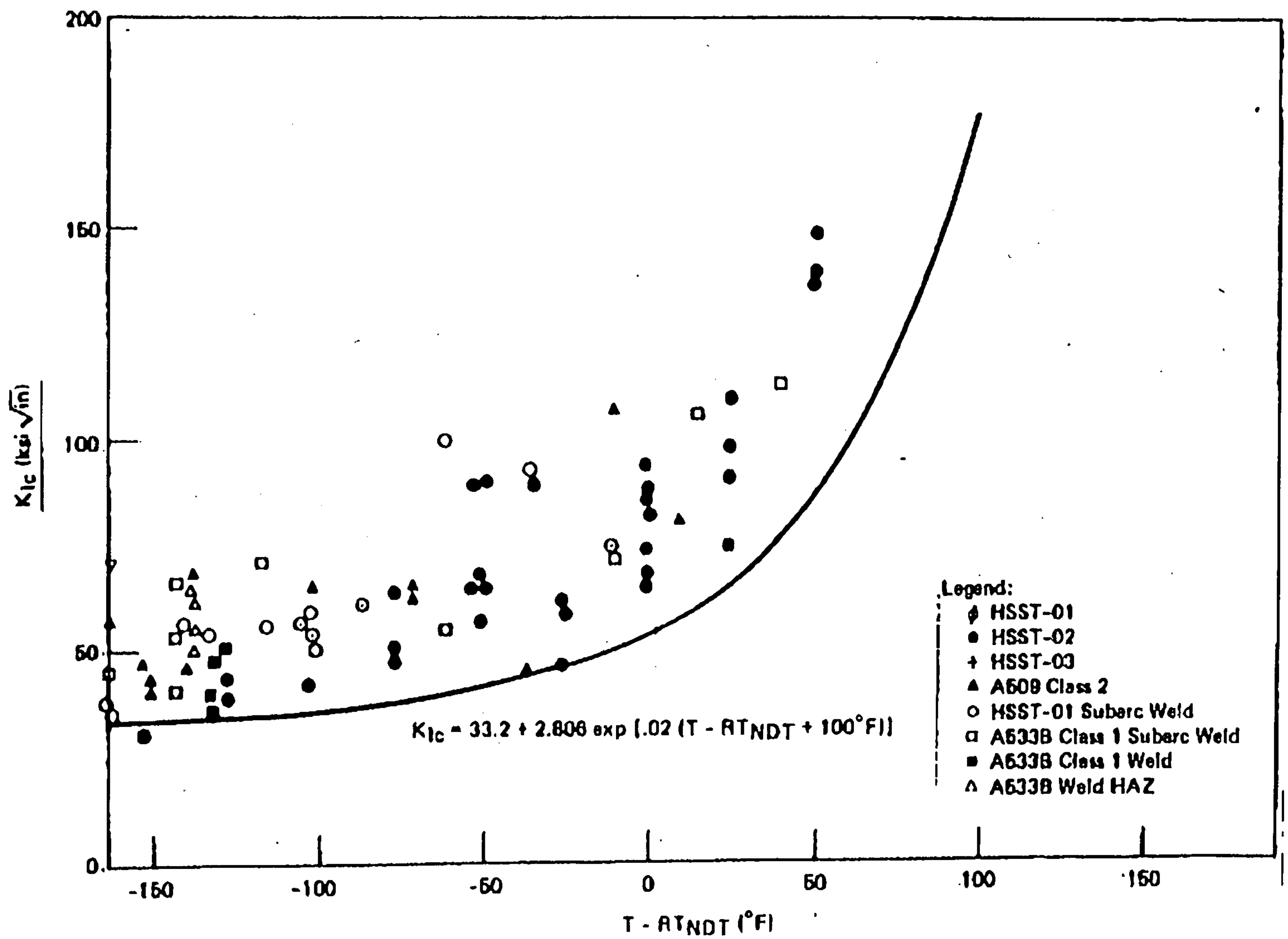

Figure 2. Original K_{Ic} Reference Toughness Curve, with Supporting Data [2]

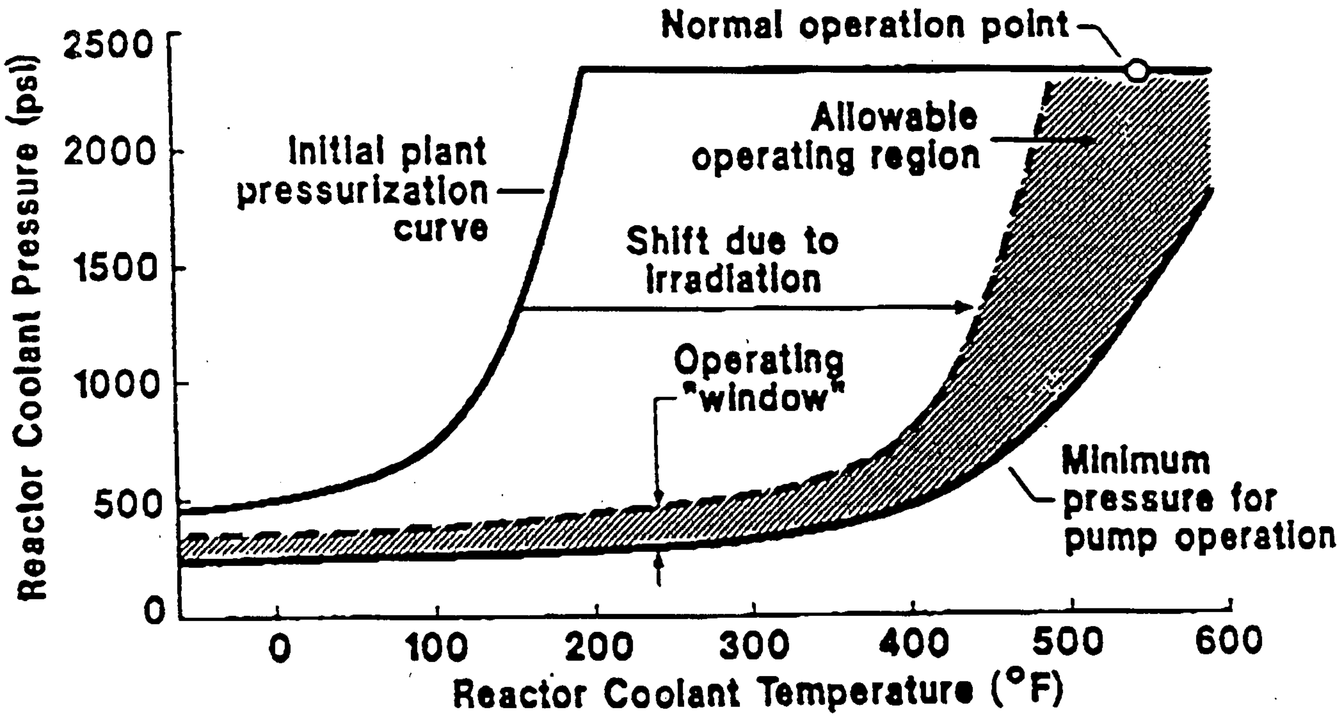

Figure 3. Operating Window From P-T Limit Curves [4]

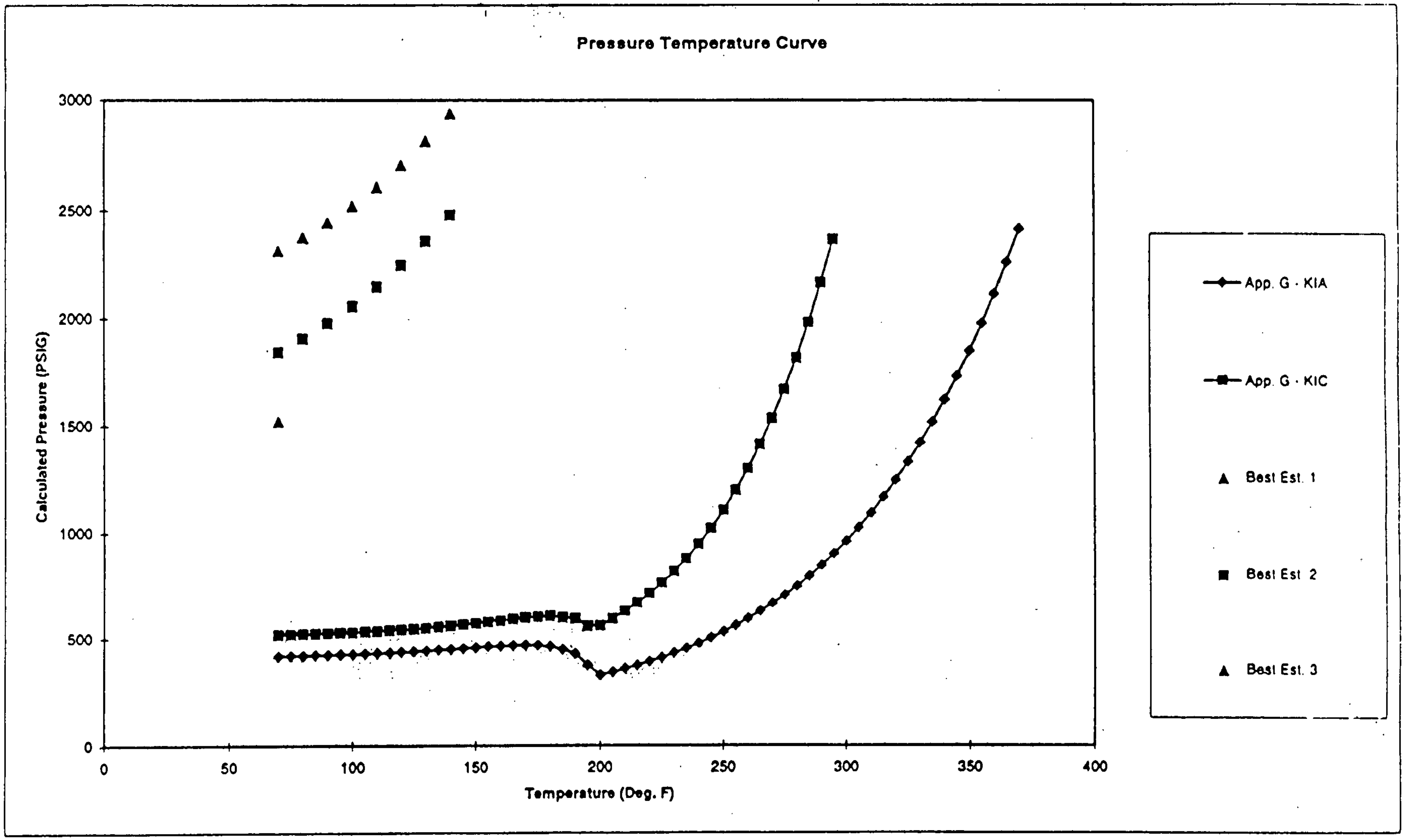

Figure 4. P-T Limit Curves Illustrating Deterministic Safety Factors

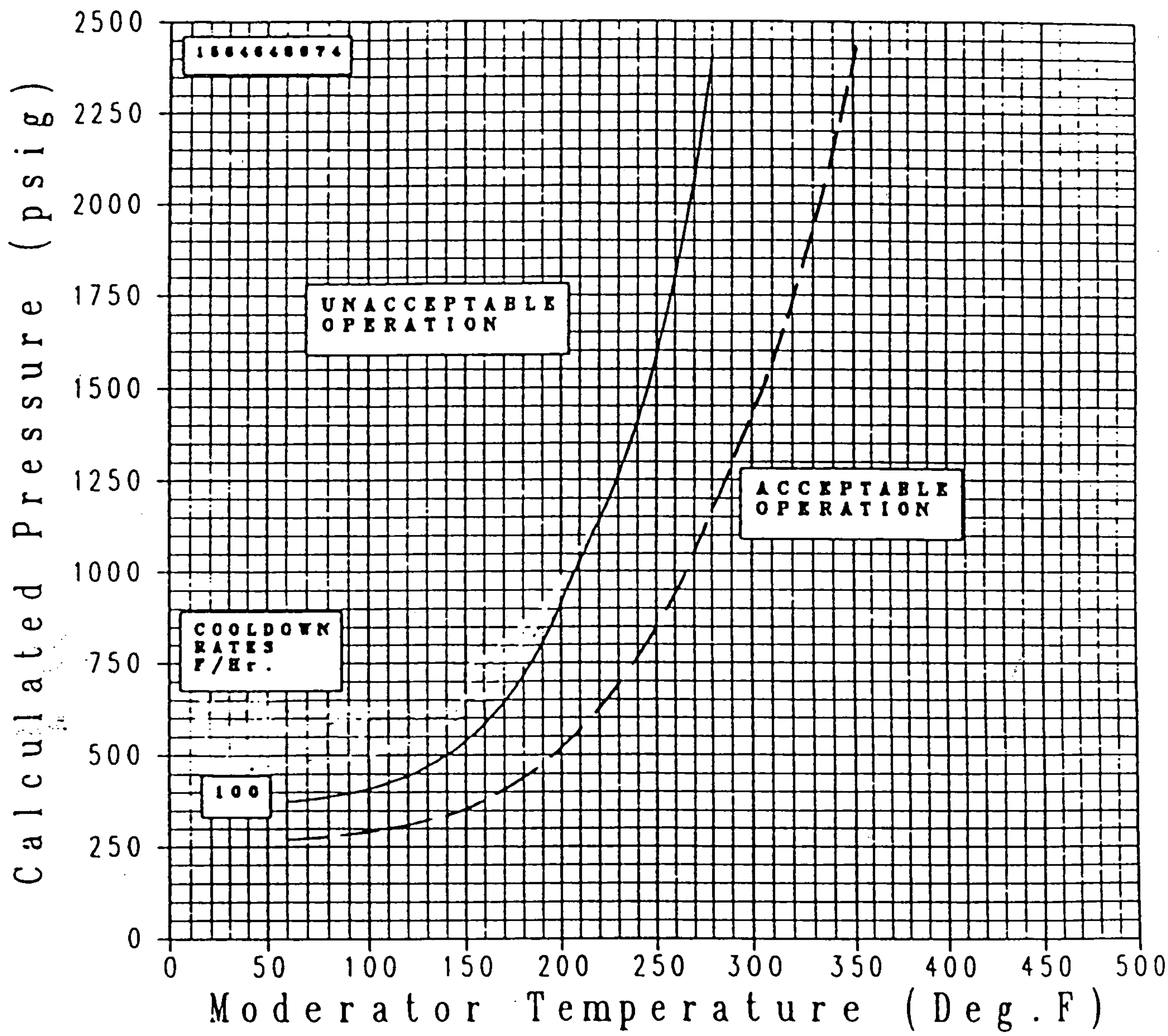

Figure 5. Comparison of Cool-Down Curves for the Existing and Proposed Methods
[Dashed Curve = Existing (K_{IA}) and Solid Curve = Proposed (K_{IC})]

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.

Guidelines for presentation of Results

The results of each of these curves should be presented in tabular form, as well as graphically. The scale on the graph should be as close as practical to the example provided.

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, 93 $K_{IC} = 36.36 + 51.59 \exp[0.0115 (T-RT_{NDT})]$
Loading:	100F/Hr cooldown from 550F to 200F — Film coefficient : 20F/Hr cooldown from 200F to 70F — h = 1000 B/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, 95

Irradiation Effects: RT_{NDT} = 236°F @ inside surface
= 220°F @ depth = 1.0 in.
= 200°F @ depth = T/4
= 133°F @ 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 @ inside surface
= 220F @ depth = 1.0 inch
= 200F @ depth = T/4
= 133F @ 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 Coe, per Terry Dickson, 7/9/98

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

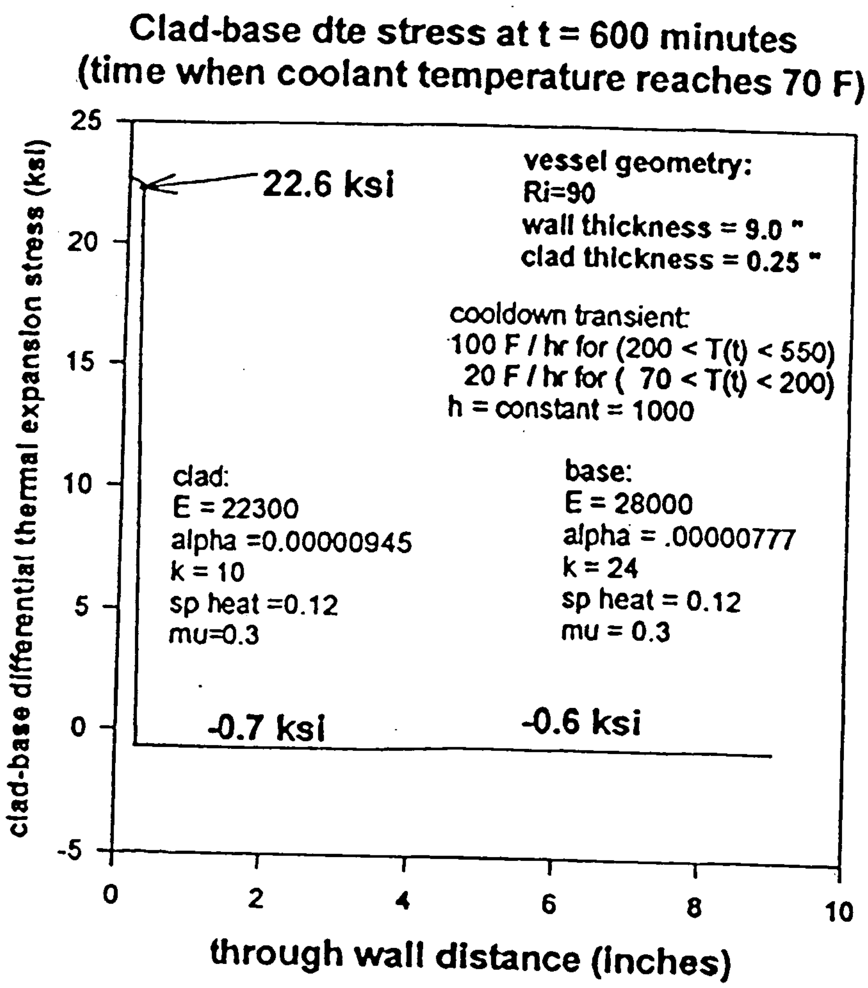

Figure A-1: Clad-base dte stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

From "Effects of Cladding on Fracture Analysis," by W. H. Bamford and A. J. Bush, to be published at ASME PVP Conference, July 1998

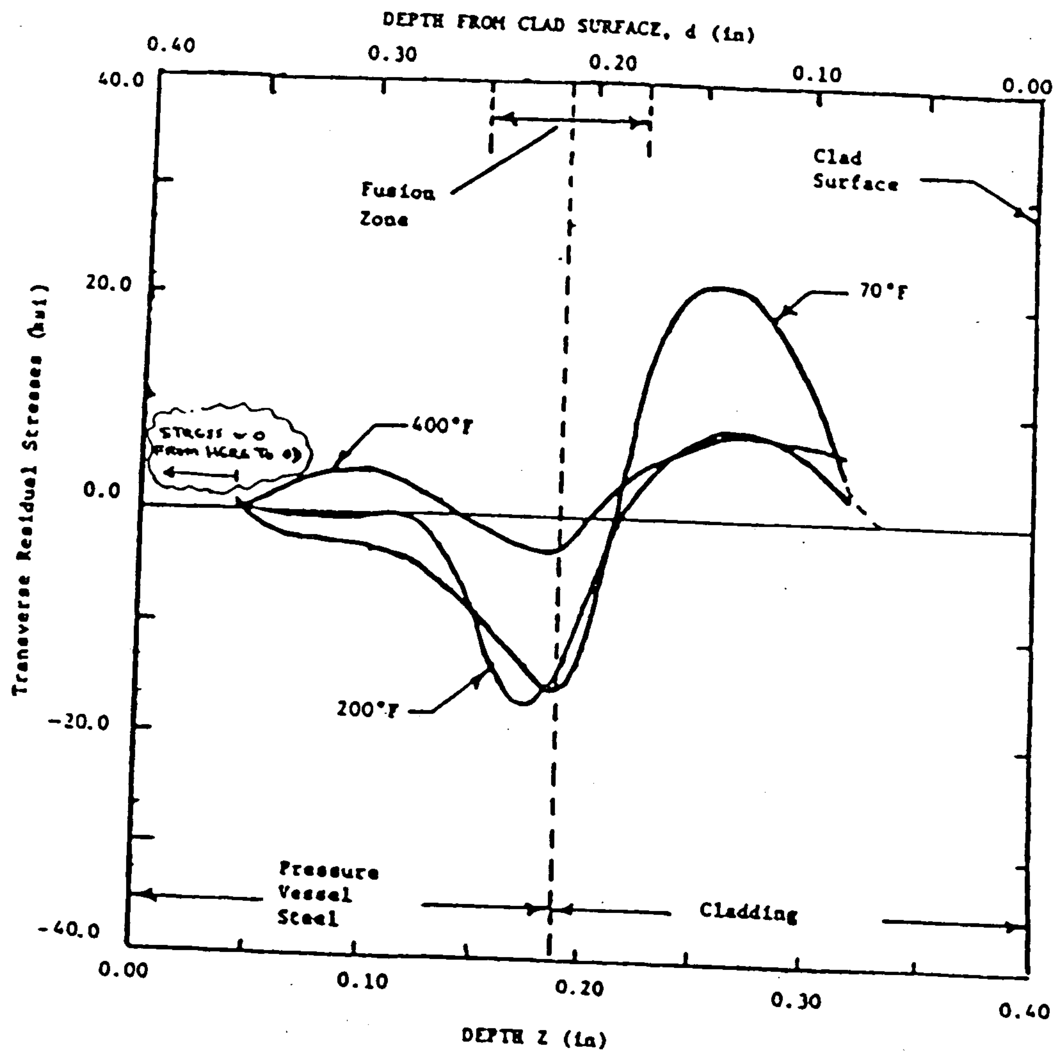

Figure A-2: Residual Stresses Transverse to Direction of Welding

ISI #98-0

ARTICLE G-1000
INTRODUCTION

This Appendix presents a procedure for obtaining the allowable loadings for ferritic pressure retaining materials in components. This procedure is based on the principles of linear elastic fracture mechanics. At each location being investigated a maximum postulated flaw is assumed. At the same location the *mode I stress intensity factor*[1] K_I is produced by each of the specified loadings as calculated and the summation of the K_I values is compared to a reference value K_{IR} which is the highest critical value of K_I that can be endured for the material and temperature involved. Different procedures are recommended for different components and operating conditions. K_{IC}

[1] The *stress intensity factor* as used in fracture mechanics has no relation to and must not be confused with the stress intensity used in Section III, Division 1. Furthermore, stresses referred to in this Appendix are calculated normal tensile stresses not stress intensities in a defect free stress model at the surface nearest the location of the assumed defect.

ARTICLE G-2000
VESSELS

G-2100 GENERAL REQUIREMENTS

G-2110 REFERENCE CRITICAL STRESS INTENSITY FACTOR

(a) Figure G-2210-1 is a curve showing the relationship that can be conservatively expected between the critical, or reference, stress intensity factor ~~K_{IR}~~ K_{IC}, ksi $\sqrt{\text{in.}}$, and a temperature which is related to the reference nil-ductility temperature RT_{NDT} determined in NB-2331. This curve is based on the lower bound of static, ~~dynamic, and crack arrest~~ critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, and SA-508-1, SA-508-2, and SA-508-3 steel. No available data points for static, ~~dynamic, or arrest~~ tests fall below the curve. An analytical approximation to the curve is:

$$K_{Ic} = 33.2 + 21.734\exp[0.02(T - RT_{NDT})]$$

$$\cancel{K_{IR} = 26.78 + 1.233\exp(0.0145(T - RT_{NDT} + 160))}$$

Unless higher ~~K_{IR}~~ K_{IC} values can be justified for the particular material and circumstances being considered, Fig. G-2210-1 may be used for ferritic steels which meet the requirements of NB-2331 and which have a specified minimum yield strength at room temperature of 50.0 ksi or less.

(b) For materials which have specified minimum yield strengths at room temperature greater than 50.0 ksi but not exceeding 90.0 ksi, Fig. G-2210-1 may be used provided fracture mechanics data ~~similar to the K_{IR} data referenced in WRCB 175~~ are obtained on at least three heats of the material on a sufficient number of specimens to cover the temperature range of interest, including the weld metal and heat-affected zone, and provided that the data are equal to or above the curve of Fig. G-2210-1. These data shall be documented by the Owner. Where these materials of higher yield strengths (specified minimum yield strength greater than 50.0 ksi but not exceeding 90.0 ksi) are to be used in conditions where radiation may affect the material properties, the effect of radiation on the ~~K_{IR}~~ K_{IC} curve shall be determined for the material. This information shall be documented by the Owner.

G-2120 MAXIMUM POSTULATED DEFECT A96

The postulated defect used in this recommended procedure is a sharp, surface defect normal to the direction of maximum stress. For section thicknesses of 4 in. to 12 in., it has a depth of one-fourth of the section thickness and a length of 1½ times the section thickness. Defects are postulated at both the inside and outside surfaces. For sections greater than 12 in. thick, the postulated defect for the 12 in. section is used. For sections less than 4 in. thick, the 1 in. deep defect is conservatively postulated. Smaller defect sizes[1] may be used on an individual case basis if a smaller size of maximum postulated defect can be ensured. Due to the safety factors recommended here, the prevention of nonductile fracture is ensured for some of the most important situations even if the defects were to be about twice as large in linear dimensions as this postulated maximum defect.

G-2200 LEVEL A AND LEVEL B SERVICE LIMITS

G-2210 SHELLS AND HEADS REMOTE FROM DISCONTINUITIES

G-2211 Recommendations

The assumptions of this Subarticle are recommended for shell and head regions during Level A and B Service Limits.

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[1] WRCB 175 (Welding Research Council Bulletin 175) "PVRC Recommendations on Toughness Requirements for Ferritic Materials" provides procedures in Paragraph 5(c)(2) for considering maximum postulated defects smaller than those described.

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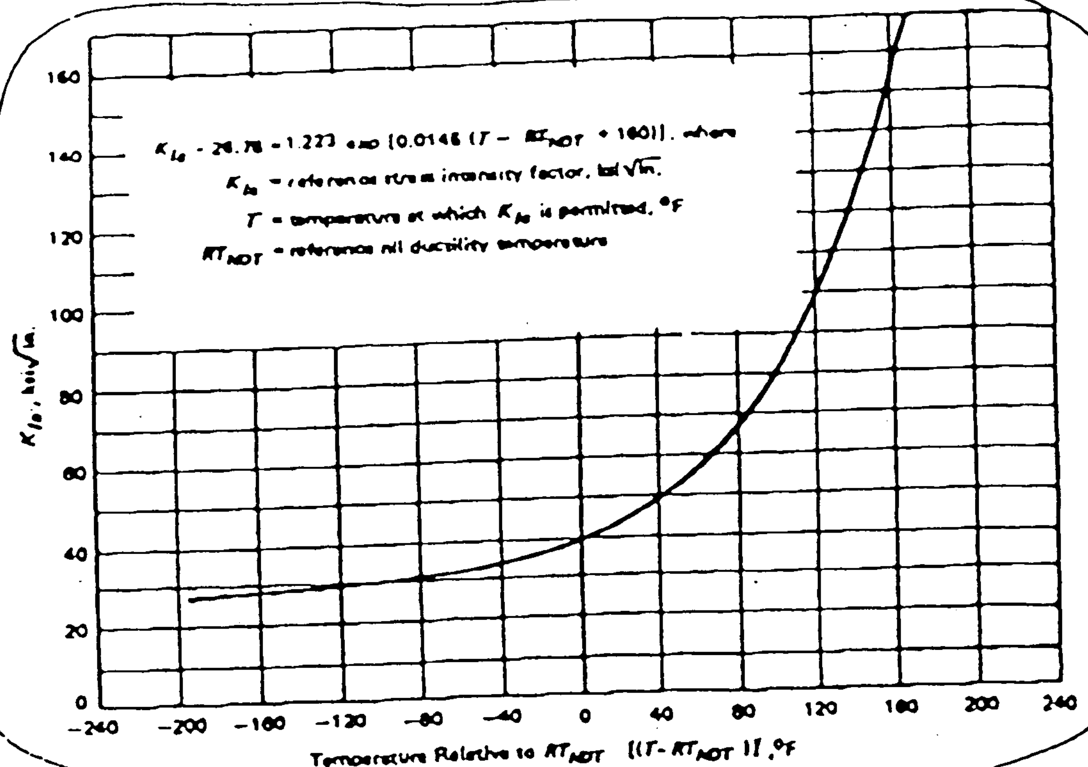


FIG. G-2210-1

G-2212 Material Fracture Toughness

G-2212.1 Reference Critical Stress Intensity Factor for Material. The K_{Ia} values of Fig. G-2210-1 are recommended. K_{Ic}

G-2212.2 Irradiation Effects. Subarticle A-4400 of Appendix A is recommended to define the change in reference critical stress intensity factor due to irradiation.

G-2213 Maximum Postulated Defect

The recommended maximum postulated defect is that described in G-2120.

G-2214 Calculated Stress Intensity Factors

A96 **G-2214.1 Membrane Tension.** The K_I corresponding to membrane tension for the postulated defect of G-2120 is K_{Im} = M_m × (pR_i/t), where M_m for an inside surface is given by

M_m = 1.85 for $\sqrt{t} < 2$

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

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

Similarly, M_m for an outside surface flaw is given by

M_m = 1.77 for $\sqrt{t} < 2$

M_m = 0.893 $\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$

M_m = 3.09 for $\sqrt{t} > 3.464$

where

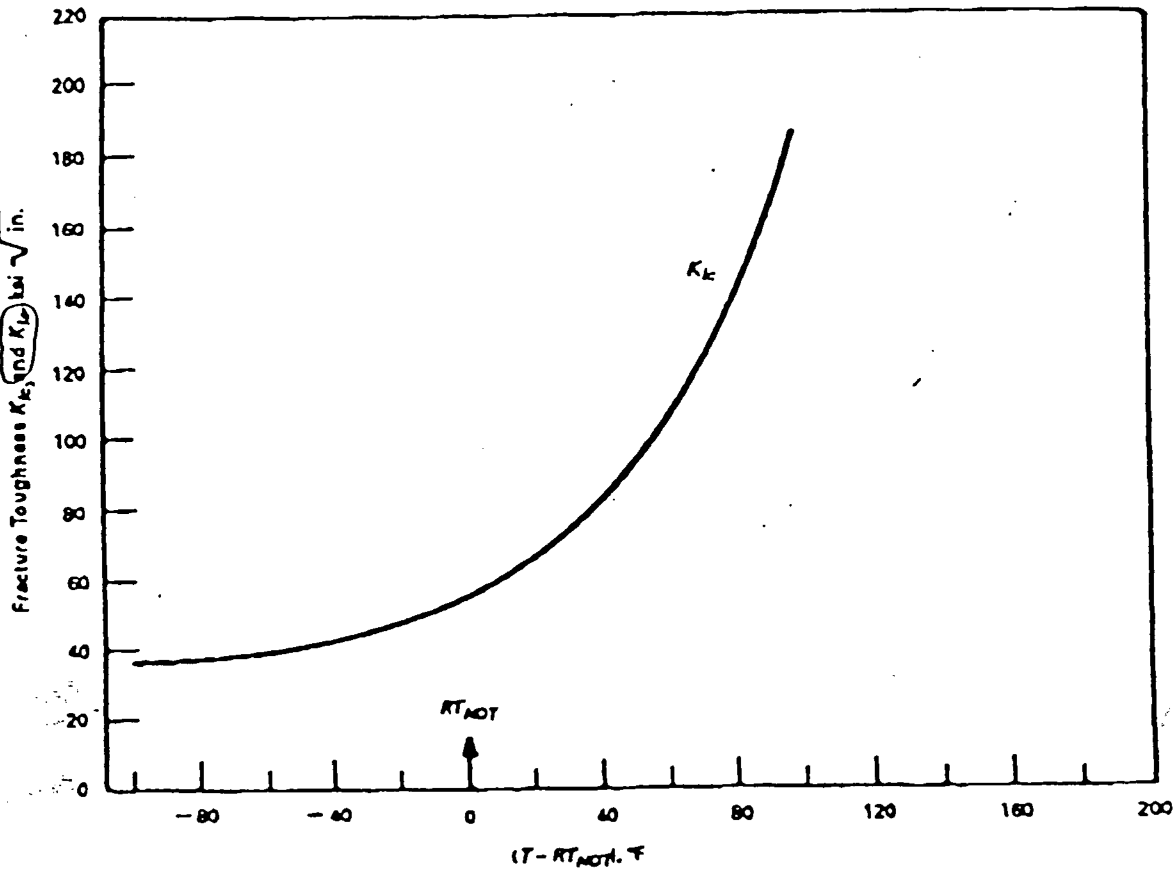
p = internal pressure (ksi)

R_i = vessel inner radius (in.)

t = vessel wall thickness (in.)

G-2214.2 Bending Stress. The K_I corresponding to bending stress for the postulated defect of G-2120 is K_{Ib} = M_b × maximum bending stress, where M_b is two-thirds of M_m. A96

G-2214.3 Radial Thermal Gradient. The maximum K_I produced by a radial thermal gradient for the postulated inside surface defect of G-2120 is K_{It} = 0.953 × 10^{-3} × CR × $t^{2.5}$, where CR is the cooldown rate in F/hr., or, for a postulated outside surface defect, K_{It} = 0.753 × 10^{-3} × HU × $t^{2.5}$, where HU is the heatup rate in F/hr. A96



G2210-1
FIG. ~~A-4200-1~~ LOWER BOUND [redacted] K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

A96 FIG. G-2214-1 DELETED

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

A96 *(a)* The maximum thermal K_I and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions in G-2214.3(a)(1) and (2).

A96 *(1)* An assumed shape of the temperature gradient is approximately as shown in Fig. G-2214-2.

(2) The temperature change starts from a steady state condition and has a rate, associated with startup and shutdown, less than about 100°F/hr. The results would be overly conservative if applied to rapid temperature changes.

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution at any specified time during cooldown for a $\frac{1}{4}$-thickness surface defect. A96

For an inside surface defect during cooldown

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a}$$

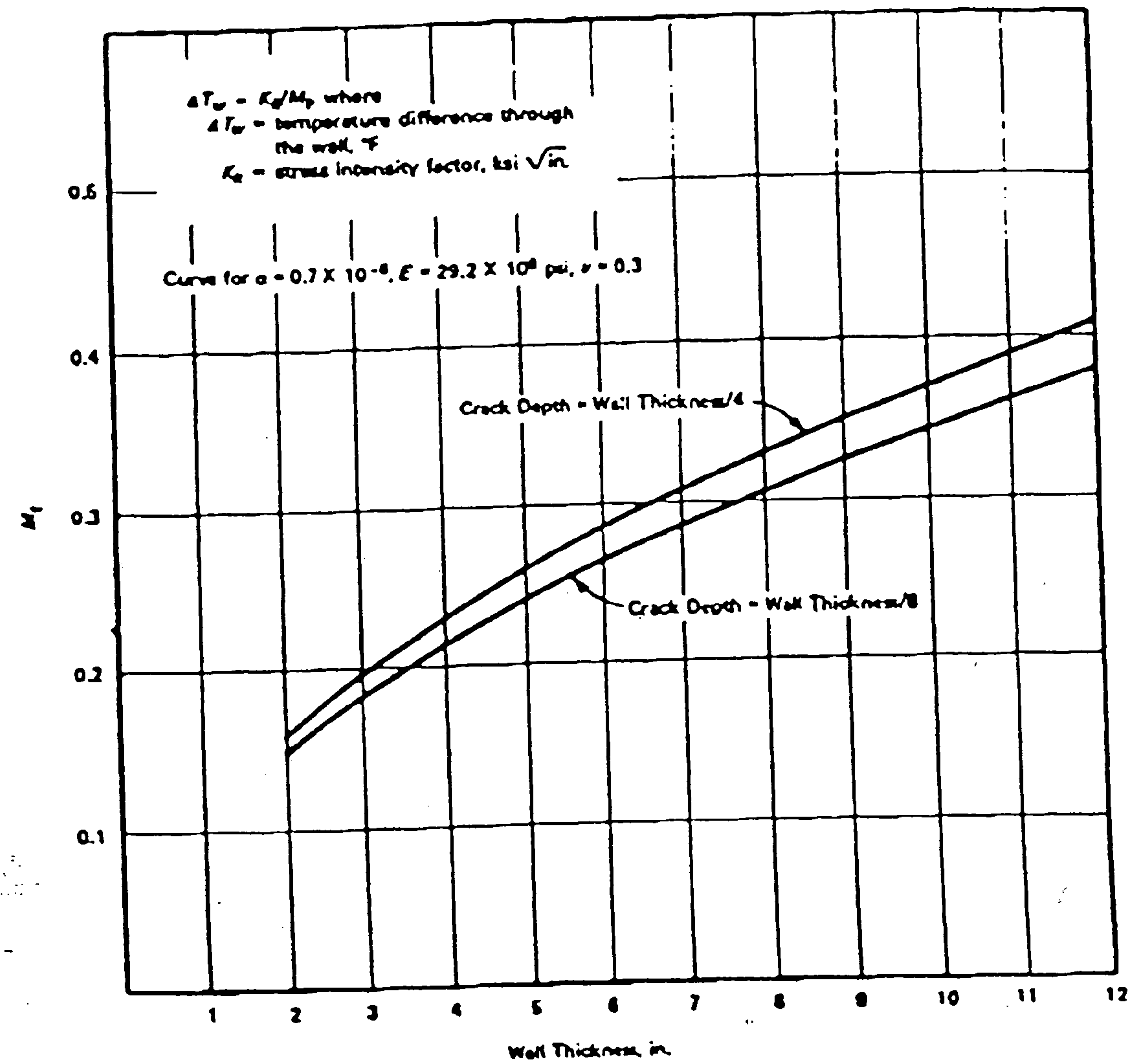


FIG. G-2214-1

A96

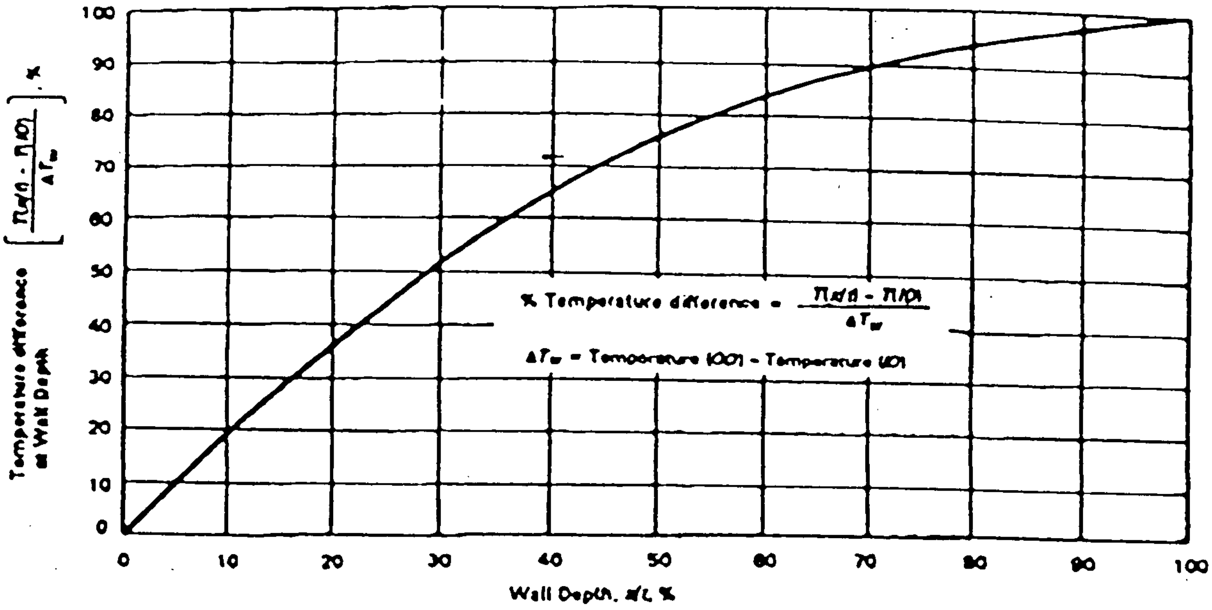


FIG. G-2214-2 A96

For an outside surface defect during heatup

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + (0.401C_3))\sqrt{\pi a}$$

The coefficients C_0, C_1, C_2, and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3$$

where x is a dummy variable that represents the radial distance, in., from the appropriate (i.e., inside or outside) surface and a is the maximum crack depth, in.

(c) For the startup condition, the allowable pressure vs. temperature relationship is the minimum pressure at any temperature, determined from (1) the calculated steady state results for the $\frac{1}{4}$-thickness inside surface defect, (2) the calculated steady state results for the $\frac{1}{4}$-thickness outside surface defect, and (3) the calculated results for the maximum allowable heatup rate using a $\frac{1}{4}$-thickness outside surface defect.

G-2215 Allowable Pressure A96

The equations given in this Subarticle provide the basis for determination of the allowable pressure at any temperature at the depth of the postulated defect during Service Conditions for which Level A and Level B Service Limits are specified. In addition to the conservatism of these assumptions, it is recommended that a factor of 2 be applied to the calculated K_I values produced by primary stresses. In shell and head regions remote from discontinuities, the only significant loadings are: (1) general primary membrane stress due to pressure; and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Therefore, the requirement to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined is:

K_{Ic}

$$2K_{Im} + K_{It} < \cancel{K_{Ia}} \quad (1)$$

throughout the life of the component at each temperature with K_{Im} from G-2214.1, K_{It} from G-2214.3, and $\cancel{K_{Ia}}$ from Fig. G-2210-1.

K_{Ic}

Those plants having low temperature overpressure protection (LTOP) systems can use the following load and temperature conditions to provide protection against

failure during reactor start-up and shutdown operation due to low temperature overpressure events that have been classified as Service Level A or B events. LTOP systems shall be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater.[2,3] LTOP systems shall limit the maximum pressure in the vessel to 110% of the pressure determined to satisfy Eq. (1).

G-2220 NOZZLES, FLANGES, AND SHELL REGIONS NEAR GEOMETRIC DISCONTINUITIES

G-2221 General Requirements

The same general procedure as was used for the shell and head regions in G-2210 may be used for areas where more complicated stress distributions occur, but certain modifications of the procedures for determining allowable applied loads shall be followed in order to meet special situations, as stipulated in G-2222 and G-2223.

G-2222 Consideration of Membrane and Bending Stresses

(a) Equation (1) of G-2215 requires modification to include the bending stresses which may be important contributors to the calculated K_I value at a point near a flange or nozzle. The terms whose sum must be $< K_{Ia}$ for normal and upset operating conditions are:

(1) $2K_{Im}$ from G-2214.1 for primary membrane stress;

(2) $2K_{Ib}$ from G-2214.2 for primary bending stress;

(3) K_{Im} from G-2214.1 for secondary membrane stress;

(4) K_{Ib} from G-2214.2 for secondary bending stress.

(b) For purposes of this evaluation, stresses which result from bolt preloading shall be considered as primary.

(c) It is recommended that when the flange and adjacent shell region are stressed by the full intended bolt preload and by pressure not exceeding 20% of the preoperational system hydrostatic test pressure, minimum metal temperature in the stressed region should be at least the initial RT_{NDT} temperature for the material in the stressed regions plus any effects of irradiation at the stressed regions.

(d) Thermal stresses shall be considered as secondary except as provided in NB-3213.13(b). The K_I of G-2214.3(b) is recommended for the evaluation of thermal stress. A96

G-2223 Toughness Requirements for Nozzles

(a) A quantitative evaluation of the fracture toughness requirements for nozzles is not feasible at this time, but preliminary data indicate that the design defect size for nozzles, considering the combined effects of internal pressure, external loading and thermal stresses, may be a fraction of that postulated for the vessel shell. Nondestructive examination methods shall be sufficiently reliable and sensitive to detect these smaller defects.

(b) WRCB 175 provides an approximate method in Paragraph 5C(2) for analyzing the inside corner of a nozzle and cylindrical shell for elastic stresses due to internal pressure stress.

(c) Fracture toughness analysis to demonstrate protection against nonductile failure is not required for portions of nozzles and appurtenances having a thickness of 2.5 in. or less, provided the lowest service temperature is not lower than RT_{NDT} plus 60°F.

G-2300 LEVEL C AND LEVEL D SERVICE LIMITS

G-2310 RECOMMENDATIONS

The possible combinations of loadings, defect sizes, and material properties which may be encountered during Level C and Level D Service Limits are too diverse to allow the application of definitive rules, and it is recommended that each situation be studied on an individual case basis. The principles given in this Appendix may be applied, where applicable, with any postulated loadings, defect sizes, and material toughness which can be justified for the situation involved.

G-2400 HYDROSTATIC TEST TEMPERATURE

(a) For system and component hydrostatic tests performed prior to loading fuel in the reactor vessel, it is recommended that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60°F. The

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[2] The coolant temperature is the reactor coolant inlet temperature.

[3] The vessel metal temperature is the temperature at a distance one fourth of the vessel section thickness from the inside wetted surface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature (for weld or base metal in the beltline region) at a distance one fourth of the vessel section thickness from the vessel wetted inner surface as determined by Regulatory Guide 1.99, Rev. 2

60°F margin is intended to provide protection against nonductile failure at the test pressure.

(b) For system and component hydrostatic tests performed subsequent to loading fuel in the reactor vessel, the minimum test temperature should be determined by evaluating K_I. The terms given in (1) through (4) below should be summed in determining K_I:

(1) $1.5K_{Im}$ from G-2214.1 for primary membrane stress;

(2) $1.5K_{Ib}$ from G-2214.2 for primary bending stress;

(3) K_{Im} from G-2214.1 for secondary membrane stress;

(4) K_{Ib} from G-2214.2 for secondary bending stress.

K_I, calculated by summing the four values given in (1) through (4) above, shall not exceed the applicable $\cancel{K_{Ia}}$ value.

K_{Ic}

(c) The system hydrostatic test to satisfy G-2400(a) or (b) should be performed at a temperature not lower than the highest required temperature for any component in the system.

ARTICLE G-3000
PIPING, PUMPS, AND VALVES

G-3100 GENERAL REQUIREMENTS

In the case of the materials other than bolting used for piping, pumps, and valves for which impact tests are required (NB-2311), the tests and acceptance standards of Section III, Division 1 are considered to be adequate to prevent nonductile failure under the loadings and with the defect sizes encountered under normal, upset, and testing conditions. Level C and Level D Service Limits should be evaluated on an individual case basis (G-2300).

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Duke Energy Corporation (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10CFR50.92. This ensures operation of the facility in accordance with the proposed amendment will not:

A. Involve a significant increase in the probability or consequences of an accident previously evaluated?

NO.

These proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N-514, N-588 and N-626, as described in the Technical Justification. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provide safety limits and margins of safety that ensure failure of a reactor vessel will not occur.

The proposed changes do not impact the capability of the reactor coolant pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed Pressure-Temperature (P-T) limits, Low Temperature Overpressure (LTOP) limits and setpoints, and allowable operating reactor coolant pump combinations are not considered to be an initiator or contributor to any accident analysis addressed in the Oconee UFSAR.

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the proposed changes do not affect any fission product barrier. The

revised PORV LTOP setpoint is established to protect reactor coolant pressure boundary. The changes do not degrade or prevent the response of the PORV or safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by approval of the requested changes.

B. Create the possibility of a new or different kind of accident from the accident previously evaluated?

NO.

The proposed license amendment revises the Oconee reactor vessel P-T limits, LTOP limits and setpoints, and allowable operating reactor coolant pump combinations. Compliance with 10 CFR 50 Appendix G, includes utilization of ASME XI, Appendix G, as modified by Code Cases N-514, N-588 and N-626 to meet the underlying intent of the regulations.

Operation of Oconee in accordance with these proposed Technical Specifications changes will not create any failure modes not bounded by previously evaluated accidents. Consequently, approval of these changes will not create the possibility of a new or different accident from any accident previously evaluated.

C. Involve a significant reduction in a margin of safety?

NO.

The proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N-514, N-588 and N-626, as described in the Technical Justification. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

of safety which ensure failure of a reactor vessel will not occur.

No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted. Therefore, there will be no significant reduction in any margin of safety as a result of approval of the requested changes.

Duke has concluded based on this information there are no significant hazards considerations involved in this amendment request.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10CFR51.22 (b), an evaluation of the proposed amendments has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the No Significant Hazards Consideration Evaluation which is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   This amendment will not significantly change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   This amendment will not significantly increase the individual or cumulative occupation radiation exposure.

In summary, this amendment request meets the criteria set forth in 10CFR51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.