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10CFR50.12

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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation"

References:

(1) Letter from R. A. Capra (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated January 21, 1998

(2) Letter from S.N. Bailey (USNRC) to O.D. Kingsley (Commonwealth Edison Company), "Quad Cities – Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G," dated February 4, 2000

(3) Letter from D. E. LaBarge (USNRC) to W. R. McCollum, Jr., (Duke Energy Corporation), "Oconee Nuclear Station, Units 1,2, and 3 Re: Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a)," dated July 29, 1999

(4) Publication PVP – Volume 407, Pressure Vessel and Piping Codes and Standards – 2000 ASME 2000, "Technical Basis for Revised P-T Limit Curve Methodology"

(5) Letter from R. M. Krich (Commonwealth Edison Company) to USNRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

10CFR50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," paragraph (a) requires that the fracture

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toughness and material surveillance program for the reactor pressure boundary for all light-water nuclear power reactors must meet the requirements set forth in 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Specifically, 10 CFR 50 Appendix G, Table 1, provides the pressure-temperature (P-T) limits and minimum temperature requirements for the reactor pressure vessel (RPV) during normal operating and hydrostatic or leak test conditions. The requirements for P-T limits are the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," limits. 10CFR50.60(b) allows alternatives to the requirements of 10 CFR 50 Appendices G and H when an exemption is granted by the NRC in accordance with 10 CFR 50.12, "Specific exemptions." The current Commonwealth Edison (ComEd) Company methodology used to generate the Pressure Temperature Limits Report (PTLR), which contains the P-T curves and the low temperature overpressure protection (LTOP) system limits, is based on the methodology approved in a Safety Evaluation transmitted to ComEd by NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," (i.e., Reference 1). This approved methodology includes the use of Westinghouse Electric Company Report, WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, ASME Code Case N-514, "Low Temperature Overpressure, Section XI, Division 1," and ASME Section XI, Appendix G, 1996 Addenda.

Pursuant to 10 CFR 50.12, we are requesting an exemption from 10 CFR 50.60. In References 2 and 3, the NRC recently approved similar changes for Quad Cities Nuclear Power Station and Oconee Nuclear Station, respectively. In particular, the requested exemption will allow the use 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," Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," and Westinghouse Electric Company Report, WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in calculating the RPV P-T limits. Upon receiving approval of the requested exemption, the provisions of Code Case N-514 for establishing the LTOP pressure setpoint at 110% of the allowable pressure, provided by the P-T limit curves, will no longer be used. The LTOP setpoint will be established at 100% of the allowable pressure established by the P-T limit curve.

ASME B&PV Code Case N-640 allows the use of alternate material fracture toughness when determining the minimum RPV reference temperature - nil ductility transition, (RT_{NDT}), i.e., the use of K_{Ic}, the lower bound static fracture toughness curve, rather than K_{Ia}, the lower bound fracture toughness curve, as defined in ASME B&PV Code Section XI, Appendix A, "Analysis of Flaws." The technical basis for the revised P-T limit curve methodology utilizing K_{Ic} can be found in Publication PVP – Volume 407, Pressure Vessel and Piping Codes and Standards – 2000 ASME 2000, "Technical Basis for

Revised P-T Limit Curve Methodology," (i.e., Reference 4). ASME B&PV Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors for axial and circumferential flaws. The methodology in WCAP 15315 eliminates the requirement that the minimum temperature of the highly stressed region of the closure flange exceed the unirradiated RT_{NDT} by 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic pressure. We have determined that the use of ASME B&PV Code Cases N-640 and N-588 and WCAP 15315 will require NRC approval of an exemption to 10 CFR 50.60(a). The justification for the exemption request is contained in Attachment A. A copy of WCAP 15315 is included in Attachment C.

In Reference 5, ComEd requested that the NRC review and approve operation of Byron Station and Braidwood Station at uprated power conditions by May 7, 2001. Based on projected uprated power operations and new reactor vessel capsule surveillance data, the current PTLRs for Byron Station and Braidwood Station need to be updated. We intend to use the alternate methods described in this exemption request to update the PTLRs for Byron Station and Braidwood Station. In order to support uprated power operations at Byron Station and Braidwood Station, we request that the NRC approve this exemption to 10 CFR 50.60(a) prior to May 7, 2001. We understand that the NRC is currently reviewing WCAP 15315 submitted by the Westinghouse Owner Group; however, in the event the NRC does not approve WCAP 15315 for generic use prior to May 7, 2001, we request that the WCAP 15315 methodology be specifically approved for use at Byron Station and Braidwood Station.

Should you have any questions concerning this letter, please contact J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Director, Licensing
Mid-West Regional Operating Group

Attachments: Attachment A, "Justification for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation"
Attachment B, Environmental Assessment
Attachment C, WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants"

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station
NRC Senior Resident Inspector – Braidwood Station

ATTACHMENT A

Justification for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation"

In accordance with 10 CFR 50.12, "Specific exemptions," Commonwealth Edison (ComEd) Company is requesting an exemption from the requirement of 10 CFR 50.60 "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, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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 Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and Westinghouse Electric Company Report, WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in lieu of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraphs IV.A.2.b and IV.A.2.c as required by 10 CFR 50.60(a).

The current ComEd methodology used to generate the Pressure Temperature Limits Report (PTLR), which contains the pressure-temperature (P-T) curves and the low temperature overpressure protection (LTOP) system limits, is based on the methodology approved in a Safety Evaluation transmitted to ComEd by NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report." This approved methodology includes the use of Westinghouse Electric Company Report, WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, ASME Code Case N-514, "Low Temperature Overpressure, Section XI, Division 1," and ASME Section XI, Appendix G, 1996 Addenda. Upon receiving approval of the requested exemption, the provisions of Code Case N-514 for establishing the LTOP system pressure setpoint at 110% of the allowable pressure, provided by the P-T limit curves, will no longer be used. The LTOP system setpoint will be established at 100% of the allowable pressure established by the P-T limit curve.

Justification for Use of Code Case N-640

10 CFR 50.12(a) Requirements

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

10 CFR 50.12 states that the NRC may grant an exemption from requirements contained in 10 CFR 50 provided that the following are 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, Appendix G and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," when an exemption is granted by the NRC under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety

The methodology ComEd will use to revise the P-T limits for Byron Station, Units 1 and 2, and Braidwood Station, Unit 1 and 2, relies in part, on the requested exemption. The P-T limits will be developed using the K_{Ic} fracture toughness curve shown on ASME B&PV Code, Section XI, Appendix A, "Analysis of Flaws," Figure A-2200-1, in lieu of the K_{Ia} static fracture toughness curve of ASME B&PV Code, Section 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 technically more correct than the K_{Ia} curve. The K_{Ic} curve models the slow heatup and cooldown process of an RPV. As discussed in Publication PVP – Volume 407, Pressure Vessel and Piping Codes and Standards – 2000 ASME 2000, "Technical Basis for Revised P-T Limit Curve Methodology," the pressure and thermal stresses associated with normal operating and anticipated operating transient events correspond to static loading conditions when determining the material fracture toughness response to these loads.

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 RPV material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of RPV materials and their fracture response to applied loads. As described in Publication PVP – Volume 407, Pressure Vessel and Piping Codes and Standards – 2000ASME 2000, "Technical Basis for Revised P-T Limit Curve Methodology," 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 RPV failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential RPV 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 P-T operating window, especially in the region of low temperature operations. The two primary safety benefits that would be realized are a reduction in the challenges to the LTOP system and a reduction in the risk of damaging the reactor coolant pump seals due to pump operation under conditions where it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

3. The requested exemption is consistent with the common defense and security

   Implementation of this exemption request (i.e., Code Case N-640) will have no effect on the common defense and security.

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) – application of the regulation is not necessary to achieve the underlying purpose of the rule;

   (a)(2)(iv) – would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption;

   (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.

10 CFR 50.12(a)(2)(ii)

ASME B&PV Code, Section XI, Appendix G, provides procedures for determining allowable loading on the RPV 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 limit 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 RPV 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 RPV 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 B&PV 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.

10 CFR 50.12(a)(2)(iv)

The RPV P-T 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 Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, with these P-T curves without the relief provided by ASME B&PV Code Case N-640 would unnecessarily restrict the P-T operating window. This operating window defines the space between the LTOP system setpoint and the minimum pressure for reactor coolant pump operation. The more restricted this space is, the greater the potential for inadvertent Power Operated Relief Valve (PORV) actuation as a result of reaching the LTOP system setpoint, or reactor coolant pump seal damage due to pump operation under conditions where it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

This constitutes an unnecessary challenge that can be alleviated by the application of ASME B&PV Code Case N-640 in the development of the P-T curves.

Implementation of the P-T curves as allowed by ASME B&PV Code Case N-640 does not significantly reduce the margin of safety below that established by the original requirement. In fact the probability of an inadvertent PORV actuation and probability of reactor coolant pump seal damage are both reduced. Both events have the potential to cause an inadvertent reduction in reactor coolant inventory. Reducing the probability of these events is a benefit to the public health and safety.

10 CFR 50.12(a)(2)(v)

The requested exemption provides only temporary relief from the applicable regulation and Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, have been in compliance with the regulation. We request the exemption be granted until such time that the NRC generically approves ASME B&PV 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 unnecessarily restrict the P-T operating window and result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV Code Case N-640 allows a reduction in the lower bound fracture toughness used by ASME B&PV Code, Section XI, Appendix G, to determine the reactor coolant system (RCS) P-T limits. This proposed alternative is acceptable because the ASME B&PV 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 B&PV Code Case N-640 for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

Justification for the Use of Code Case N-588

10 CFR 50.12(a) Requirements

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

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, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent P-T limits specified in the Braidwood and Byron Stations' Technical Specifications (TS), Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits." Therefore, this requested exemption does not present an undue risk to the public health and safety.

3. <u>The requested exemption is consistent with the common defense and security</u>

   Implementation of this exemption request (i.e., Code Case N-588) will have no effect on the common defense and security.

4. <u>Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60</u>

   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) – application of the regulation is not necessary to achieve the underlying purpose of the rule;

(a)(2)(iv) - would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption;

(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.

10 CFR 50.12(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 requirements that:

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

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

10 CFR 50.12(a)(2)(iv)

The RPV P-T 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 with the current P-T limit curves without the relief provided by ASME B&PV Code Case N-588 would unnecessarily restrict the P-T operating window for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. This operating window defines the space between the LTOP system setpoint and the minimum pressure for reactor coolant pump operation. The more restricted this space is, the greater the potential for inadvertent PORV actuation or reactor coolant pump seal damage due to insufficient cooling water pressure.

The present methodology provides a restrictive setpoint, which constitutes an unnecessary challenge that can be alleviated by the application of ASME B&PV Code Case N-588 in the development the P-T curves. Implementation of the P-T limit curves as allowed by ASME B&PV Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or the ASME. Applying

Code Case N-588 decreases the possibility of inadvertent PORV actuation or reactor coolant pump damage and thereby, is of benefit to the public health and safety.

10CFR50.12(a)(2)(v)

The exemption provides only temporary relief from the applicable regulation and Byron and Braidwood Stations have been in compliance with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME B&PV Code Case N-588 for use by the nuclear industry.

ASME B&PV Code Case N-588, Conclusion for Exemption Acceptability

Compliance with the specified requirements of 10 CFR 50.60 would unnecessarily restrict the P-T operating window and result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV 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, in the axial direction, from one plate or forging to the adjoining plate or forging. A circumferential defect orientation was not considered at the time ASME B&PV Code, Section XI, Appendix G was developed and; therefore, Appendix G imposes overly conservative restrictions on P-T operating limits based on an assumed axial orientation of the defect.

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 B&PV Code Case N-588 for Byron and Braidwood Stations will ensure an acceptable margin of safety for all design bases overpressure events.

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, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent P-T limits specified in the PTLR. Therefore, this exemption does not present an undue risk to the public health and safety.

Justification for the Use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirement Evaluation for Operating PWR and BWR Plants"

10 CFR 50.12(a) Requirements

The requested exemption to allow use of WCAP-15315, in conjunction with ASME Section XI, Appendix G, to determine the P-T limits meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the NRC 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 NRC 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, contains requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s, to ensure that appropriate margins would be maintained. As described in Publication PVP – Volume 407, Pressure Vessel and Piping Codes and Standards – 2000 ASME 2000, "Technical Basis for Revised P-T Limit Curve Methodology," additional material fracture toughness 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 RPV failure.

   The proposed methodology for the Byron Station and Braidwood Station P-T limits relies, in part, on ASME Code Case N-640 which allows the use of the K_{Ic} fracture toughness curve rather than the K_{Ia} curve. P-T limits developed using the K_{Ic} fracture toughness curve permit a much higher allowable pressure through the entire range of temperatures. However, the benefit is negated at temperatures below RT_{NDT} +120°F because of the additional flange requirement of 10 CFR part 50, Appendix G. It restricts the ability to raise the PORV setpoint as temperature rises by restricting the setpoint to the flange temperature limit rather than the ASME Code Case N-640 P-T limit.

   Using the K_{Ic} toughness, WCAP-15315 presents analyses which show that there is significant margin between the applied stress intensity factor at boltup and the material fracture toughness at virtually all crack depths. Further, the analyses also showed the boltup temperature requirement for Pressurized Water Reactors (PWRs), using K_{Ic}, ranges from 10°F to 51°F, which is below the nominal boltup temperature of 60°F for a PWR. The results presented in WCAP-15315 demonstrate that the 10 CFR 50 Appendix G closure head/flange requirement can be eliminated and appropriate fracture margins are still maintained.

   Applying WCAP-15315 methodology to the P-T curves will enhance overall plant safety by widening the P-T operating window especially in the region of low temperature operations. The two primary safety benefits that would be realized are a

reduction in the potential challenges to the LTOP system and a reduction in the risk of damaging the reactor coolant pump seals. This will produce a significant improvement in plant safety by reducing the probability of an inadvertent reduction in reactor coolant inventory, and easing the burden on the operators.

3. The requested exemption is consistent with the common defense and security

Implementation of this exemption request (i.e., WCAP 15315) will have no effect on the common defense and security.

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) – application of the regulation is not necessary to achieve the underlying purpose of the rule;

(a)(2)(iv) - would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption;

(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.

10 CFR 50.12(a)(2)(ii)

The underlying purpose of the rule (i.e., 10 CFR 50, Appendix G) will continue to be achieved. 10 CFR 50, Appendix G contains requirements for P-T limits for the RCS, and requirements for the metal temperature of the closure head flange and vessel flange regions. The P-T limits are determined using the methodology of ASME Section XI, Appendix G, with additional, more restrictive, flange temperature requirements specified in 10 CFR 50, Appendix G. This rule states that the metal temperature of the highly stressed region of the closure assembly must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure.

This requirement was originally based on concerns regarding the fracture margin in the closure flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrostatic test pressure were developed in the mid-1970s using the K_{Ia} fracture toughness to ensure that appropriate margins would be maintained.

WCAP-15315, presents analysis results using the K_{Ic} fracture toughness for the flange, which shows the flange fracture margins are still maintained for boltup without

the RT_{NDT} + 120°F requirement. Use of the newly accepted K_{Ic} fracture toughness for flange boltup limits leads to the conclusion that the 10 CFR 50, Appendix G flange requirement can be eliminated.

Another objective of the requirements in 10 CFR 50, Appendix G is to ensure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface where there are no environmental effects, there is even greater assurance of fracture margin. Note that the inside surface is in compression. Therefore, it may be concluded that the integrity of the closure head/flange region is not a concern for any operating plants using the K_{Ic} fracture toughness.

Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage factor for this region is less than 0.1; therefore, it may be concluded that flaws are unlikely to initiate in this region.

Therefore, the underlying purpose of the regulation, to maintain appropriate fracture margin in the closure flange region, will continue to be achieved.

10 CFR 50.12(a)(2)(iv)

The RPV P-T operating window is defined by the P-T operating and test limits developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedures. ComEd has also requested an exemption to use ASME Code Case N-640 which will provide higher allowable pressure limits through the entire range of operating temperatures. However, the metal temperature requirement of the closure head flange, as specified in 10 CFR 50 Appendix G, restricts the allowable pressure at temperatures below RT_{NDT} +120 °F. The continued operation with these P-T limits, without the relief provided by WCAP-15315, would unnecessarily restrict the P-T operating window for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. This operating window is defined by the space between the LTOP system setpoint and the minimum pressure for reactor coolant pump operation. The more restricted this space is, the greater the potential for inadvertent PORV actuation or reactor coolant pump seal damage due to pump operation under conditions where it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

The present methodology provides a more restrictive setpoint, which constitutes an unnecessary challenge that can be alleviated by the application of WCAP-15315. Implementation of the P-T limits as allowed by WCAP-15315 does not reduce the margin of safety originally contemplated by either the NRC or ASME. Applying WCAP-15315 decreases the possibility of inadvertent PORV actuation or reactor coolant pump seal damage and thereby, is of benefit to the public health and safety.

10CFR50.12(a)(2)(v)

The exemption provides only temporary relief from the applicable regulation and Byron and Braidwood Stations have been in compliance with the regulation. We request that the exemption be granted until such time that the NRC generically approves WCAP-15315 for use by the nuclear industry.

WCAP-15315, Conclusion for Exemption Acceptability

Compliance with the specified requirement of 10 CFR 50.60(a) would unnecessarily restrict the P-T operating window and result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV Code Case N-640 allows a reduction in the lower bound fracture toughness used in ASME B&PV Code, Section XI, Appendix G, to determine RCS P-T limits. Compliance with the 10 CFR 50 Appendix G closure head/flange limit would be incongruous with the use of Code Case N-640. Code Case N-640 uses K_{Ic} methodology and the present closure head flange limit is based on K_{Ia} methodology.

WCAP-15315 provides a valid basis for changing the closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR 50, Appendix G requirement was issued. Further, there is a positive benefit to the public by reducing potential inadvertent challenges to the PORVs and increasing the operating margin for reactor coolant pumps seals. Therefore, application of WCAP-15315 for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

ATTACHMENT B

Environmental Assessment

In accordance with 10 CFR 51.30, "Environmental assessment," and 10 CFR 51.32, "Finding of no significant impact," the following information is provided in support of an environmental assessment and finding of no significant impact for the proposed action.

The proposed action would grant an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." Specifically, the exemption would allow the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," and Westinghouse Electric Company Report, WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in calculating the reactor pressure vessel (RPV) pressure-temperature (P-T) limits.

The requested exemption is needed because utilization of Code Case N-588, Code Case N-640 and WCAP-15315 will widen the current narrow P-T operating window, especially in the region of low temperature operations. The two primary safety benefits that would be realized are a reduction in the challenges to the low temperature overpressure protection (LTOP) system resulting in an inadvertent opening of a power operated relief valve (PORV), and a reduction in the risk of damaging the reactor coolant pump seals due to pump operation under conditions where it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

The principal alternative to the proposed action would be to deny the requested exemption and require adherence to the current 10 CFR 50.60 requirements. Denial of the exemption request would result in no change in environmental impacts. Regarding alternative use of resources, granting the requested exemption will not involve the use of resources not previously considered in the Final Environmental Statements for Byron Station and Braidwood Station, (i.e., NUREG-0848, "Final Environmental Statement related to the operation of Byron Station, Units 1 and 2," dated April 1982 and NUREG-1026, "Final Environmental Statement related to the operation of Braidwood Station, Units 1 and 2," dated June 1984).

The proposed action (i.e., granting the exemption) will not increase the probability or consequences of accidents, no changes are being made in the types or quantities of any radiological effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

The proposed action does not affect non-radiological plant effluents and has no other environmental impact. Therefore, there are no significant non-radiological impacts associated with the proposed action.

ATTACHMENT B
(continued)

The environmental impacts of the proposed action and the alternative action are similar. Based on the assessment above, the proposed action will not have a significant effect on the quality of the human environment.

ATTACHMENT C

WCAP – 15315
"Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants"

WCAP-15315

Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants

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1.0 INTRODUCTION

10 CFR Part 50, Appendix G contains requirements for pressure-temperature limits for the primary system, and requirements for the metal temperature of the closure head flange and vessel flange regions. The pressure-temperature limits are to be determined using the methodology of ASME Section XI, Appendix G, but the flange temperature requirements are specified in 10CFR50 Appendix G. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure, which is 621 psig for a typical PWR, and 300 psig for a typical BWR.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s, to ensure that appropriate margins would be maintained.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in ASME Code Case N640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1".

Figure 1-1 illustrates the problem created by the flange requirements for a typical PWR heatup curve. It is easy to see that the heatup curve using K_{Ic} provides for a much higher allowable pressure through the entire range of temperatures. For this plant, however, the benefit is negated at temperatures below RT_{NDT} +120°F because of the flange requirement of 10 CFR Part 50, Appendix G. The flange requirement of 10 CFR 50 was originally developed using the K_{Ia} fracture toughness, and this report will show that use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated.

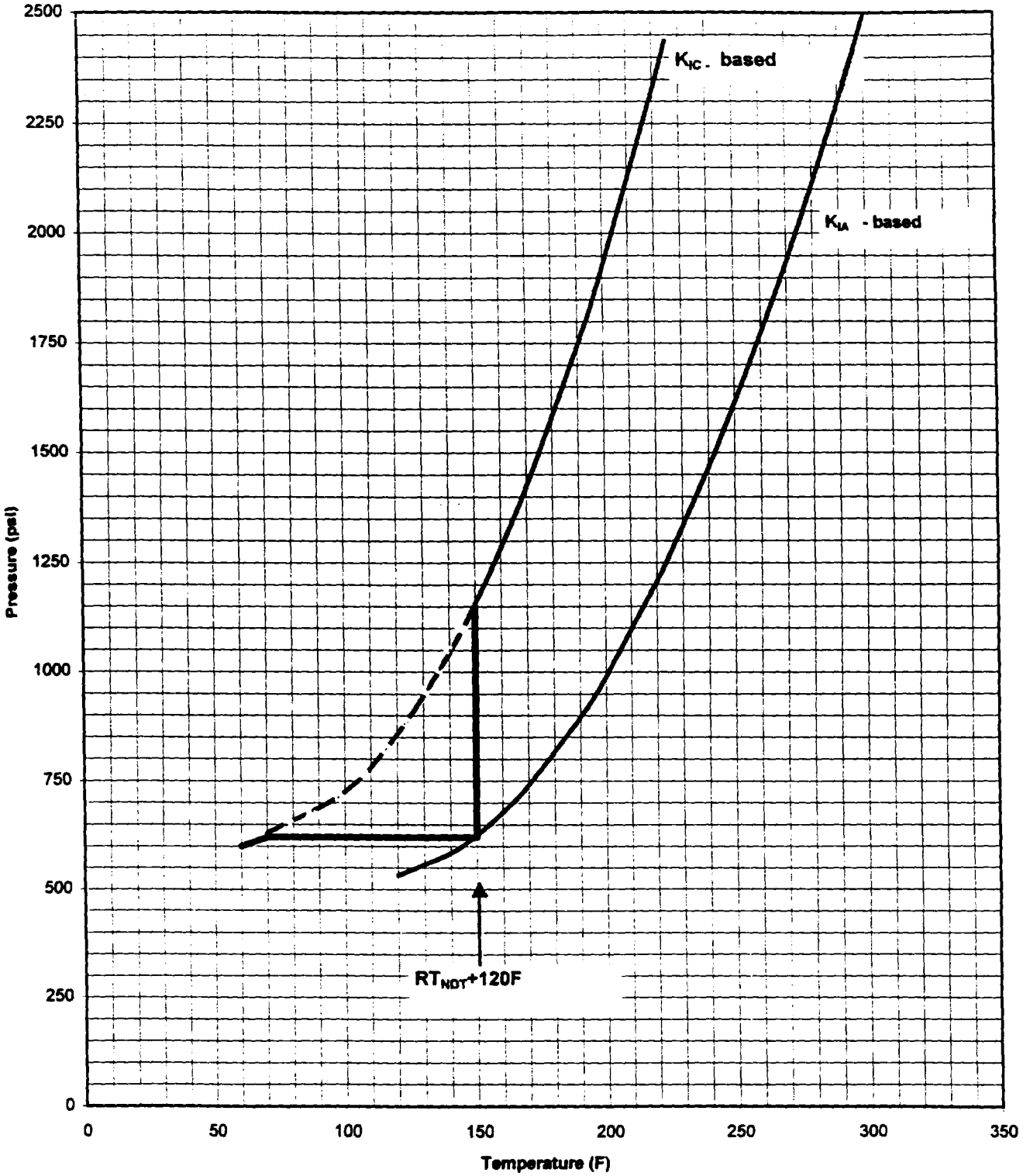


Figure 1-1 Illustration of the Impact of the Flange Requirement for a Typical PWR Plant

2.0 TECHNICAL APPROACH

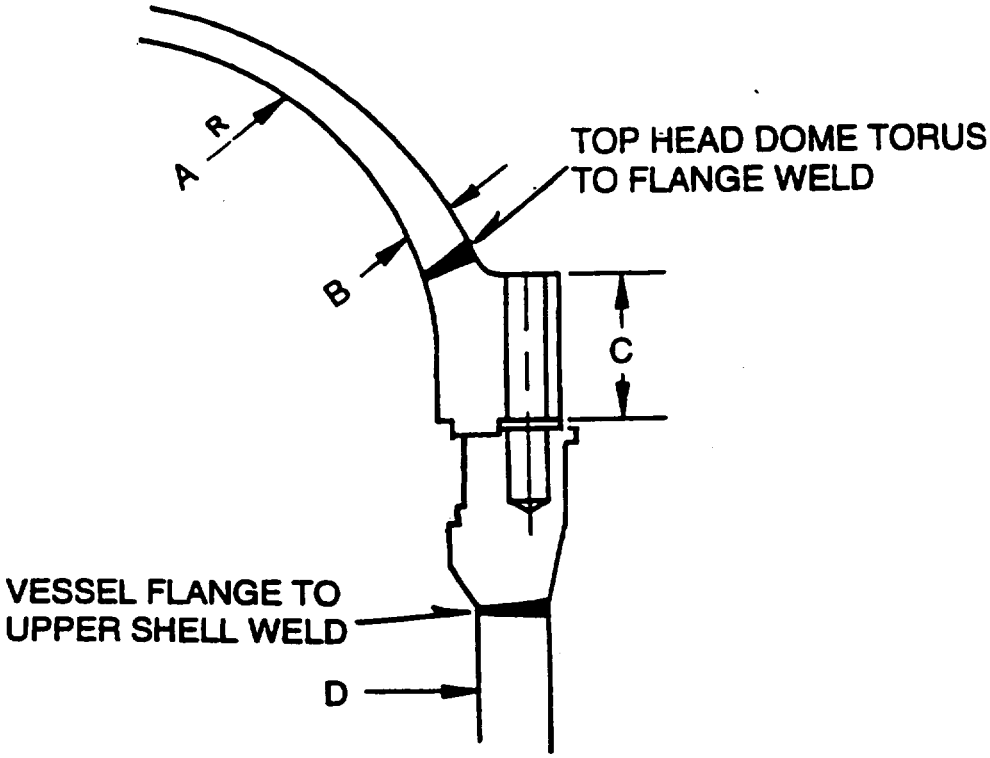
The evaluation to be presented here is intended to cover all operating light water reactor vessels. Fracture evaluations have been performed on the range of geometries which exist, and results will be tabulated and discussed.

The geometry of the closure head region for all the vessels analyzed are shown in Figures 2-1 through 2-4. The geometries for the various PWR vessels are similar, and the same is true for the various BWR vessels. This is also reflected in the stresses, as will be discussed further in Section 4.

Stress analyses have been performed on all of these designs, and these stress results were used to perform fracture mechanics evaluations. The highest stress location in the closure head and vessel flange region is in the head, just above the bolting flange. This corresponds with the location of a weld in nearly all the designs. The highest stressed location is near the outside surface of the head in that region, and so the fracture evaluations have assumed a flaw at this location.

The goal of the evaluation is to compare the integrity of the closure head during the boltup process to the integrity during steady state operation. The question to be addressed is: With the higher K_{Ic} fracture toughness now known to be applicable, is there still a concern about the integrity of the closure head during boltup?

Table 2-1 Geometry Comparison		
Plant Type	**Head Thickness**	**Vessel Diameter**
Westinghouse 2 Loop	5.66"	132.4
3 Loop	5.75	155.5
4 Loop	7.0	178.9
B&W	6.63	168.4
Combustion Engineering	7.4	173.4
GE Design 1 (CE)	3.6	109.5
Design 2 (B&W)	4.0	122.4
Design 3 (CBI)	4.8	124.8



UPPER HEAD REGION

	2 LOOP	3 LOOP	4 LOOP
A	83.46	74.59	85.78
B	5.66	5.75	7.00
C	27.56	29.56	27.25
D	132.40	155.50	170.88

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-1 **Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel**

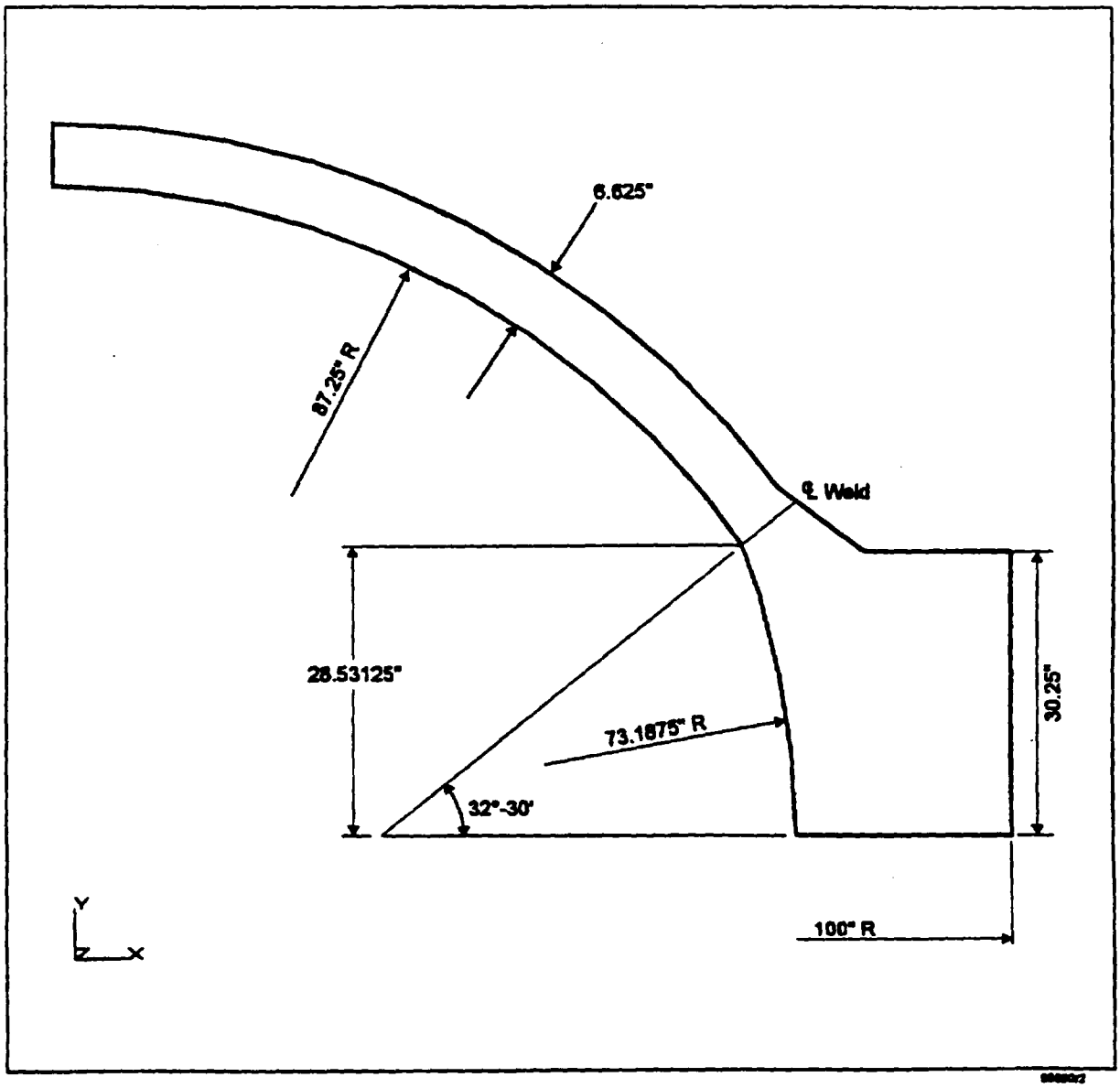


Figure 2-2 Geometry of Closure Head Region – Babcock and Wilcox Reactor Vessels

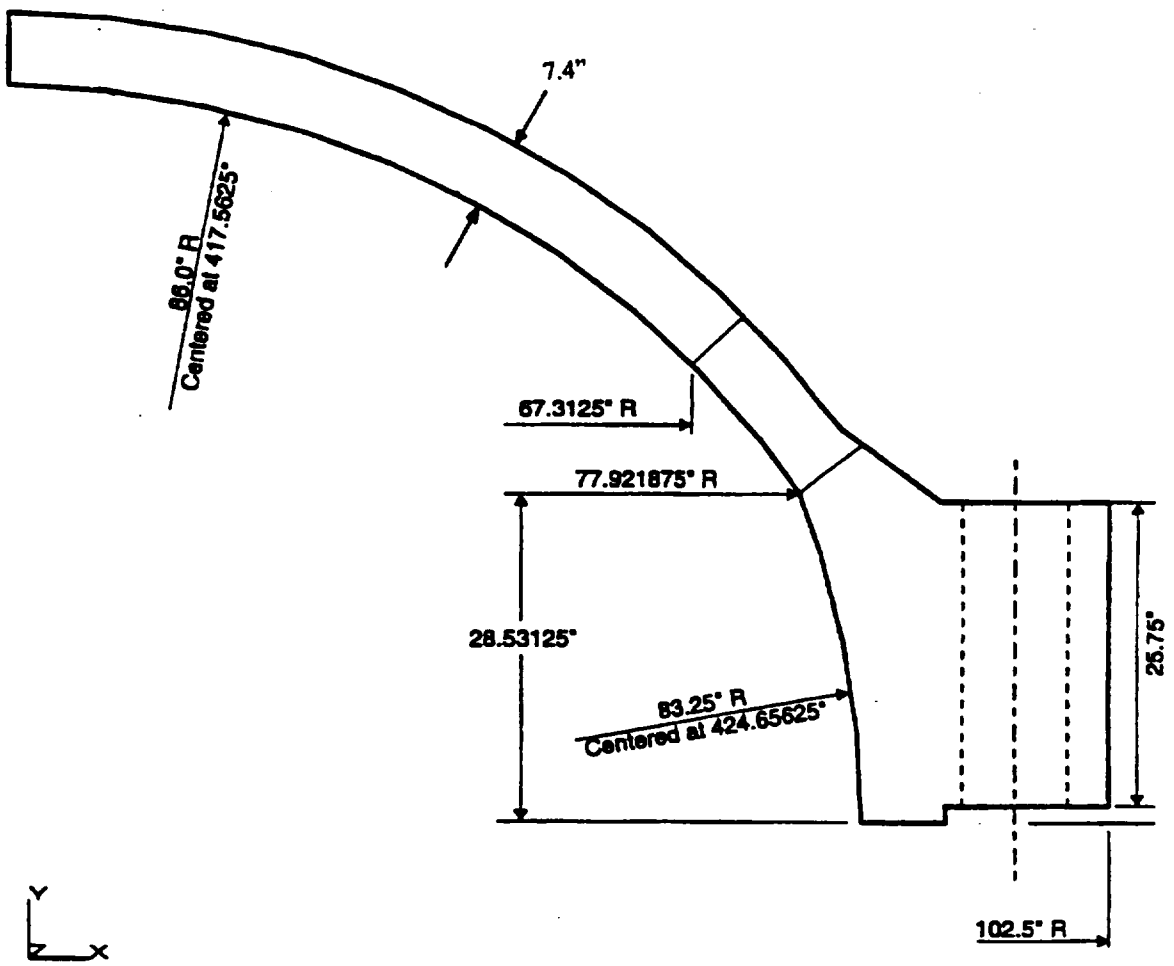
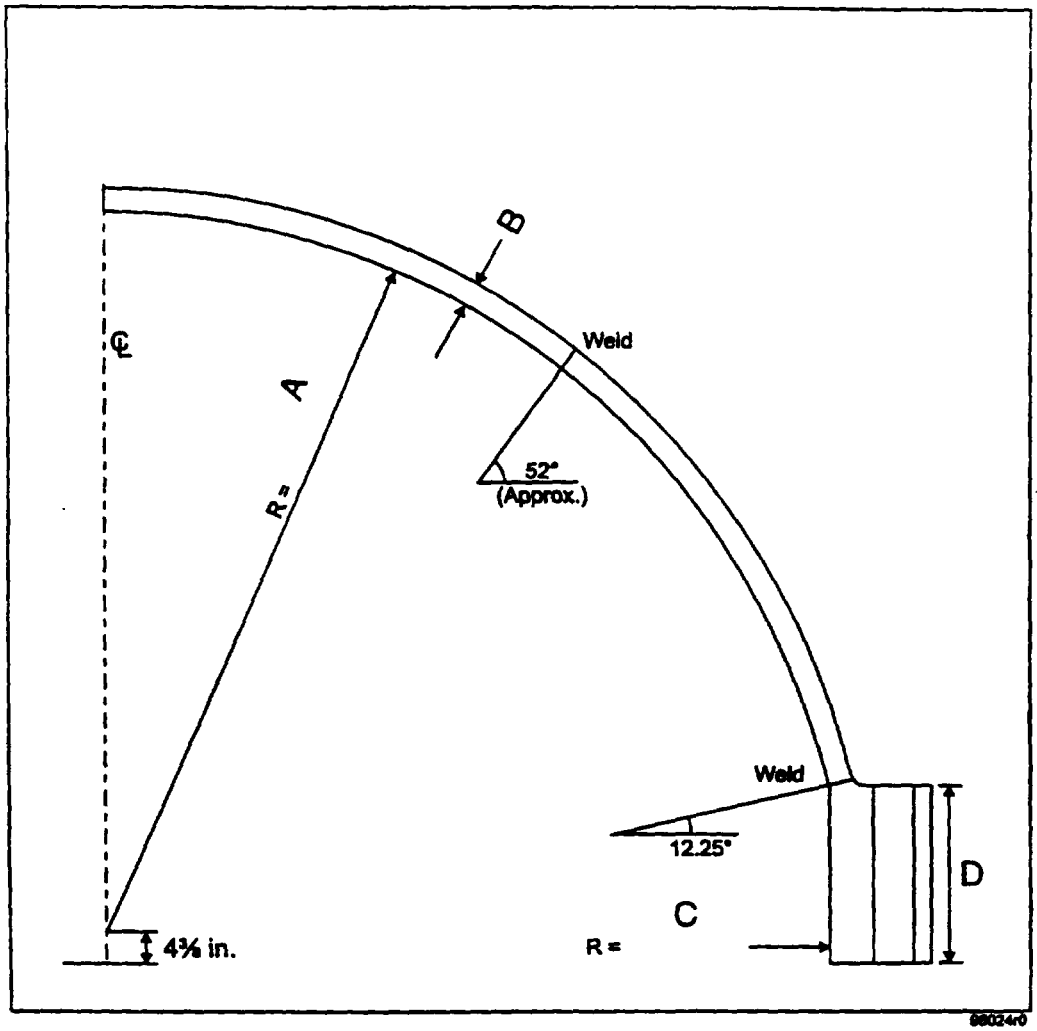


Figure 2-3 Geometry of Closure Head Region: Combustion Engineering Reactor Vessels



	CE	B&W	CB&I
A	109.5	125.6	124.8
B	3.6	4.0	4.8
C	109.5	122.4	124.8
D	24.4	31.0	28.2

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-4 Geometry of Closure Head Region: General Electric Reactor Vessels

3.0 FRACTURE ANALYSIS METHODS AND MATERIAL PROPERTIES

The fracture evaluation was carried out using the approach suggested by Section XI Appendix G.[1] A semi-elliptic surface flaw was postulated to exist in the highest stress region, which is at the outside surface of the closure flange. The flaw depth was set at 25 percent of the wall thickness, and the shape was set at a length six times the depth.

3.1 STRESS INTENSITY FACTOR CALCULATIONS

One of the key elements of a fracture evaluation is the determination of the driving force or stress intensity factor (K_I). This was done using expressions available from the literature. In most cases, the stress intensity factor for the integrity calculations utilized a representation of the actual stress profile rather than a linearization. The stress profile was represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1 \frac{x}{t} + A_2 \left(\frac{x}{t}\right)^2 + A_3 \left(\frac{x}{t}\right)^3 \tag{3-1}$$

where
- x = is the coordinate distance into the wall, in.
- t = wall thickness, in.
- σ = stress perpendicular to the plane of the crack, ksi
- A_i = coefficients of the cubic fit

For the surface flaw with length six times its depth, the stress intensity factor expression of Raju and Newman [2] was used when a complete stress distribution was available. The stress intensity factor $K_I(\phi)$ can be calculated anywhere along the crack front. The point of maximum crack depth is represented by $\phi = 0$, and this location was found to also be the point of maximum K_I for the cases considered here. The following expression is used for calculating $K_I(\phi)$, where ϕ is the angular location around the crack. The units of $K_I(\phi)$ are $\text{ksi}\sqrt{\text{in}}$.

$$K_I(\phi) = \left[\frac{\pi a}{Q}\right]^{0.5} \sum_{j=1}^{4} G_j\left(a/c,\ a/t,\ t/R,\ \phi\right) A_j a^j \tag{3-2}$$

The magnification factors $G_1(\phi)$, $G_2(\phi)$, $G_3(\phi)$ and $G_4(\phi)$ are obtained by the procedure outlined in reference [2]. The dimension "a" is the crack depth, and "c" is the crack length, while t is the wall thickness. In some cases only surface stress values were available, and in these cases the stresses were linearized through the thickness of the head, and the Raju-Newman expression was used.

3.2 FRACTURE TOUGHNESS

The other key element in a fracture evaluation is the fracture toughness of the material. The fracture toughness has been taken directly from the reference curves of Appendix A, Section XI.

In the transition temperature region, these curves can be represented by the following equations:

$$K_{Ic} = 33.2 + 20.734 \exp. [0.02 (T\text{-}RT_{NDT})] \quad (3\text{-}4)$$

$$K_{Ia} = 26.8 + 12.445 \exp. [0.0145 (T\text{-}RT_{NDT})] \quad (3\text{-}5)$$

where K_{Ic} and K_{Ia} are in $ksi\sqrt{in}$.

The upper shelf temperature regime requires utilization of a shelf toughness which is not specified in the ASME Code. A value of 200 $ksi\sqrt{in}$ has been used here for all the regions except the nozzle inner radius regions, since the upper shelf Charpy energy exceeds 50 ft-lb, even after irradiation. This value is consistent with general practice in such evaluations, as shown for example in reference [3], which provides the background and technical basis of Appendix A of Section XI.

The other key element in the determination of the fracture toughness is the value of RT_{NDT}, which is a parameter determined from Charpy V-notch and drop-weight tests.

The value of RT_{NDT} for the closure flange region of operating PWR plants was surveyed for 82 PWR plants world wide, and the average value of RT_{NDT} was found to be 9°F. The results ranged from –50°F to +60°F, with the 60°F cases representing the few cases where a test result was not available or the maximum allowed by the ordering requirement. For the head region of operating BWR plants, results ranged up to 40°F, which was the ordering requirement, while the average value of RT_{NDT} was found to be 10°F. Therefore, the value of 10°F was used for the illustrations to be discussed in Sections 4 and 5.

3.3 IRRADIATION EFFECTS

Neutron irradiation has been shown to produce embrittlement which reduces the toughness properties of reactor vessel steels. The decrease in the toughness properties can be assessed by determining the shift to higher temperatures of the reference nil-ductility transition temperature, RT_{NDT}.

The location of the closure flange region is such that the irradiation levels are very low and therefore the fracture toughness is not measurably affected.

4.0 FLANGE INTEGRITY

The first step in evaluation of the closure head/flange region is to examine the stresses. The stresses which are affected by the boltup event are the axial, or meridional stresses, which are perpendicular to the nominal plane of the closure head to flange weld. The stresses in this region during steady state operation are summarized in Table 4-1.

The table shows that the stresses in the various PWR designs are very similar during steady state operation, and stresses are not very high. The loadings are primarily membrane stress, and the bending stresses are somewhat lower. For the BWR designs, the membrane stress is very similar, as might be expected from use of the same design code. The bending stresses are higher for the BWR designs, due to the larger diameter and smaller thickness.

Table 4-2 provides a comparison of the stresses at boltup with those at steady state. It is easy to see that the stresses at boltup are mostly bending, with a very small membrane stress. As the vessel is pressurized, the membrane stresses increase.

The relative impact of these stresses can best be addressed through a fracture evaluation. A semi-elliptic surface flaw was postulated at the outer surface of the closure head flange, and the stress intensity factor, K, (or crack driving force) was calculated. The results are shown for the boltup condition in Figures 4-1 and 4-2. Figure 4-1 shows the results for the governing PWR design (B&W), while Figure 4-2 shows the results for the governing BWR design (B&W, 251 inches). In both cases it can be seen that the applied stress intensity factor at boltup reaches a maximum for a flaw about half way through the head thickness, and then decreases as the flaw extends into the lower stress region near the inside surface of the head. The maximum value of the stress intensity factor for each of the designs is tabulated in Table 4-3, and plots for each of the other design cases appear in the Appendix.

Also shown in Table 4-3 is the fracture toughness at boltup for typical PWR and BWR plants. The boltup temperature for a PWR is typically 60°F, while the boltup temperature for a BWR is typically 80°F. Since we know that the average value of RT_{NDT} is 10°F for all the plants, both the K_{Ic} and K_{Ia} values are easily calculated.

Study of Table 4-3 shows the difference in the integrity story using the two values of fracture toughness. Using the K_{Ia} toughness (which was the basis for the original flange requirements) it can be seen that the applied stress intensity factor exceeds the toughness for two cases, cases 2 and 6, for flaws about half way through the head thickness.

Using the K_{Ic} toughness, which has now been adopted by Section XI for P-T Curves, it can be seen that there is significant margin between the applied stress intensity factor and the fracture toughness at virtually all crack depths. Another objective of the requirements in Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore it may be concluded that the integrity of the closure head/flange region is not a concern for any of the operating plants using the K_{Ic} toughness.

Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

Table 4-1 Axial Stress Comparison Steady State Operation @ 2250 psi (PWR), 1000 psi (BWR)

Plant	OD Stress (ksi)	Membrane Stress (ksi)	Bending Stress (ksi)
W 4 Loop	22.8	10.0	12.8
W 3 Loop	20.9	11.6	9.3
CE	46.4	12.8	33.6
B&W	55.7	19.0	36.7
GE BWR Design 1 (CE)	49.6	18.0	31.6
GE BWR Design 2 (B&W)	53.0	15.5	37.5
GE BWR Design 3 (CBI)	52.5	14.3	38.2

Table 4-2 Stress Comparison Boltup vs. Steady State

Plant	Boltup Membrane (ksi)	Boltup Bending (ksi)	SS Membrane (ksi)	SS Bending (ksi)
W 4 Loop	1.1	14.2	10.0	12.8
W 3 Loop	2.1	14.5	11.6	9.3
CE	0.8	22.8	12.8	33.6
B&W	4.3	27.6	19.0	36.7
GE BWR Design 1 (CE)	0.8	26.3	18.0	31.6
GE BWR Design 2 (B&W)	0.5	48.5	15.5	37.5
GE BWR Design 3 (CBI)	0.5	35.5	14.3	38.2

Table 4-3 Flange Integrity Results at Boltup

Design	Maximum K, in ksi$\sqrt{in}$ (Flaw Depth/Thickness)	Fracture Toughness at Boltup*	
		K_{Ia} (ksi$\sqrt{in}$)	K_{Ic} (ksi$\sqrt{in}$)
1. CE	41 (.42)	52.7	89.6
2. B&W	56 (.60)	52.7	89.6
3. W Four Loop	31 (.44)	52.7	89.6
4. W Three Loop	32 (.44)	52.7	89.6
5. GE BWR (CBI 251)	56 (.42)	61.4	117.3
6. GE BWR (B&W 251)	69 (.40)	61.4	117.3
7. GE BWR (CE 218)	37 (.42)	61.4	117.3

*Boltup is typically at 60°F for PWRs, and 80°F for BWRs.

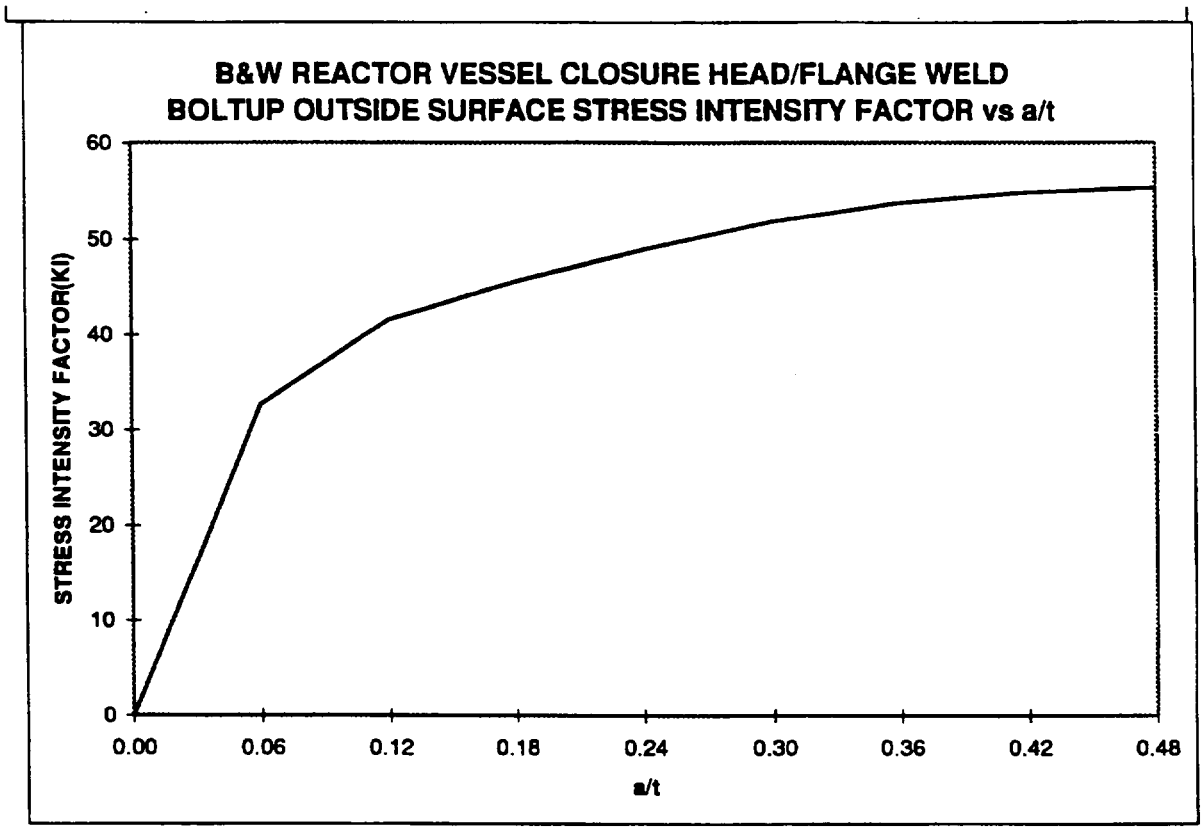


Figure 4-1. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Governing PWR Design

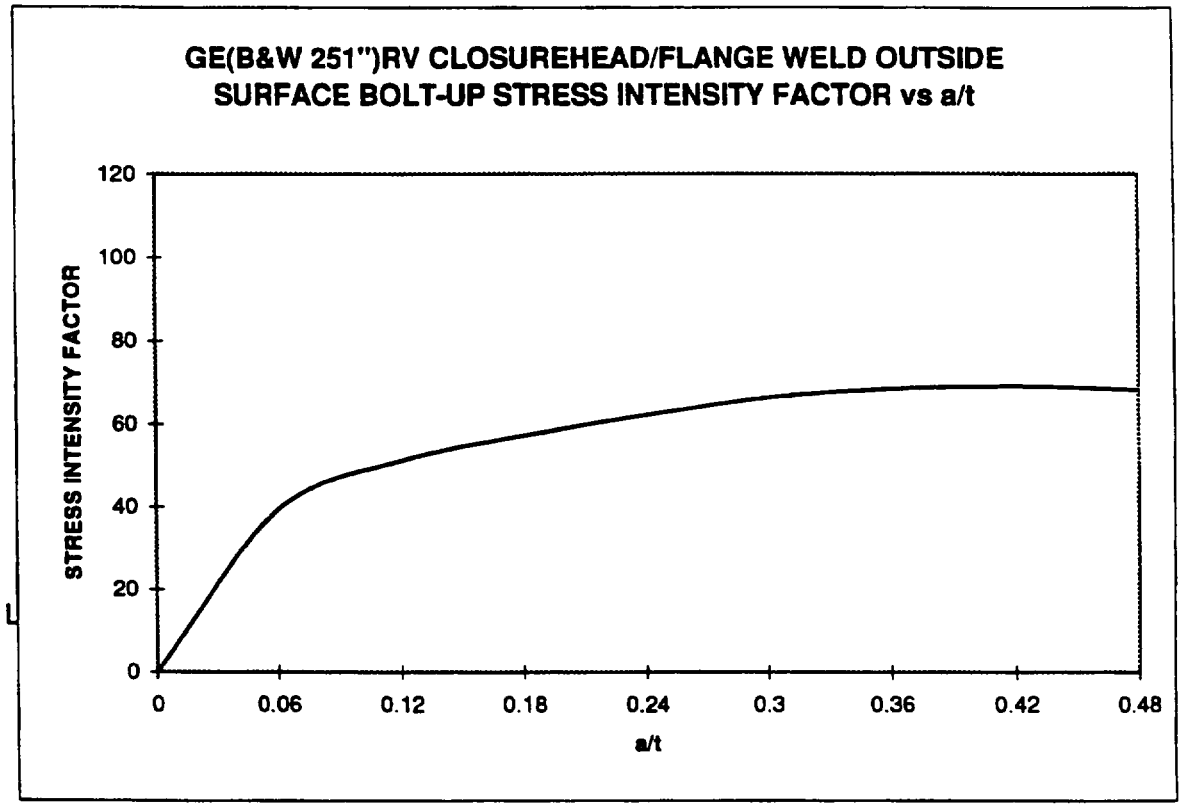


Figure 4-2. Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld, for the Governing BWR Design

5.0 ARE FLANGE REQUIREMENTS NECESSARY?

Using the K_{ic} curve can support the elimination of the flange requirement. This can be illustrated by examining the stress intensity factor change for a postulated flaw as the vessel is pressurized after boltup, progressing up to steady state operation.

The stresses at the region of interest are shown in Table 4-1, for steady state operation. Included here are the stresses at the outside surface, which is the highest stress location for this region, as well as the membrane and bending stresses. Table 4-2 shows a comparison of the boltup and steady state stresses for the same plant designs. The results are similar for the designs shown, which bracket all plants in service. No comparisons are available for two loop Westinghouse plants, but they are conservatively covered by the four loop Westinghouse plant results.

As the vessel is pressurized, the stresses in the closure flange region gradually change from mostly bending stresses to a combination of bending and membrane stresses. The stress intensity factor, or driving force, increases for a postulated flaw at the outside surface, as the vessel is pressurized.

As mentioned in Section 4, the boltup temperature for a PWR is nominally 60°F, while that for a BWR is nominally 80°F. From Section 3, the average value of RT_{NDT} for the closure head material is 10°F for all the designs, so boltup is typically at RT_{NDT} + 50 for PWRs, and RT_{NDT} + 70°F for BWRs.

A direct comparison between the original basis for the boltup requirement and the new K_{Ic} approach is provided in Table 5-1. This table provides calculated boltup requirements for all the designs, using a safety factor of 2, and a reference flaw depth of a/t = 0.10, which was used by Randall as the basis for the original requirement [4]. The boltup requirements using K_{Ia} are shown in the right-most column, and the governing case would have a boltup requirement of RT_{NDT} + 118°F, which closely matches the requirement of RT_{NDT} +120°F now in 10CFR50 Appendix G.

Now consider the equivalent result using K_{Ic}, which is just to the left of the column just discussed. The boltup requirement using the same margin now ranges from RT_{NDT} to RT_{NDT} + 41F for PWR plants, and from RT_{NDT} to RT_{NDT} + 56 for BWR plants. Since the average value of RT_{NDT} is 10°F for all the plants, the boltup requirements can be easily translated into actual temperatures. For PWRs the requirement for boltup ranges from 10°F to 51°F, and the actual boltup temperature is 60°F. For BWRs the requirement ranges from 10°F to 66°F, and the actual boltup temperature is 80°F. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement can be eliminated from 10CFR50 Appendix G.

Table 5-1 Comparison of Various Plant Designs Boltup Requirements

Plant	K (a/t = .1)	K with SF = 2	$T - RT_{NTD}$ (°F) using K_{Ic} (a/t = .10)	$T - RT_{NTD}$ (°F) using K_{Ia} (a/t = .10)
CE	30.0	60.0	13	68
B&W	39.4	79.8	41	100
W 4 Loop	19.7	39.4	0	1
W 3 Loop	19.4	38.8	0	0
GE (CBI 251")	38.7	77.4	38	97
GE (B&W 251")	48.0	96.0	56	118
GE (CE 218")	25.1	50.2	0	43

*All units in $ksi\sqrt{in}$

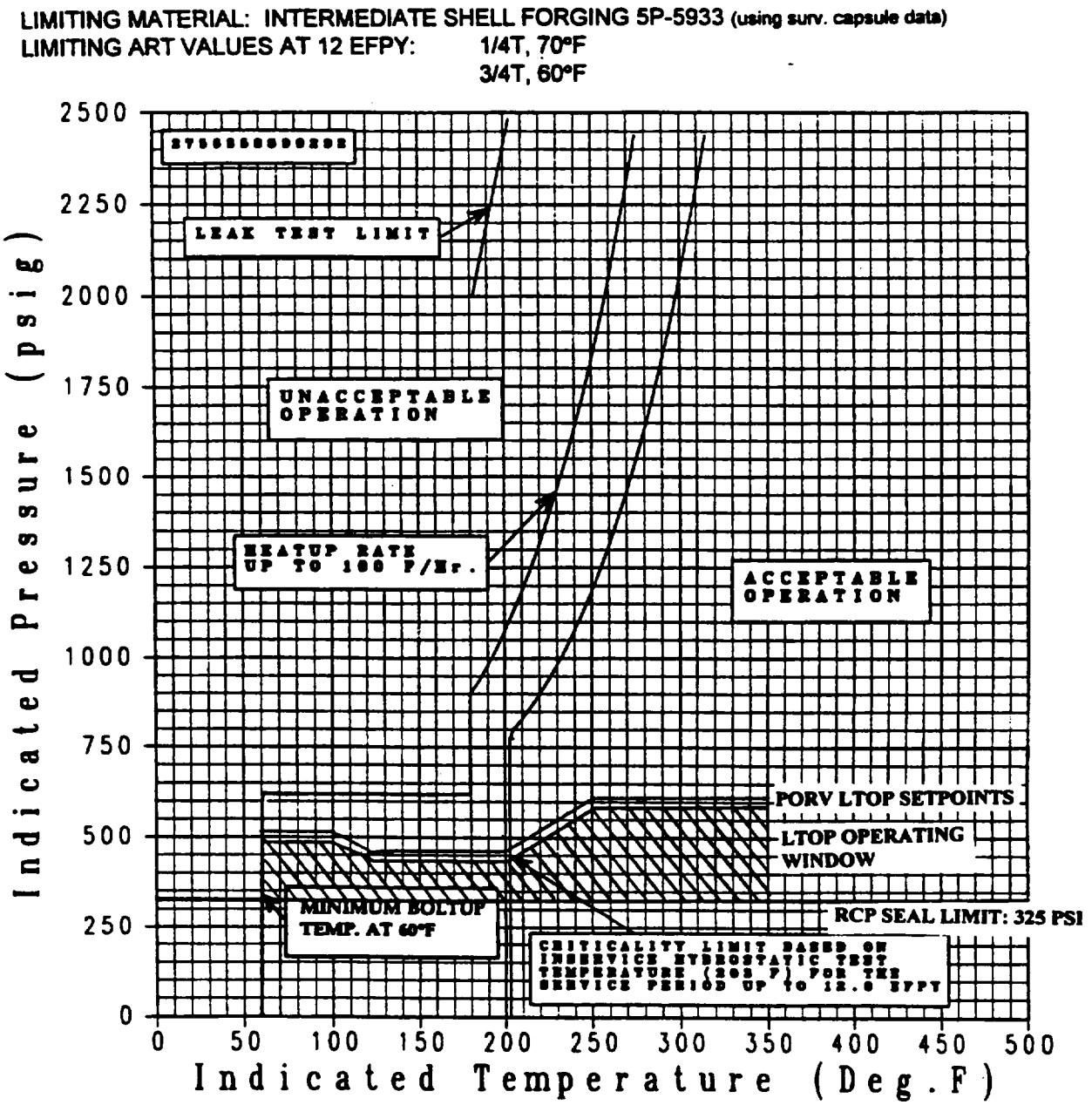


Figure 6-2 **Illustration of the Actual Operating Window for Heatup of Byron Unit 1, a Low Copper Plant at 12 EFPY**

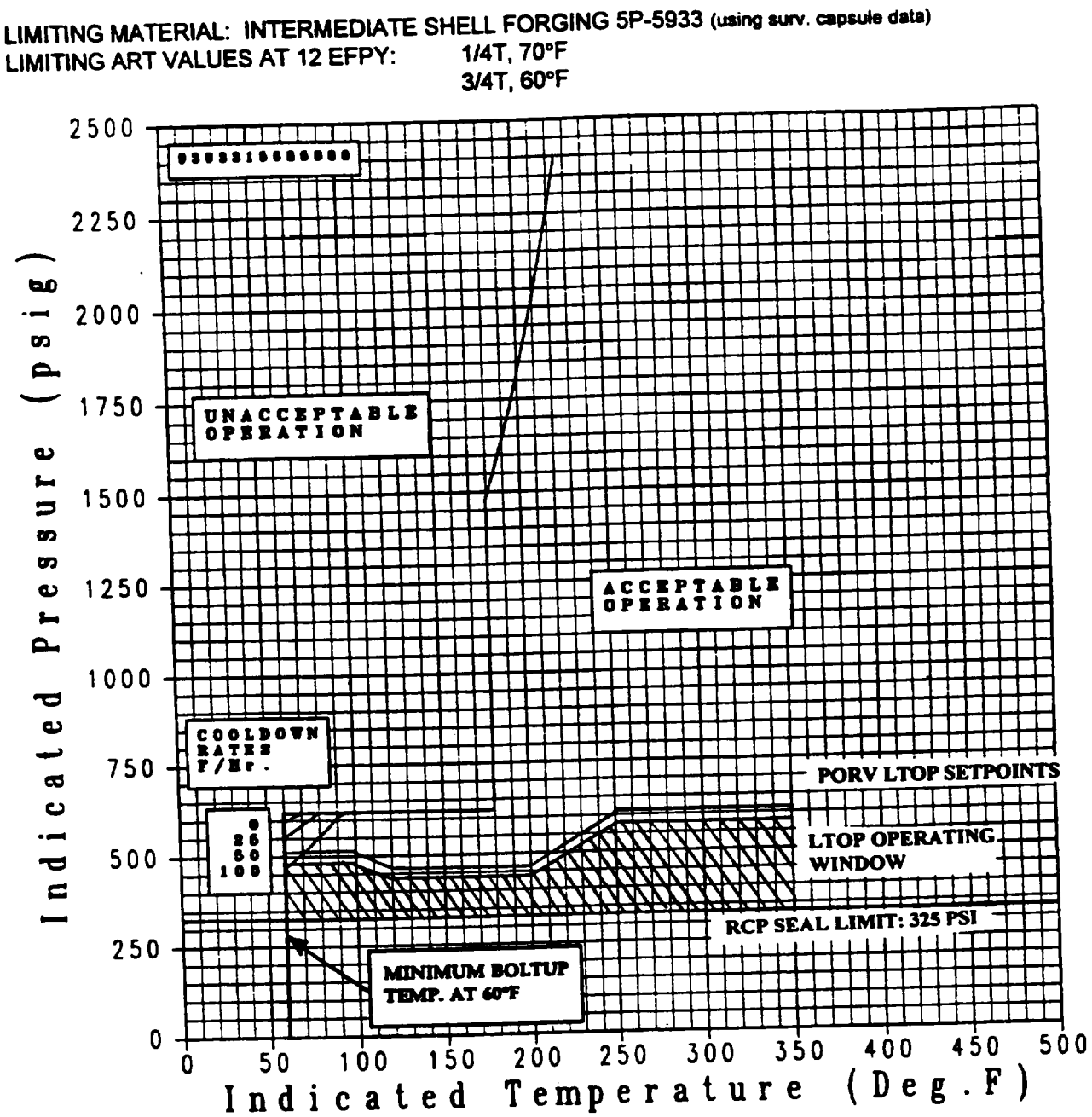


Figure 6-3 Illustration of the Actual Operating Window for Cooldown of Byron Unit 1, a Low Copper Plant at 12 EFPY

6.0 SAFETY IMPLICATIONS OF THE FLANGE REQUIREMENT

There are important safety implications which are associated with the flange requirement, as illustrated by Figure 6-1. The safety concern is the narrow operating window at low temperatures forced by the flange requirement. The flange requirement sets a pressure limit of 621 psi for a PWR (20 percent of hydrotest pressure). Thus, no matter how good the toughness of the vessel, the P-T limit curve may be superceded by the flange requirement for temperatures below RT_{NDT} + 120°F. This requirement was originally imposed to ensure the integrity of the flange region during boltup, but Section 4 has shown that this is no longer a concern.

The flange requirement can cause severe operational limitations when instrument uncertainties are added to the lower limit (621 psi), for the Low Temperature Overpressure Protection system of PWRs. The minimum pressure required to cool the seals of the main coolant pumps is 325 psi, so the operating window sometimes becomes very small, as shown schematically in Figure 6-1. If the operator allows the pressure to drop below the pump seal limit, the seals could fail, causing the equivalent of a small break LOCA, a significant safety problem. Elimination of the flange requirement will significantly widen the operating window for most PWRs.

An example will be provided to illustrate this situation for an operating PWR plant, Byron Unit 1. This is a forging-limited vessel at 12 EFPY, with a low leakage core, and low copper weld material in the core region. The vessel has excellent fracture toughness, which means that the flange notch is very prominent, as shown in the vessel heatup curve of Figure 6-2. As illustrated before in Figure 6-1, Byron has the LTOP setpoints significantly below the flange requirement of 621 psi, because of a relatively large instrument uncertainty. The setpoints of the two power operated relief valves are staggered by about 16 psi to prevent a simultaneous activation. The two PORVs have different instrument uncertainties, and for conservatism the higher uncertainty is used. A similar situation exists for cooldown, as shown in Figure 6-3.

Elimination of the flange requirement for Byron Unit 1 would mean that the PORV curve could become level at 604/587 psig, which are the leading/trailing setpoints to protect the PORV downstream piping, through the temperature range of the 350°F down to boltup at 60°F. The operating window between the leading PORV and the pump seal limit rises from 121 psig (446-325) to 262 psig (587-325). This change will make a significant improvement in plant safety by reducing the probability of a small LOCA, and easing the burden on the operators.

This is only one example of the impact of the flange requirement. Every operating PWR plant will have a different situation, but the operational safety level will certainly be generally improved by the elimination of this unnecessary requirement.

Elimination of the flange requirement has no impact on BWRs. The saturation temperature corresponding to the 300 psig operating pressure (20% of the pre-service hydrostatic test pressure) is 420°F. This is well in excess of the RT_{ndt} + 120°F requirement. Therefore the flange temperature requirements are satisfied regardless of whether they exist or not. Therefore, elimination of the flange temperature requirement has no impact on BWR flange integrity.

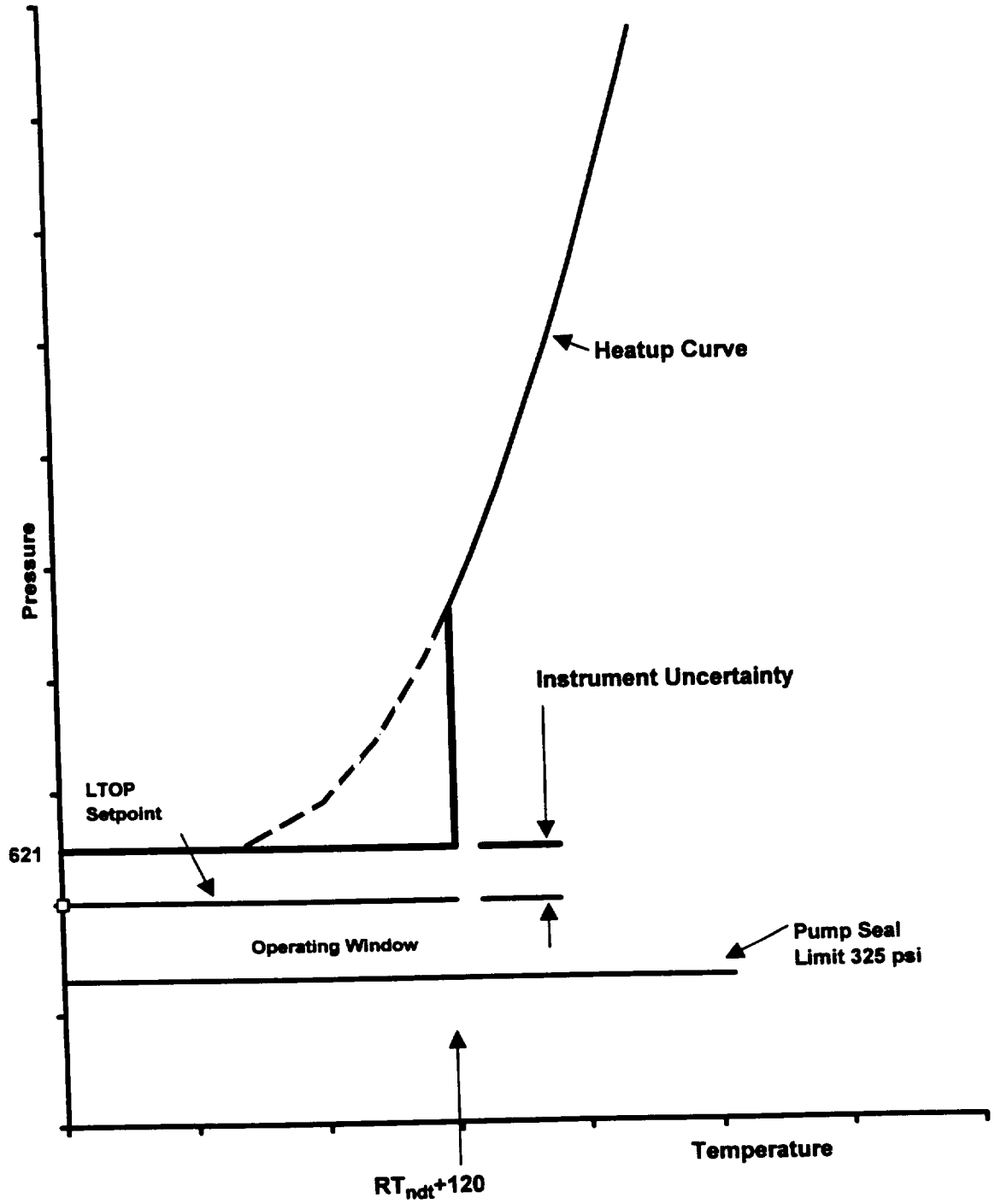


Figure 6-1 Illustration of the Flange Requirement and its Effect on the Operating Window for a Typical Heatup Curve

7.0 REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1996 Addenda, ASME, New York.

2. Newman, J. C. Jr. and Raju, I. S., "Stress Intensity Factors for Internal Surface Cracks in Cylindrical Pressure Vessels," Trans. ASME, Journal of Pressure Vessel Technology, Vol. 102, 1980, pp 342-346.

3. Marston, T. U., ed., "Flaw Evaluation Procedures: ASME Section XI," Electric Power Research Institute Report EPRI-NP-719 SR, August 1978.

4. Randall, N., Abstract of Comments and Staff Response to Proposed Revision to 10 CFR Part 50, Appendices G and H, Published for Comment in the Federal Register, November 14, 1980.

Appendix 1

Stress Intensity Factor Curves for the Boltup Condition

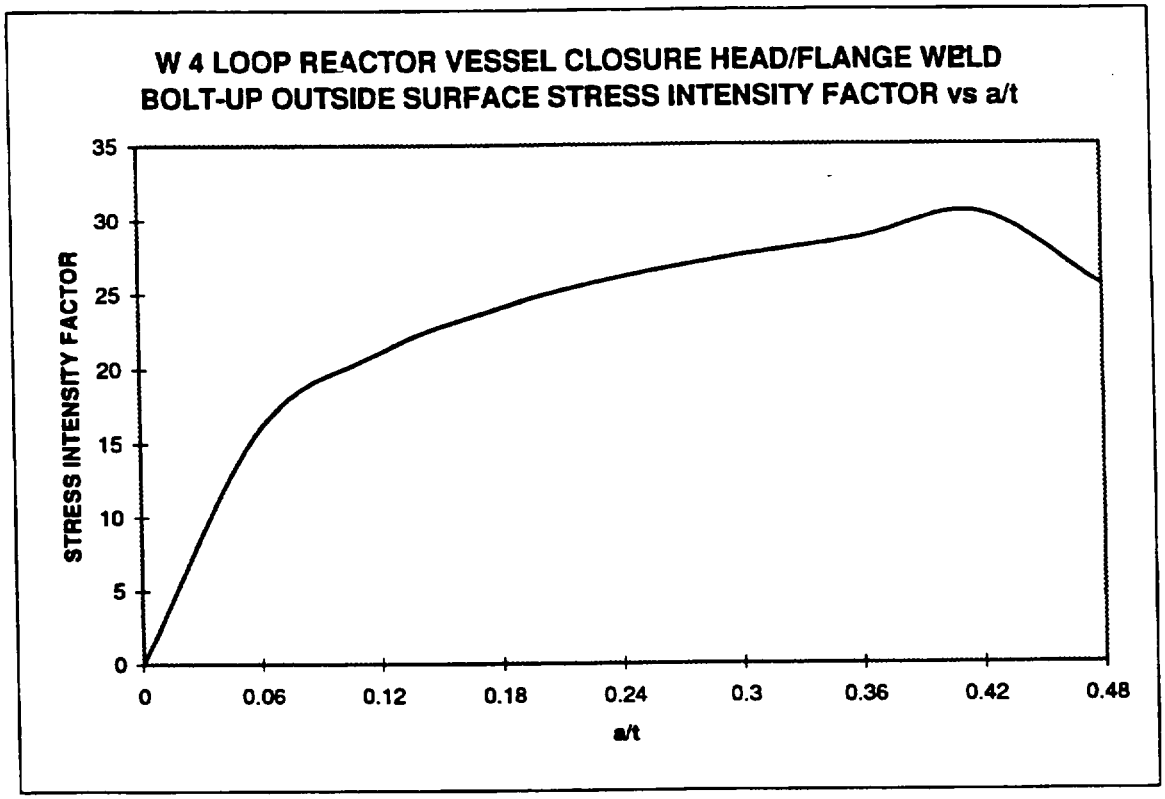


Figure A-1. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Westinghouse Four Loop Plant Design**

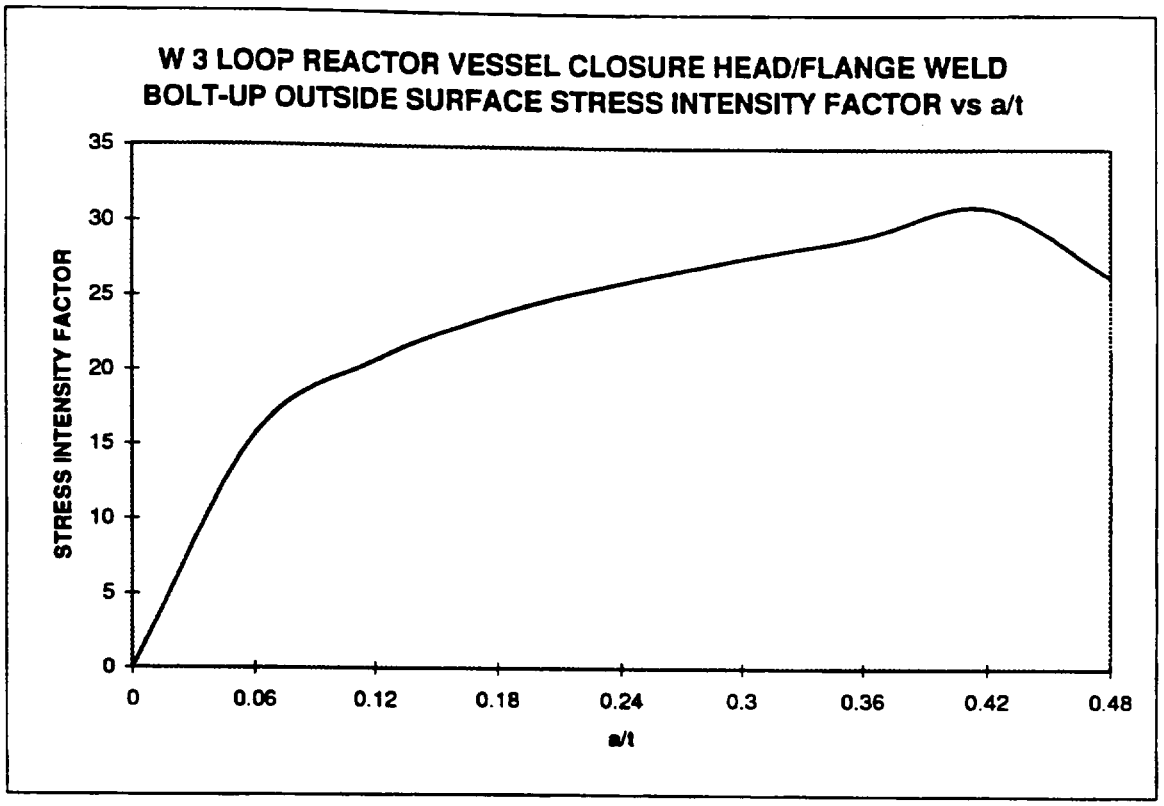


Figure A-2. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Westinghouse Three Loop Plant Design**

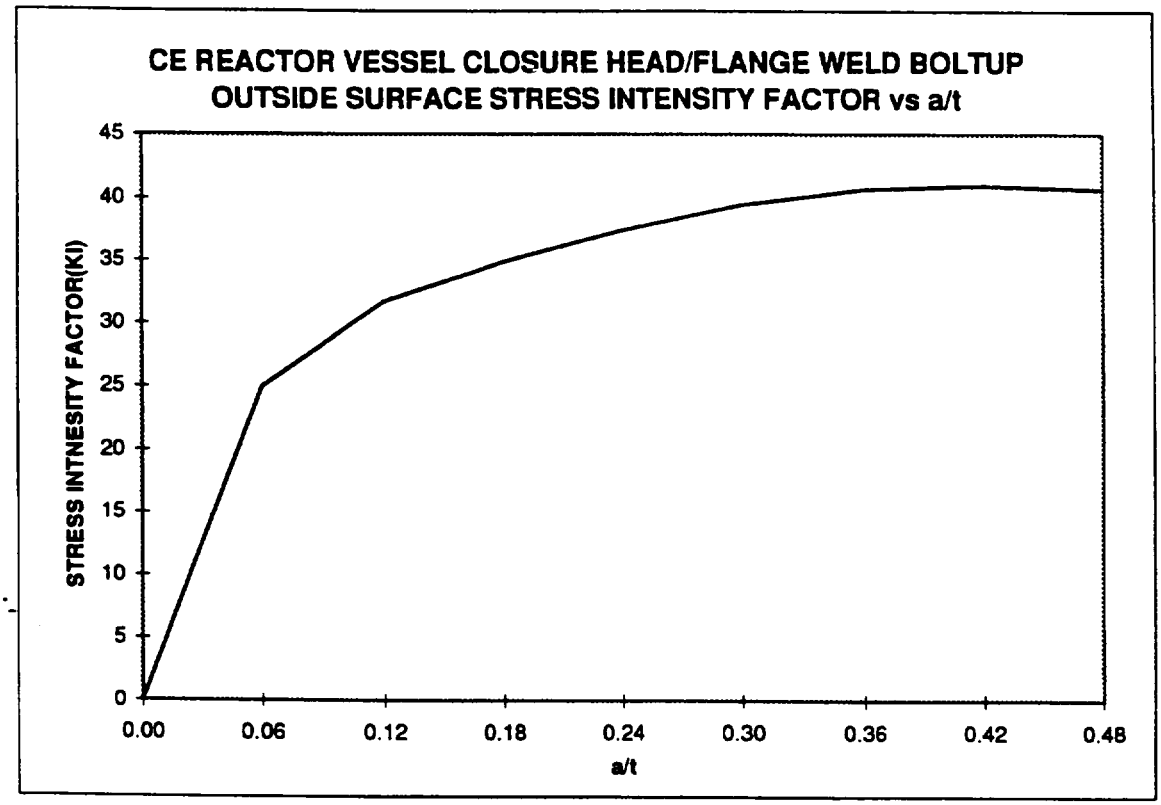


Figure A-3. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the Combustion Engineering Design**

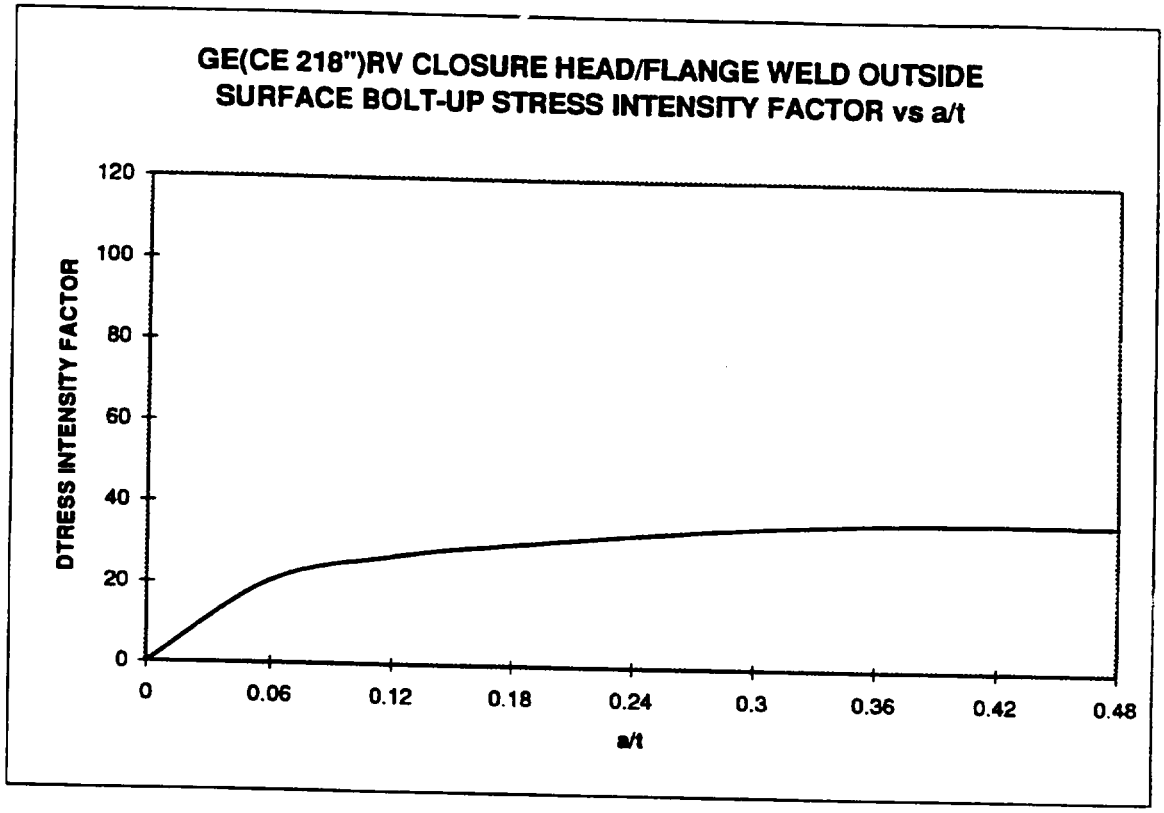


Figure A-4. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the General Electric – CE Fabricated Design**

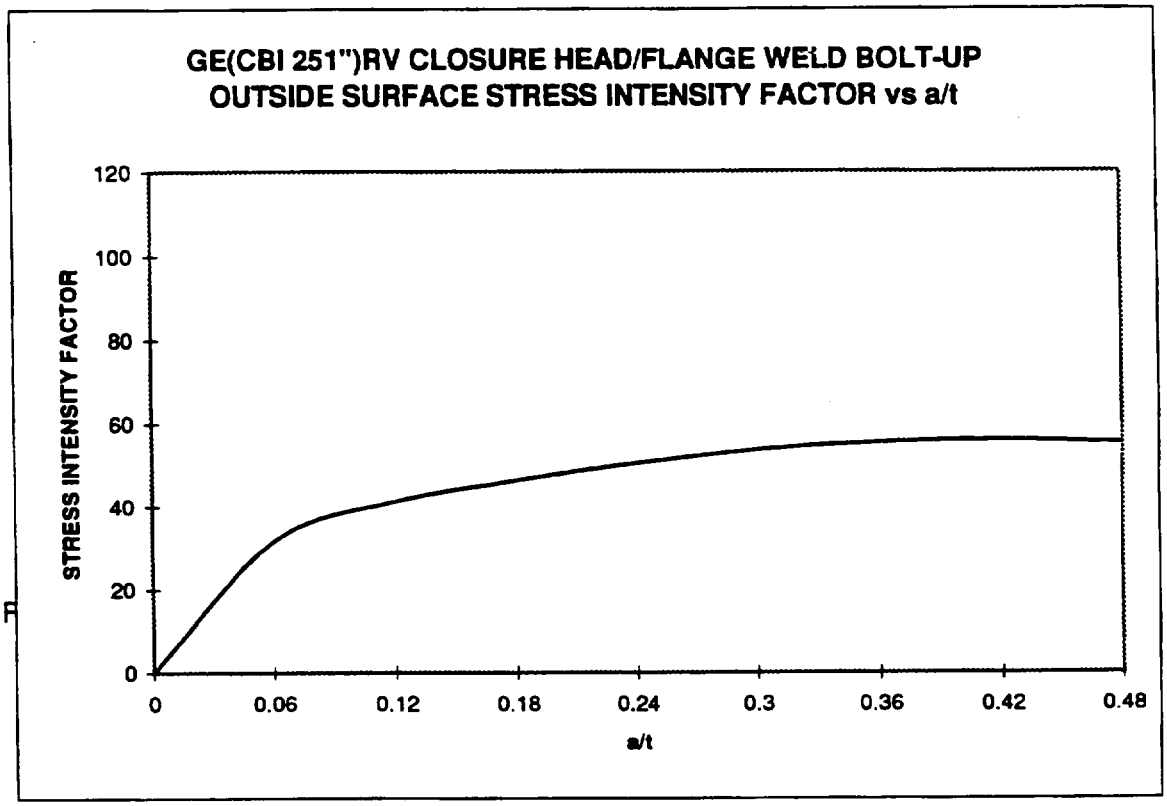


Figure A-5. **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld for the General Electric – CBI Fabricated Design**