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RS-14-284 10 CFR 50.90

October 16, 2014

U. S. Nuclear Regulatory Commission
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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: License Amendment Request to Utilize WCAP-16143-P, Revision 1, as an Analytical Method to Determine the Reactor Coolant System Pressure and Temperature Limits

References:

(1) Letter from K. R. Jury (Exelon Generation Company, LLC) to NRC, "License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limits Report and Request for Exemption from 10 CFR 50.60, 'Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,'" dated October 3, 2005

(2) Letter from R. F. Kuntz (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Exemption From the Requirements of 10 CFR 50, Appendix G," dated November 22, 2006

(3) Letter from R. F. Kuntz (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of amendments Re: Reactor Coolant System Pressure and Temperature Limits Report," dated November 27, 2006

(4) Letter from D. E. Hills (NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Notice of Violation and Braidwood Station, Units 1 and 2, NRC Baseline Inservice Inspection Report 05000456/2013008; 05000457/2013008," dated November 14, 2013

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(5) Letter from E. R. Duncan (NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2, NRC Integrated Inspection Report 05000454/2013005; 05000455/2013005," dated February 3, 2014

(6) Letter from M. E. Kanavos (Exelon Generation Company, LLC) to NRC, "Reply to a Notice of Violation, EA-13-209," dated December 13, 2013

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. This amendment request proposes to utilize WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated October 2014, as an analytical method to determine the reactor coolant system pressure and temperature limits.

In Reference 1, EGC submitted a Request for Exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." This Exemption Request proposed to use WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated November 2003 (i.e., Revision 0), for calculating the reactor pressure vessel (RPV) pressure-temperature (P-T) limits for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, in lieu of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.c as required by 10 CFR 50 .60(a). Reference 1 also requested a corresponding Technical Specifications (TS) change to add WCAP-16143 as a reference in TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." The NRC subsequently approved the Request for Exemption and the associated license amendment in References 2 and 3, respectively.

During an NRC Baseline Inservice Inspection at Braidwood Station (Reference 4), conducted from September 9, 2013, to October 29, 2013, the NRC issued a Severity Level IV Violation for failure to provide information to the NRC that was complete and accurate in all material respects. Specifically, WCAP-16143, Section 4, "Flange Integrity," demonstrated adequate RPV margins based upon the original closure head flange configuration of 54 RPV head studs and did not represent the modified closure head configuration of 53 RPV head studs applicable to the Unit 2 RPV (i.e., one RPV head stud was removed from service in 1994). Consequently, operation of the Braidwood Station Unit 2 RPV with 53 RPV head studs was not within the bounds and limitations of what the NRC had reviewed in Reference 1. Braidwood Station committed to submitting a revised WCAP to the NRC for review as documented in Reference 6.

Similarly, during an NRC Integrated Inspection at Byron Station (Reference 5), conducted from October 1, 2013, to December 31, 2013, the NRC issued a Green Finding with associated non-cited violation (NCV) of TS 5.6.6, "Reactor Coolant System Pressure and Temperature Limits Report," for failing to maintain the analytical basis for deriving the P-T limit curves consistent with the Unit 2 RPV head stud configuration. Specifically, the analytical model used in WCAP-16143 was based on the original closure head configuration of 54 RPV head studs and did not represent the modified closure head configuration of 53 RPV head studs applicable to the Unit 2 RPV (i.e., one RPV head stud was removed from service in 2010 under the 10 CFR 50.59, "Changes, tests, and experiments," process). The NRC concluded that EGC made a nonconservative assumption that the 10 CFR 50.59 process could be applied to authorize a change in the WCAP-16143 analysis, and therefore, in error, did not seek prior NRC approval.

In response to this concern, each station performed a prompt Operability Assessment and a follow-up Operability Evaluation in accordance with station procedures. It was concluded that the Unit 2 reactor vessel at each station remains operable.

As a result of these violations, WCAP-16143 has been revised to include an analysis of the RPV with 53 head studs. Specifically, WCAP-16143-P, Revision 1 (provided in Attachment 4), addresses the effect of the missing RPV head stud on the technical basis for elimination of the 10 CFR 50, Appendix G fracture toughness requirements. In summary, the stress analysis and fracture mechanics evaluation for the "missing head stud" case determined that, for the boltup condition, the "all studs intact" case is more limiting. The results of the missing head stud evaluation remain in agreement with the conclusions of WCAP-16143, Revision 0, submitted to the NRC in Reference 1. It should be noted that WCAP-16143, Revision 1, addresses both the originally designed 54 RPV head stud configuration and the 53 RPV head stud configuration for all Braidwood and Byron units. EGC requests that the NRC review and approve WCAP-16143-P, Revision 1, which will be utilized as a TS referenced Topical Report to prepare the PTLR.

The attached request is subdivided as follows:

- Attachment 1 provides a description and evaluation of the proposed changes.
- Attachment 2A provides the markup of the affected Braidwood Station TS page.
- Attachment 2B provides the markup of the affected Byron Station TS page.
- Attachment 3 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-14-4042, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.
- Attachment 4 provides WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014 (Proprietary)
- Attachment 5 provides WCAP-16143-NP, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014 (Non-Proprietary)

Attachment 4 contains information proprietary to Westinghouse Electric Company, LLC, and is therefore supported by an affidavit (i.e., Attachment 3) signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure. A non-proprietary version of this information is provided in Attachment 5.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-14-4042 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

The proposed amendment has been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State of Illinois official.

EGC requests approval of the proposed license amendment within one year of this submittal date; i.e., by October 16, 2015. Once approved, the amendment will implemented within 60 days.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Joseph A. Bauer at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of October 2014.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2A. Markup of Technical Specifications Page – Braidwood Station
2B. Markup of Technical Specifications Page – Byron Station
3. Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-14-4042, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice
4. WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014 (Proprietary)
5. WCAP-16143-NP, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014 (Non-Proprietary)

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Changes

Subject: License Amendment Request to Utilize WCAP-16143-P, Revision 1, as an Analytical Method to Determine the Reactor Coolant System Pressure and Temperature Limits

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. This amendment request proposes to utilize WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated October 2014, as an analytical method to determine the reactor coolant system pressure and temperature limits.

In Reference 1, EGC submitted a Request for Exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." This Exemption Request proposed to use WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated November 2003 (i.e., Revision 0), for calculating the reactor pressure vessel (RPV) pressure-temperature (P-T) limits for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, in lieu of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.c as required by 10 CFR 50 .60(a). Reference 1 also requested a corresponding Technical Specifications (TS) change to add WCAP-16143 as a reference in TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." The NRC subsequently approved the Request for Exemption and the associated license amendment in References 2 and 3, respectively.

During an NRC Baseline Inservice Inspection at Braidwood Station (Reference 4), conducted from September 9, 2013, to October 29, 2013, the NRC issued a Severity Level IV Violation for failure to provide information to the NRC that was complete and accurate in all material respects. Specifically, WCAP-16143, Section 4, "Flange Integrity," demonstrated adequate RPV margins based upon the original closure head flange configuration of 54 RPV head studs and did not represent the modified closure head configuration of 53 RPV head studs applicable to the Unit 2 RPV (i.e., one RPV head stud was removed from service in 1994). Consequently, operation of the Braidwood Station Unit 2 RPV with 53 RPV head studs was not within the bounds and limitations of what the NRC had reviewed in Reference 1. Braidwood Station committed to submitting a revised WCAP to the NRC for review as documented in Reference 6.

Similarly, during an NRC Integrated Inspection at Byron Station (Reference 5), conducted from October 1, 2013, to December 31, 2013, the NRC issued a Green Finding with associated non-cited violation (NCV) of TS 5.6.6, "Reactor Coolant System Pressure and Temperature Limits Report," for failing to maintain the analytical basis for deriving the P-T limit curves consistent with the Unit 2 RPV head stud configuration. Specifically, the analytical model used in WCAP-16143 was based on the original closure head configuration of 54 RPV head studs and did not represent the modified closure head configuration of 53 RPV head studs applicable to the Unit 2 RPV (i.e., one RPV head stud was removed from service in 2010 under the 10 CFR 50.59, "Changes, tests, and experiments," process). The NRC concluded that EGC made a nonconservative assumption that the 10 CFR 50.59 process could be applied to authorize a change in the WCAP-16143 analysis, and therefore, in error, did not seek prior NRC approval.

In response to this concern, each station performed a prompt Operability Assessment and a follow-up Operability Evaluation in accordance with station procedures. It was concluded that the Unit 2 reactor vessel at each station remains operable.

As a result of these violations, WCAP-16143 has been revised to include an analysis of the RPV with 53 head studs. Specifically, WCAP-16143-P, Revision 1 (provided in Attachment 4), addresses the effect of the missing RPV head stud on the technical basis for elimination of the 10 CFR 50, Appendix G fracture toughness requirements. In summary, the stress analysis and fracture mechanics evaluation for the "missing head stud" case determined that, for the boltup condition, the "all studs intact" case is more limiting. The results of the missing head stud evaluation remain in agreement with the conclusions of WCAP-16143, Revision 0, submitted to the NRC in Reference 1. It should be noted that WCAP-16143, Revision 1, addresses both the originally designed 54 RPV head stud configuration and the 53 RPV head stud configuration for all Braidwood and Byron units.

2.0 DETAILED DESCRIPTION

WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014, presents a methodology for calculating the RPV P-T limits for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, in lieu of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.c as required by 10 CFR 50 .60(a). EGC requests that the NRC review and approve WCAP-16143-P, Revision 1, which will be utilized as a TS referenced Topical Report to prepare the PTLR. Additional discussion regarding the contents of WCAP-16143-P, Revision 1, are presented in Section 3.0, "Technical Evaluation," and in WCAP-16143-P, Revision 1, presented in Attachment 4.

The proposed TS changes supporting use of WCAP-16143-P, Revision 1, entail changes to TS Table 1.1-1, "MODES," footnotes (b) and (c), as shown below.

<u>Table 1.1-1, "MODES"</u>

Table 1.1-1 defines the criteria for MODES 1 through 6. MODE 4, "Hot Shutdown," and MODE 5, "Cold Shutdown," are annotated with footnote (b) which currently states:

> *(b) All reactor vessel head closure bolts fully tensioned.*

The proposed change would revise this footnote to state:

> *(b) All required reactor vessel head closure bolts fully tensioned.*

In addition, MODE 6, "Refueling," is annotated with footnote (c) which currently states:

> *(c) One or more reactor vessel head closure bolts less than fully tensioned.*

The proposed change would revise this footnote to state:

> (c) *One or more required reactor vessel head closure bolts less than fully tensioned.*

The addition of the word "required" will avoid any confusion regarding the state of a reactor head closure bolt that may be out of service as is the case for Braidwood Station, Unit 2, and Byron Station, Unit 2.

It should be noted that WCAP-16143 (with no revision number) is currently listed as a reference in TS 5.6.6.b.3. TS 5.6.6.b.4 states:

> *"The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date and any supplements)."*

This provision was added to the Braidwood Station and Byron Station TS in Amendments 142 and 148, respectively, as documented in Reference 3. Therefore, upon NRC approval of WCAP-16143, Revision 1, no change is required to TS 5.6.6.b.3; however, the PTLR will be updated to reflect the revision number and report date.

The marked-up TS pages showing the proposed changes are provided in Attachment 2A and 2B for Braidwood Station and Byron Station respectively.

Note that Byron Station, Unit 2, currently has one RPV head stud out of service; however, EGC has plans to restore the RPV head studs back to the original 54-stud configuration during the Unit 2 Fall 2014 refueling outage.

3.0 TECHNICAL EVALUATION

WCAP-16143, Revision 1, Summary

Justification for elimination of the 10 CFR 50, Appendix G flange requirement from the Braidwood Station and Byron Station, Units 1 and 2, P-T limit curves was previously provided in WCAP-16143, Revision 0. The Braidwood Station and Byron Station RPV closure heads were designed to use 54 RPV head studs to couple the closure head and vessel flanges; and seal the main RPV closure joint. Currently 53 RPV head studs are in-service in Braidwood Station, Unit 2, and Byron Station, Unit 2 (as noted above, Byron Station, Unit 2, has plans to restore the reactor head closure bolts back to the original 54-stud configuration during the Fall 2014 refueling outage); thus, the load previously carried by the out-of-service RPV head stud is distributed throughout the surrounding studs. The evaluation performed in Revision 1 of WCAP-16143 (Attachment 4) determined that the flange requirements (see Figure 1-1 of WCAP-16143) remain eliminated from the P-T limit curves for Braidwood Station, Unit 2, and Byron Station, Unit 2, with 53 in-service RPV head studs. The elimination of the Appendix G flange requirements from Braidwood Station, Unit 1, and Byron Station, Unit 1, (each with 54 in-service RPV head studs) P-T limit curves is also reassessed based on stresses resulting from updated finite element stress analysis models. Results are summarized in WCAP-16143, Section 5, "Are Flange Requirements Necessary," and are fully discussed in Appendix E, "Reevaluation of Byron and Braidwood Reactor Vessel Closure Head/Vessel Flange

Requirements to Account for a Missing Closure Stud," and Appendix F, "Stress Distributions in the Closure Head Region for Plant Specific Three-Dimensional Finite Element Models," of the WCAP.

The evaluation performed in Revision 1 determined that the elimination of the flange requirements from the P-T limit curves for Braidwood Station, Unit 2, and Byron Station, Unit 2, remains applicable for operation with 53 in-service RPV head studs. A brief conclusion is given here with the complete discussion provided in Attachment 4, Appendix E.

Two separate plant specific three-dimensional finite element models (FEMs) were developed to assess the impact of the missing RPV head stud on the elimination of the Appendix G flange requirements from the P-T limit curves. A complete discussion of the FEMs and stress analysis are provided in Attachment 4, Appendix F.

First, a baseline FEM was created with all RPV head studs present to benchmark the original analysis in Revision 0 of the WCAP report. Due to the improved 3-D finite element modeling capability, along with the use of plant specific geometry, the baseline FEM, with all RPV head studs in-service, provided slightly higher boltup stresses as compared to the original stress analysis that was performed with the use of a 2-D FEM in Revision 0. To account for the higher boltup stresses, a slightly smaller reference flaw size is used in the fracture mechanics evaluation for the baseline case to demonstrate the continued elimination of the Appendix G flange requirement in the P-T limit curves with all RPV head studs in-service. The smaller reference flaw size is well within the detection capability of NDE examinations. Therefore, the continued elimination of the Appendix G flange requirement in the P-T limit curves is reaffirmed for the baseline case with all RPV head studs in-service as described in detail in Appendix E.

Next, to evaluate the case for the missing RPV head stud, another 3-D FEM was created to determine the impact of the one stud out-of-service on the boltup, heatup, and cooldown stresses. Based on the stress analysis, it was determined that the boltup stresses from the 3-D FEM with the missing RPV head stud were less than the boltup stresses from the baseline 3-D model with all RPV head studs in-service. For the fracture mechanics evaluation, the boltup condition is limiting during heatup and cooldown. Therefore, the stress analysis and subsequent fracture mechanics evaluation for the missing RPV head stud case determined that for the boltup condition, the all RPV head studs intact case is more limiting. Furthermore, the results of the missing RPV head stud case are in agreement with the conclusions determined in Revision 0 of the WCAP. Specifically, the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves. Thus, the conclusions reached in WCAP-16143, Revision 0, remain valid for Braidwood Station, Unit 2, and Byron Station, Unit 2, each with an out-of-service RPV head stud.

In conclusion, based on the stress analysis and the fracture mechanics evaluation, the original conclusions determined in WCAP-16143, Revision 0, are still valid and the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves for Byron Station and Braidwood Station, Units 1 and 2.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether the applicable regulations and requirements, noted below, continue to be met.

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," part (a) requires that the fracture toughness and material surveillance program for the reactor coolant system pressure boundary must meet the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements." 10 CFR 50.60(b) allows alternatives to the requirements when an exemption is granted under 10 CFR 50.12, "Specific exemptions."

TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," Item (b) requires that the analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC; and provides a specific list of references.

EGC has determined that the previous Exemption Request to allow the use of WCAP-16143 in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.c as required by 10 CFR 50.60(a), approved by the NRC in Reference 2, remains valid and properly supported by WCAP-16143, Revision 1. Once approved, the PTLR will be updated to reflect the WCAP revision number and report date, consistent with TS 5.6.6.

4.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. This amendment request proposes to utilize WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated October 2014, as an analytical method to determine the reactor coolant system pressure and temperature limits.

In Reference 1, EGC submitted a Request for Exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." This Exemption Request proposed to use WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," dated November 2003 (i.e., Revision 0), for calculating the reactor pressure vessel (RPV) pressure-temperature (P-T) limits for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, in lieu of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.c as required by 10 CFR 50 .60(a). Reference 1 also requested a corresponding Technical Specifications (TS) change to add WCAP-16143 as a reference in TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." The NRC subsequently approved the Request for Exemption and the associated license amendment in References 2 and 3, respectively.

It has since been recognized that WCAP-16143, Revision 0, did not address a RPV head stud configuration of 53 RPV head studs, which currently exists at Braidwood Station, Unit 2, and Byron Station, Unit 2. (Note that the normal RPV head stud configuration consists of 54 RPV head studs). WCAP-16143, Revision 1, addresses the effect of the missing RPV head stud on the technical basis for elimination of the 10 CFR 50, Appendix G fracture toughness requirements; and is being submitted to the NRC for review and approval. In summary, the stress analysis and fracture mechanics evaluation for the "missing head stud" case determined that, for the boltup condition, the "all studs intact" case is more limiting. The results of the missing RPV head stud evaluation remain in agreement with the conclusions of WCAP-16143, Revision 0, submitted to the NRC in Reference 1. It should be noted that WCAP-16143, Revision 1, addresses both the originally designed 54 RPV head stud configuration and the 53 RPV head stud configuration for all Braidwood and Byron units.

In addition, a change to TS Table 1.1-1, "MODES," footnotes (b) and (c) is requested for clarity to reflect the potential missing RPV head stud condition addressed by WCAP-16143, Revision 1. Note that Table 1.1-1 defines the criteria for MODES 1 through 6.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change for Braidwood Station and Byron Station, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

Criteria

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   These changes to the analysis do not adversely affect accident initiators or precursors, nor alter the design assumptions or conditions of the facility previously approved by the NRC, or the manner in which the plant is operated and maintained. The revisions to the subject WCAP do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The changes also do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated; do not increase the types or amounts of radioactive effluent that may be released offsite; and do not significantly increase individual or cumulative occupational/public radiation exposures.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The use of WCAP-16143, Revision 1, for generation of RPV P-T limits, will continue to ensure that RPV integrity is maintained under all conditions. The revisions contained in WCAP-16143, Revision 1, and the changes proposed to TS Table 1.1-1 do not change the conclusions of WCAP-16143, Revision 0, previously approved by the NRC; nor do they change the way the RPV is analyzed or performs its safety function. Subsequently, these changes do not result in the creation of any new accident initiators or precursors; do not result in changes to any existing accident scenarios; and do not introduce any operational changes or mechanisms that would create the possibility of a new or different kind of accident.

   Based on the above discussion, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes do not change any safety limits or reduce the margin of safety to any safety limits. The stress analysis and fracture mechanics evaluation, documented in the revision to WCAP-16143, determined that for the RPV boltup condition, the RPV 54-stud case (i.e., all RPV head studs in-service) was more limiting than the RPV 53-stud case (i.e., one RPV head stud out-of-service). In addition, the conclusions of the updated analysis for the RPV 54-stud case confirmed the conclusions from WCAP-16143, Revision 0. This change, subsequently, has no impact on the current RPV P-T limit curves.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed changes do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the site licensing basis and Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has evaluated this proposed operating license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that the changes made in WCAP-16143, Revision 1, and the proposed changes to TS Table 1.1-1 meet the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that these changes are being proposed as an amendment to the license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.2, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed revisions made in WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," Revision 1, dated October 2014, and the proposed changes to TS Table 1.1-1 do not result in an increase in power level, do not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts. These changes do not affect any plant equipment that are assumed to operate as designed in the event of an accident to minimize the potential for leakage of radioactive effluents. The proposed changes will have no impact on the amounts of radiological effluents released offsite during normal at-power operations or during the accident scenarios.

Based on the above evaluation, the proposed changes will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There is no change in individual or cumulative occupational radiation exposure due to the proposed changes. Specifically, the revisions made in WCAP-16143, Revision 1, and the proposed changes to TS Table 1.1-1 have no impact on any radiation monitoring system setpoints. The proposed action will not change the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed action result in any change in the normal radiation levels within the plant.

Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Therefore, in accordance with 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from K. R. Jury (Exelon Generation Company, LLC) to NRC, "License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limits Report and Request for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," dated October 3, 2005

2. Letter from R. F. Kuntz (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Exemption From the Requirements of 10 CFR 50, Appendix G," dated November 22, 2006

3. Letter from R. F. Kuntz (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of amendments Re: Reactor Coolant System Pressure and Temperature Limits Report," dated November 27, 2006

4. Letter from D. E. Hills (NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Notice of Violation and Braidwood Station, Units 1 and 2, NRC Baseline Inservice Inspection Report 05000456/2013008; 05000457/2013008" dated November 14, 2013

5. Letter from E. R. Duncan (NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2, NRC Integrated Inspection Report 05000454/2013005; 05000455/2013005" dated February 3, 2014

6. Letter from M. E. Kanavos (Exelon Generation Company, LLC) to NRC, "Reply to a Notice of Violation, EA-13-209," dated December 13, 2013

ATTACHMENT 2A

Markup of Technical Specifications Page

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

Facility Operating License Nos. NPF-72 and NPF-77

<u>REVISED TS PAGE</u>

1.1-9

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All required reactor vessel head closure bolts fully tensioned. |

(c) One or more required reactor vessel head closure bolts less than fully tensioned. |

ATTACHMENT 2B

Markup of Technical Specifications Page

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

Facility Operating License Nos. NPF-37 and NPF-66

<u>REVISED TS PAGE</u>

1.1-9

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[(b)]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All required reactor vessel head closure bolts fully tensioned.

(c) One or more required reactor vessel head closure bolts less than fully tensioned.

ATTACHMENT 3

BRAIDWOOD STATION, UNIT 1 AND UNIT 2
BYRON STATION, UNIT 1 AND UNIT 2

Westinghouse Electric Company, LLC

Application for Withholding Proprietary Information from Public Disclosure, CAW-14-4042

Affidavit

Proprietary Information Notice and Copyright Notice

Westinghouse

Westinghouse Electric Company
Engineering, Equipment and Major Projects
1000 Westinghouse Drive, Building 3
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

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e-mail: greshaja@westinghouse.com
Proj letter: CAE-14-MUR-5

CAW-14-4042

October 8, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-14-4042 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Exelon.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference CAW-14-4042 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 8th day of October 2014

Anne M. Stegman

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Anne M. Stegman, Notary Public
North Huntingdon Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16143-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2" (Proprietary), dated October 2014, for submittal to the Commission, being transmitted by Exelon letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of WCAP-16143-P Revision 1, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Obtain NRC approval of WCAP-16143-P Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2."

(ii) Provide justification to support the elimination of the 10 CFR 50, Appendix G flange requirements from the pressure-temperature limit curves for Byron and Braidwood Units 1 and 2.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing justification to support the elimination of the 10 CFR 50, Appendix G flange requirements from the pressure-temperature limit curves.

(ii) Westinghouse can sell support and defense of providing justification to support the elimination of the10 CFR 50, Appendix G flange requirements from the pressure-temperature limit curves.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

ATTACHMENT 5

WCAP-16143-NP
Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation
for
Byron/Braidwood Units 1 and 2

Revision 1
October 2014

(Non-Proprietary)

Westinghouse Non-Proprietary Class 3

WCAP-16143-NP
Revision 1

October 2014

Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2

Westinghouse

WCAP-16143-NP
Revision 1

Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2

Nathan Glunt*
Piping Analysis & Fracture Mechanics

Warren Bamford*
Primary Systems Design & Repair

October 2014

Reviewer: Amy Freed*
Materials Center of Excellence

Reviewer: Anees Udyawar*
Piping Analysis & Fracture Mechanics

Approved: John McFadden, Manager*
Piping Analysis & Fracture Mechanics

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

Record of Revisions		
Rev.	Date	Revision Description
0	November 2003	Original Issue
1	October 2014	Revision 1 incorporates the additional evaluation of the Byron and Braidwood Units 1 and 2 reactor vessel closure head/vessel flange requirements with one closure head stud out of service. Appendix E is added to include this additional evaluation and Appendix F is added to include the updated stresses from a plant specific three-dimensional finite element model. The flange elimination from Byron and Braidwood Units 1 and 2 Pressure-Temperature limit curves with all studs intact is also re-affirmed based on stresses resulting from plant specific three-dimensional finite element models.

Note that there are several locations in this report where proprietary information has been identified and bracketed. For each of the bracketed locations, the reason for the proprietary classification is given, using a standardized system. These codes are listed with their meanings in WCAP-7211, Revision 5 “Proprietary Information and Intellectual Property Management Policies and Procedures.” The proprietary brackets are labeled with three different letters to provide this information and the explanation for each letter is given below:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc., and the prevention of its use by Westinghouse's competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

c. The information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

e. The information reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

TABLE OF CONTENTS

1 INTRODUCTION

10 CFR Part 50, Appendix G contains requirements for pressure-temperature (P-T) 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 [1], but the flange temperature requirements are specified in 10CFR50 Appendix G. This rule states that the metal temperature at 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 N-640, "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 for Byron/Braidwood Units 1 and 2.

Revision 1

Justification for elimination of the flange requirement from the Byron and Braidwood Units 1 and 2 P-T limits curves was previously provided in Revision 0 of this report. The Byron and Braidwood closure heads were designed to use 54 closure studs to couple the closure head and vessel flanges and seal the main closure joint. Currently 53 studs are active in Byron and Braidwood Units 2 [13], thus the load previously carried by the out-of-service stud is distributed throughout the surrounding studs. Therefore, the evaluation performed in Revision 1 of this report will determine whether the flange requirements (see Figure 1-1) remain eliminated from the P-T limit curves for Byron Unit 2 and Braidwood Unit 2 with 53 operational closure studs. Additionally, the elimination of the flange requirement from the Byron Unit 1 and Braidwood Unit 1 P-T limits will also be evaluated assuming a hypothetical missing stud, as requested by the customer. The elimination of the flange requirement from Byron and Braidwood Units 1 and 2 P-T limit curves will be reassessed based on stresses resulting from more up-to-date finite element stress analysis models. Results are summarized in Section 5 of this report and are fully discussed in Appendices E and F.

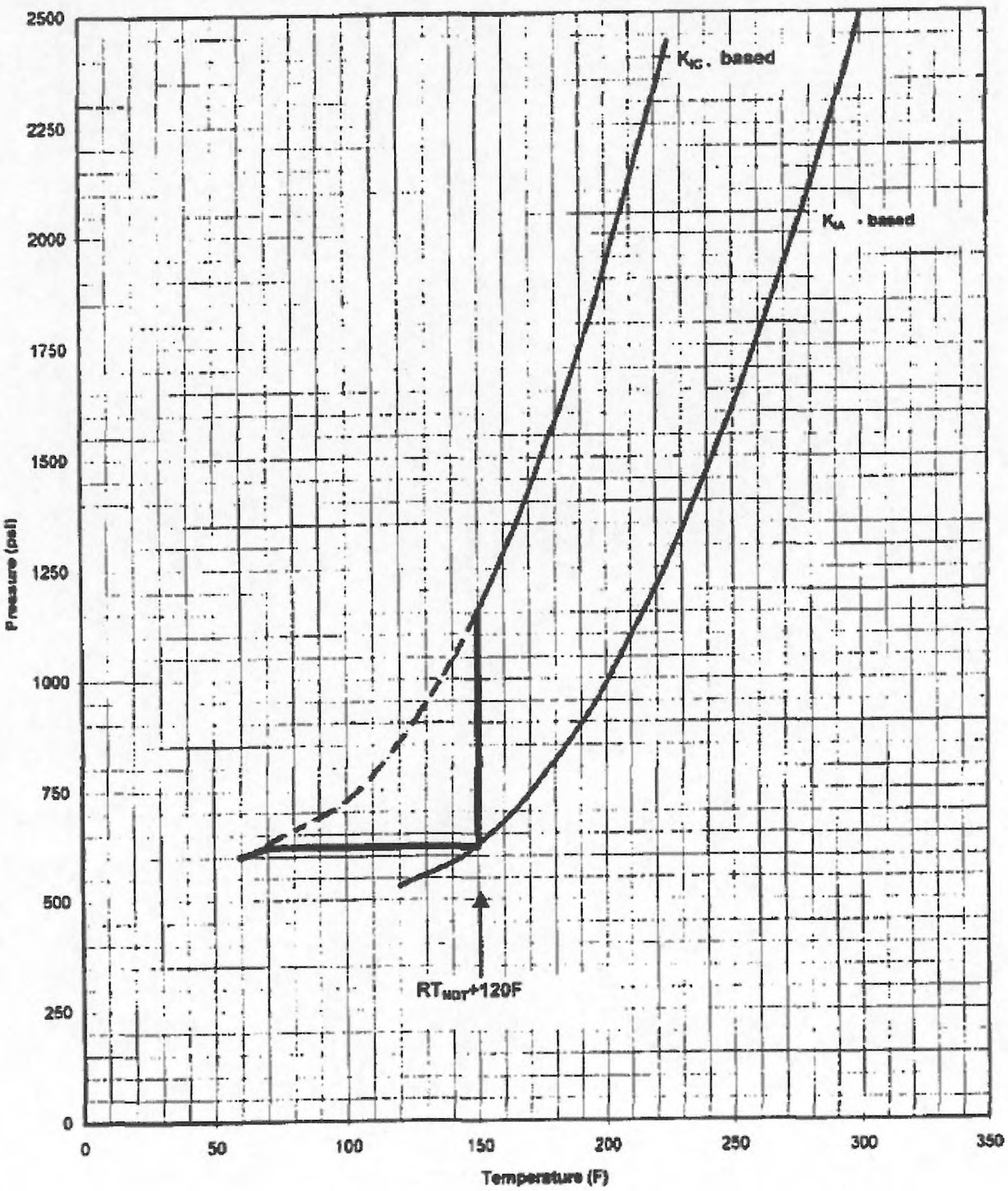


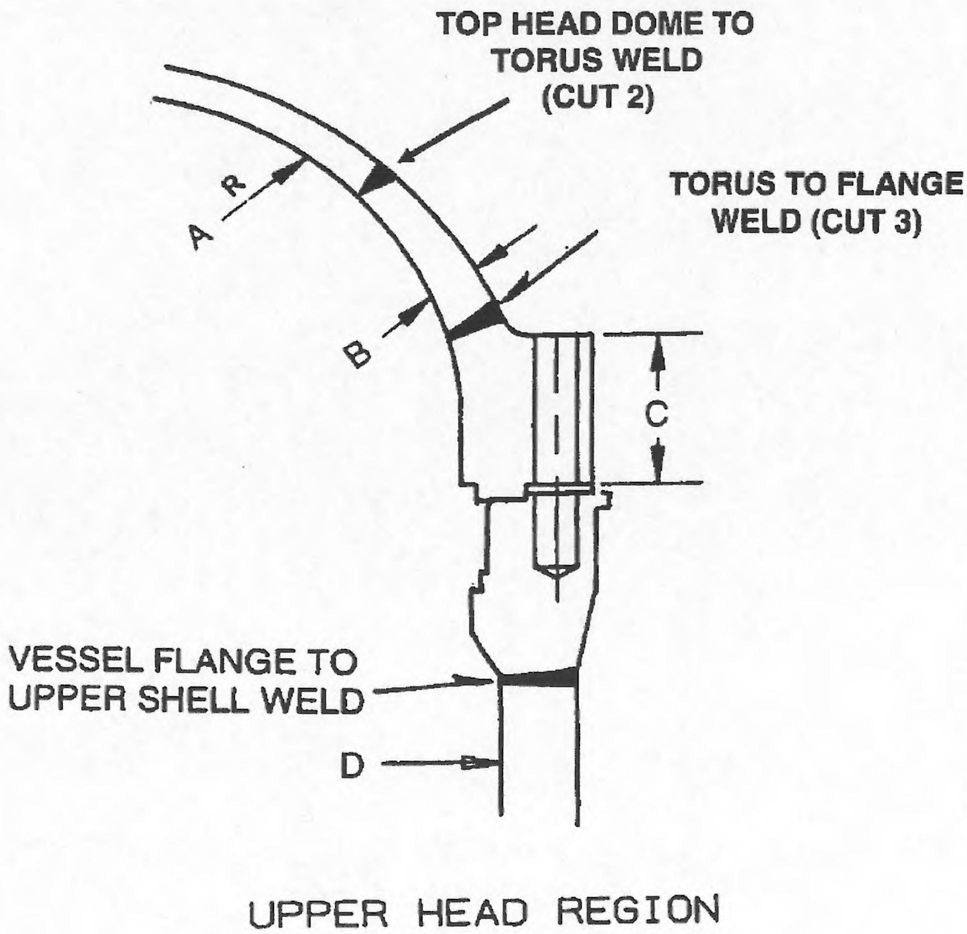
Figure 1-1 Illustration of the Impact of the Flange Requirement for a Typical PWR Plant

2 TECHNICAL APPROACH

The evaluation presented here is intended to cover the Byron/Braidwood Units 1 and 2 reactor vessels. Fracture evaluations have been performed on the closure head geometry specific to these units, and results will be tabulated and discussed. The geometry of the closure head region for Byron/Braidwood Units 1 and 2 is shown in Figure 2-1.

Stress analyses have been performed, and these stress results were used to perform fracture mechanics evaluations. Details of the finite element stress analysis results are provided in Appendix C. 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 two welds as shown in Figure 2-1. 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 the outside surface.

The goal of the evaluation is to compare the structural integrity of the closure head during the boltup, plant heatup and plant cooldown processes, to the structural 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 structural integrity of the closure head during boltup?



	Byron Units 1 and 2	Braidwood Units 1 and 2
A	88.3	88.3
B	6.625	6.625
C	30.05	30.11
D	170.88	170.88

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-1 Geometry of the Upper Head/Flange Region of the Byron/Braidwood Units 1 and 2 Reactor Vessels

3 FRACTURE ANALYSIS METHODS AND MATERIAL PROPERTIES

The fracture evaluation was carried out using the approach suggested by Section XI Appendix G (Ref. 1). A semi-elliptic surface flaw was postulated to exist in the highest stressed region, which is at the outside surface of the closure flange. The flaw depth was assumed to encompass a range of depths into 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). In most cases, the stress intensity factor for the structural integrity calculations utilized a representation of the actual stress profile rather than a linearization. The stress profile was represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1x + A_2x^2 + A_3x^3 \tag{3-1}$$

where:

x	=	the coordinate distance into the wall, 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 (Ref. 2) was used. The stress intensity factor K_I can be calculated anywhere along the crack front. The point of maximum crack depth is represented by $\varphi = 0$, 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 as a function of the angular location around the crack front (φ). The units of K_I are ksi$\sqrt{\text{in}}$.

$$K_I = \left[\frac{\pi a}{Q}\right]^{0.5} \sum_{j=0}^{3} G_j\ (a/c, a/t, t/R, \varphi)\ A_j a^j \tag{3-2}$$

The boundary correction factors G_0, G_1, G_2, and G_3 are obtained by the procedure outlined in reference (2). The dimension "a" is the crack depth, "c" is the crack half length, "t" is the wall thickness, "R" is the inside radius, and "Q" is the flaw shape factor, which can be approximated by $Q = 1 + 1.464\ (a/c)^{1.65}$.

3.2 FRACTURE TOUGHNESS

Another 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:

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

$$K_{Ia} = 26.8 + 12.445 \exp[0.0145\ (T - RT_{NDT})] \tag{3-4}$$

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. This value is consistent with general practice in such evaluations, as shown for example in reference 3, which provided the background and technical basis for the development of Appendix A of Section XI.

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

The value of RT_{NDT} for the closure flange region of the Byron/Braidwood units was obtained from the certified material test reports [12]. The results are shown in Table 3-1. The highest value was 60°F and so this value 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.

Table 3-1 RT_{NDT} Values for Closure Head Region

a,c,e

Location	Byron		Braidwood	
	Unit 1	Unit 2	Unit 1	Unit 2

4 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 the entire heatup and cooldown process are summarized in Appendix C.

The boltup is the key condition to review here, in comparison with the heatup and cooldown operation, since the flange requirement applies to boltup conditions. No other transients result in stresses in this region at low temperatures. One might suggest that the cooldown might be of similar concern, but the boltup is governing for a number of reasons:

1. The heatup and cooldown transient is structured to ensure generous margins are maintained (SF = 2) for a large flaw in the irradiated beltline region, not for the unirradiated flange region.
2. The cooldown transient has much higher temperatures in the head region than the boltup, and
3. The thermal stresses caused by the cooldown transient tend to counteract the boltup stresses; cooldown thermal stresses are tensile on the inside surface and compressive on the outside surface.

Table 4-1 provides a comparison of the stresses at boltup with those at the governing time step of heatup and cooldown which is end of heatup. 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. These results were taken from a finite element analysis of the heatup/cooldown process, and the boltup stress alone was compared with the most limiting time step of the entire heatup/cooldown transient, which includes pressure, thermal, and boltup stresses.

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_I, (or crack driving force) was calculated. The results are shown for cut 3 weld region in Figure 4-1, and for the cut 2 weld region in Figure 4-2. For a semi-elliptic surface flaw with depth equal to 10 percent of the wall thickness postulated in the highest stress region of the head, the following values were determined for the stress intensity factor.

Boltup: $K_I = 24.84$ ksi$\sqrt{in}$ (for a/t = 0.1)

End of Heatup: $K_I = 49.58$ ksi$\sqrt{in}$ (for a/t = 0.1)

It will be useful to highlight the difference in the integrity for the head region using the two values of fracture toughness. The boltup temperature for a typical PWR is 60°F, so if RT_{NDT} = 60°F the ASME reference toughness values are $K_{Ia} = 39.2$ ksi$\sqrt{in}$ and $K_{Ic} = 53.9$ ksi$\sqrt{in}$. Using the K_{Ia} toughness (which was the basis for the original flange requirements) it can be seen that the toughness exceeds the applied stress intensity factor for boltup for flaws of any depth in the head thickness. From Figure 4-1, the smallest margin = 1- K_I/K_{Ia} = 0.24, when a/t = 0.36. For the heatup and cooldown transient, the coolant temperature at the governing time steps, near the end of heatup, is 557°F. The fracture toughness is therefore 200 ksi$\sqrt{in}$, so again the margin is very large.

Using the K_{Ic} toughness, which has now been adopted by Section XI for P-T Curves, it can be seen that there is also a significant margin between the fracture toughness and the applied stress intensity factor, for both the boltup and the heatup cooldown transient. 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 the Byron/Braidwood units using the K_{Ic} toughness. There are two possible mechanisms of degradation for this region, thermal aging and fatigue.

Effect of 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.

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Table 4-1 Stress Distributions at the Closure Flange Region – Byron/Braidwood Units 1 and 2

Distance (x/t)	Boltup Stress at Cut 3 (ksi)	Heatup* (344 minutes) at Cut 2 (at p=2317 psig, t=557°F)
0 (ID)	-14.38	-16.23
0.1	-10.77	
0.2	-7.83	-4.23
0.3	-5.14	
0.4	-2.66	4.14
0.5	-0.26	
0.6	-2.16	12.30
0.7	4.72	
0.8	7.54	22.63
0.9	11.24	
1.0 (OD)	19.70	40.80

* With boltup stress superimposed.

Notes: Cut 3 has the highest boltup stress
Cut 2 has the highest transient stress

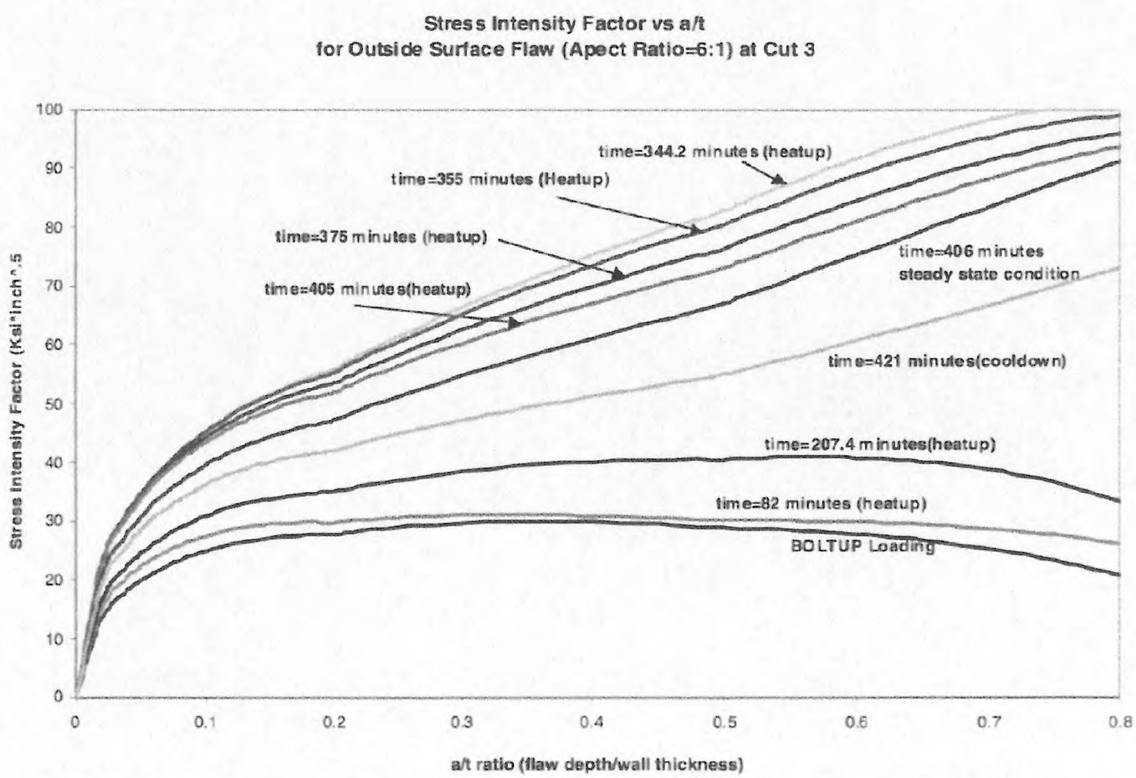

Figure 4-1 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Torus to Flange Region Weld for Byron/Braidwood Units 1 & 2

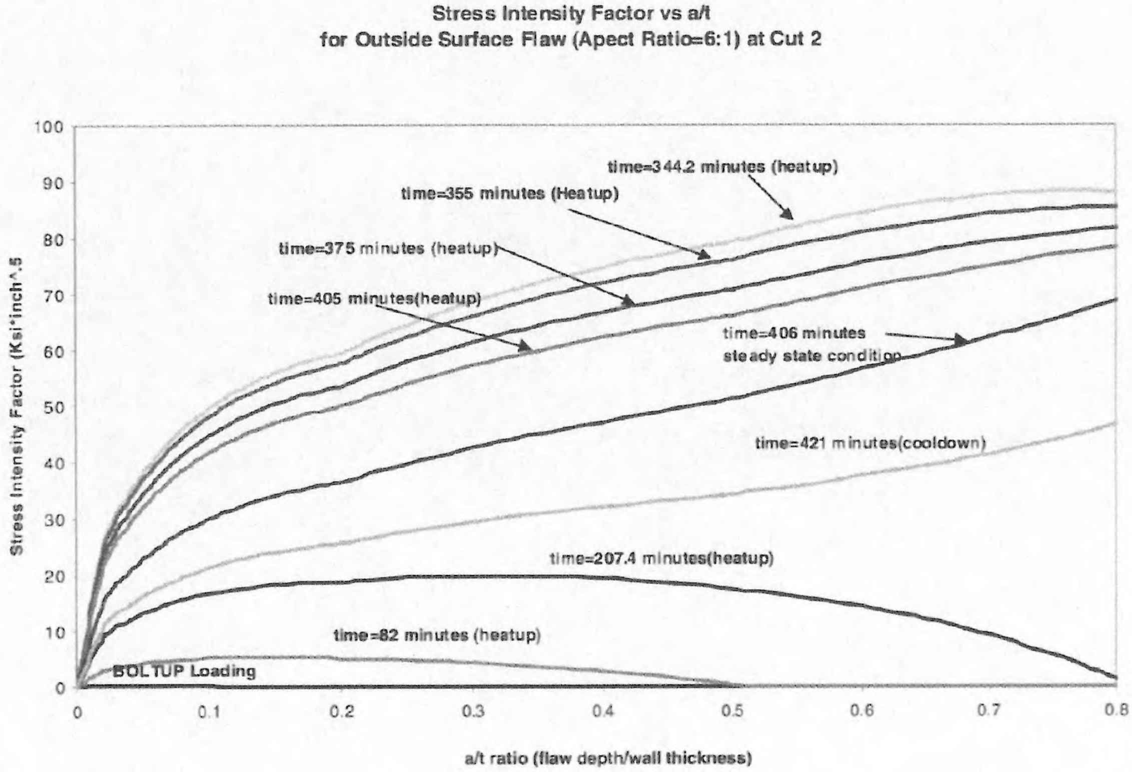

Figure 4-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Dome to Torus Region Weld for Byron/Braidwood Units 1 & 2

5 ARE FLANGE REQUIREMENTS NECESSARY?

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

The stresses at the region of interest are shown in Table 4-1, for the end of heatup, as well as boltup. Included here are the stress distributions through the wall, showing that the highest stress location for this region is the outer surface.

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.

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 (Ref. 11). Before discussing the table, it will be helpful to discuss the basis for the reference flaw, in light of current technology, and using the results of the Performance Demonstration Initiative.

Basis for the Reference Flaw Size. Regulatory Guide 1.150 stimulated improvement in examinations of the clad to base-metal interface. The same techniques have been used for more than 10 years for reactor vessel head examinations performed from the outside surface. Capability demonstrations for the clad to base-metal interface have been conducted at the EPRI NDE Center since 1983. These demonstrations were performed initially for the belt-line region. However, similar techniques are used for both the vessel belt-line and the reactor vessel head, although the head exams are done manually.

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Revision 1

The evaluation performed in Revision 1 of this report determined that the elimination of the flange requirements from the P-T limit curves for Byron and Braidwood Units 1 and 2 remains applicable for operation with 53 active closure studs. A brief conclusion is given below with the complete discussion provided in Appendix E of this report.

Two separate plant specific three-dimensional finite element models (FEMs) were developed to assess the impact of the missing stud on the elimination of the flange requirements from the P-T limit curves. Discussion of the finite element models and stress analysis are provided in Appendix F.

First, a baseline finite element model was created with all studs present to benchmark the original analysis in Revision 0 of this WCAP report. Due to the improved 3-D finite element modeling capability, along with the use of plant specific geometry, the baseline FEM with all studs active provided slightly higher boltup stresses as compared to the original stress analysis that was performed with the use of a 2-D finite element model in Revision 0. To account for the higher boltup stresses, a slightly smaller reference flaw size is used in the fracture mechanics evaluation for the baseline case to demonstrate the continued elimination of the flange requirement in the P-T limit curves with all studs active. The smaller reference flaw size is well within the detection capability of NDE examinations. Therefore, the continued elimination of the flange requirement in the P-T limit curves is reaffirmed for the baseline case with all studs active as described in detail in Appendix E.

Next, to evaluate the case for the missing stud, another 3-D finite element model was created to determine the impact of the one stud out of operation on the boltup, heatup, and cooldown stresses. Based on the stress analysis (see Appendix F), it was determined that the boltup stresses from the 3-D FEM with the missing stud were less than the boltup stresses from the baseline 3-D model with all studs active. For the fracture mechanics evaluation the boltup condition is limiting during heatup and cooldown. Therefore, the stress analysis and subsequent fracture mechanics evaluation for the missing stud case determined that for the boltup condition, the all studs intact case is more limiting. Furthermore, the results of the missing stud case are in agreement with the conclusions determined in Revision 0 of this report. Specifically, the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves. Thus, the conclusions reached in Revision 0 of this report remain valid for Byron and Braidwood Units 1 and 2 with a missing closure stud.

In conclusion, based on the stress analysis and the fracture mechanics evaluation, the original conclusions determined in Revision 0 of this report are still valid, and the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves for Byron and Braidwood Units 1 and 2.

Table 5-1 Comparison of Various Plant Designs Boltup Requirements

Plant	K_I (ksi$\sqrt{in}$) (a/t = .1)	K_I (ksi$\sqrt{in}$) (with a/t = 0.1, SF = 2)	T – RT_{NDT} (°F) using K_{Ic} (a/t = .10)	T – RT_{NDT} (°F) using K_{Ia} (a/t = .10)
CE	30.0	60.0	13	68
B&W	39.4	79.8	41	100
Byron/Braidwood	24.9	49.8	0*	43
W 3 Loop	28.7	57.5	8	63
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

* The calculated value of T-RT_{NDT} is negative, so zero is used for conservatism.

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Figure 5-1 Probability of Correct Rejection/Reporting (PCR) Considering Passed plus Failed Candidates, Appendix VIII Supplement 4, Detection from the Outside Surface. Reporting Criterion A′ = 0.15 inch, TWE Represents Flaw Depth.

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Figure 5-2 Probability of Correct Rejection/Reporting (PCR) Considering Only Passed Candidates, Appendix VIII Supplement 4, Detection from the Outside Surface. Reporting Criterion A′ = 0.15 inch, TWE Represents Flaw Depth.

6 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 temperature range 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 the case of 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.

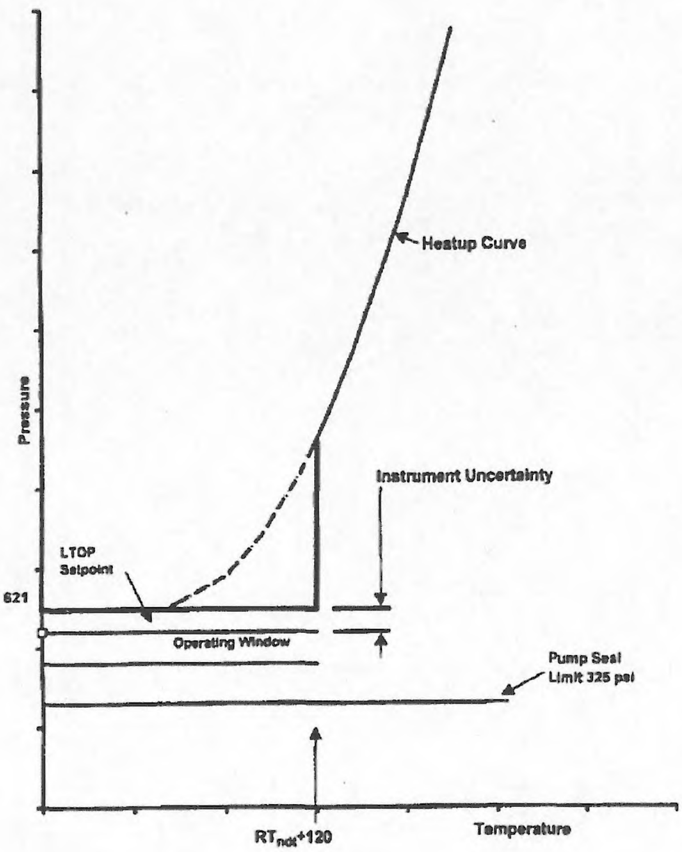

Figure 6-1 Illustration of the Flange Requirement and its Effect on the Operating Window for a Typical Heatup Curve

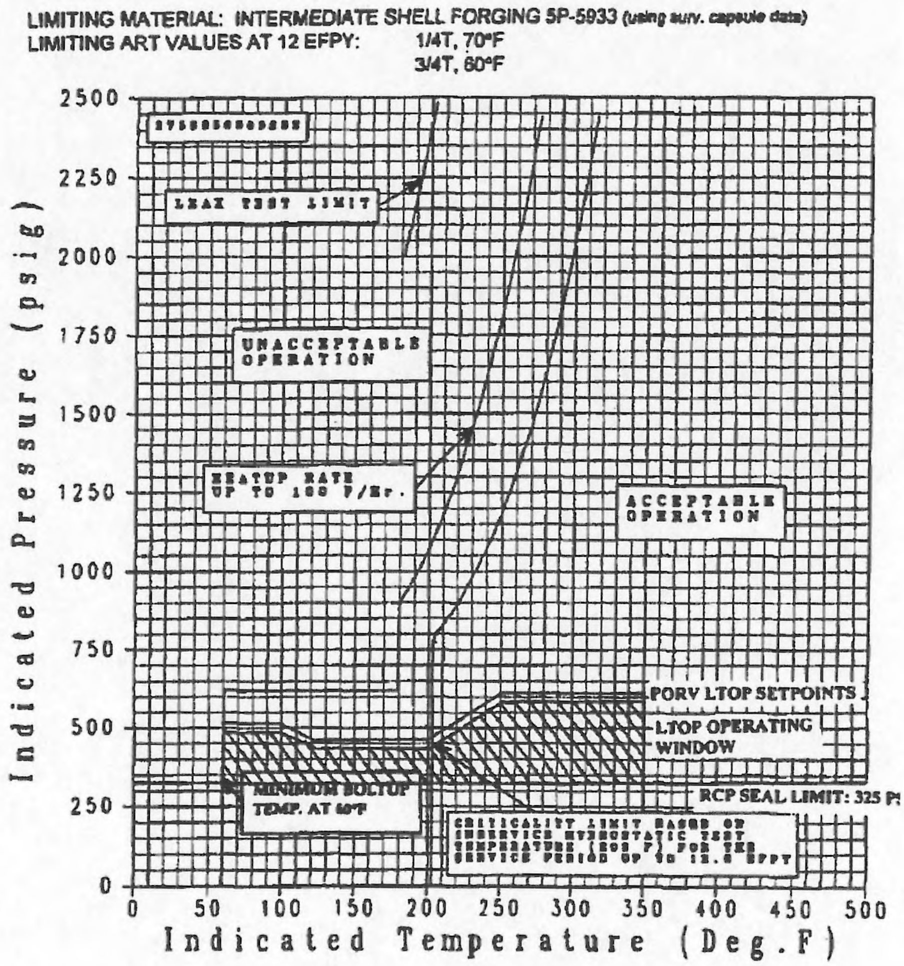

Figure 6-2 **Illustration of the Actual Operating Window for Heatup of Byron Unit 1, a Low Copper Plant at 12 EFPY**

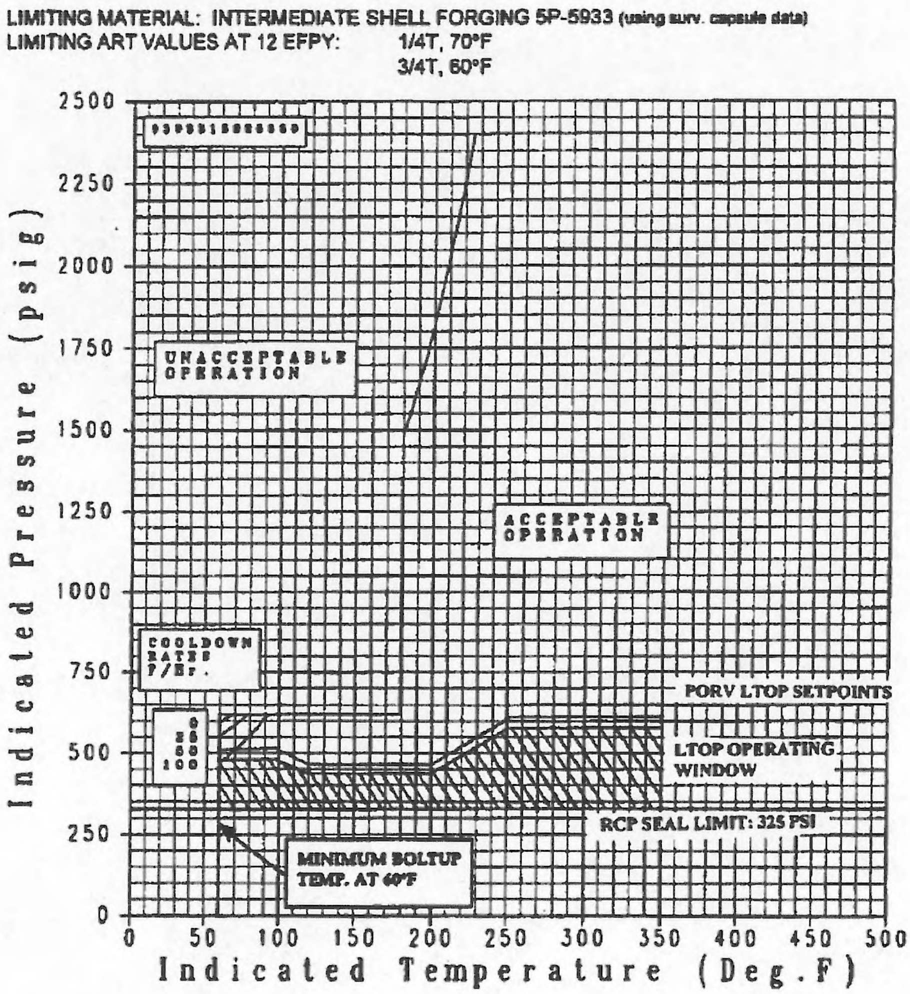

Figure 6-3 **Illustration of the Actual Operating Window for Cooldown of Byron Unit 1, a Low Copper Plant at 12 EFPY**

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APPENDIX A
REACTOR PRESSURE VESSEL INSPECTION RELIABILITY*

F. L. Becker

EPRI

Charlotte NC

1 ABSTRACT

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2 INTRODUCTION

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* Presented at the Joint EC-IAEA Technical Meeting on Improvements in Inservice Inspection Effectiveness, Pettan, The Netherlands, November 2002, to be published.

3 DETECTION

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3.1 OUTSIDE SURFACE DEMONSTRATION

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Figure A-1 Probability of Detection Performance for Passed and Passed Plus Failed Candidates for Appendix VIII Supplement 4, from the Outside Surface as a function of the flaw through wall extent (TWE). Both automated and manual techniques are included.

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Figure A-2 POD for Inside Surface Examinations, Pass and Pass + Failed Candidates, Passed and Pass Plus Failed Candidates are included.

Table A-1 Number of Measurements

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3.2 COMBINED ID AND OD DETECTION

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Figure A-3 Probability of Detection for Automated RPV Examinations Considering Both Inside and Outside Access. Passed and Passed Plus Failed Candidates are shown.

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Figure A-4 POD for Pass and Failed Candidates, Considering ID and OD Automated Demonstrations and Manual OD Demonstrations.

4 SIZING

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Figure A-5 Histogram of Depth Successful Sizing Candidate Test Scores, Appendix VIII, Supplement 4. Examinations Were Performed Both From the Inside and Outside Surfaces.

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Figure A-6 Sizing Error Surface Model

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Figure A-7 Plan View of Sizing Error Surface Model

5 ACCEPTABILITY EVALUATION

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Figure A-8 Probability of Correct Sizing for Passed Candidates, Appendix VIII Supplement 4. Reporting Threshold A′ = 0.15 inch.

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Figure A-9 **Probability of Correct Rejection/Reporting (PCR) for automated techniques, Considering Passed and Passed plus Failed Candidates, includes both inside and outside surface information. Reporting Criterion A′ = 0.15 inch.**

6 SUMMARY

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7 REFERENCES

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APPENDIX B
THERMAL AGING OF FERRITIC RPV STEELS AT REACTOR OPERATING TEMPERATURES

1 BACKGROUND

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2 KEY THERMAL AGING DATA AT TEMPERATURES CLOSE TO REACTOR OPERATING TEMPERATURES

2.1 DATA FROM LONG-TERM AGING STUDIES

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Table B-1 Compositions of the Materials Studied by DeVan et al. [3]				

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Table B-2 T_{41J} Before and After Long-Term Thermal Aging (DeVan et al. [3])					

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2.2 DATA FROM LABORATORY STUDIES

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Table B-3 Composition of A533B-1 Materials Studied by Williams and Ellis [11]

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Figure B-1 Plot of Vickers Hardness Versus Time for Thermal Aging at 330°C [11]

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3 CONCLUSIONS AND APPLICATION

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4 REFERENCES

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APPENDIX C
STRESS DISTRIBUTIONS IN THE CLOSURE HEAD REGION

C.1 STRESS ANALYSIS

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C.1.1 Mechanical Boundary Conditions

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C.1.2 Thermal Boundary Conditions

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C.1.3 Bolt Pre-Load

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C.1.4 Stress Results

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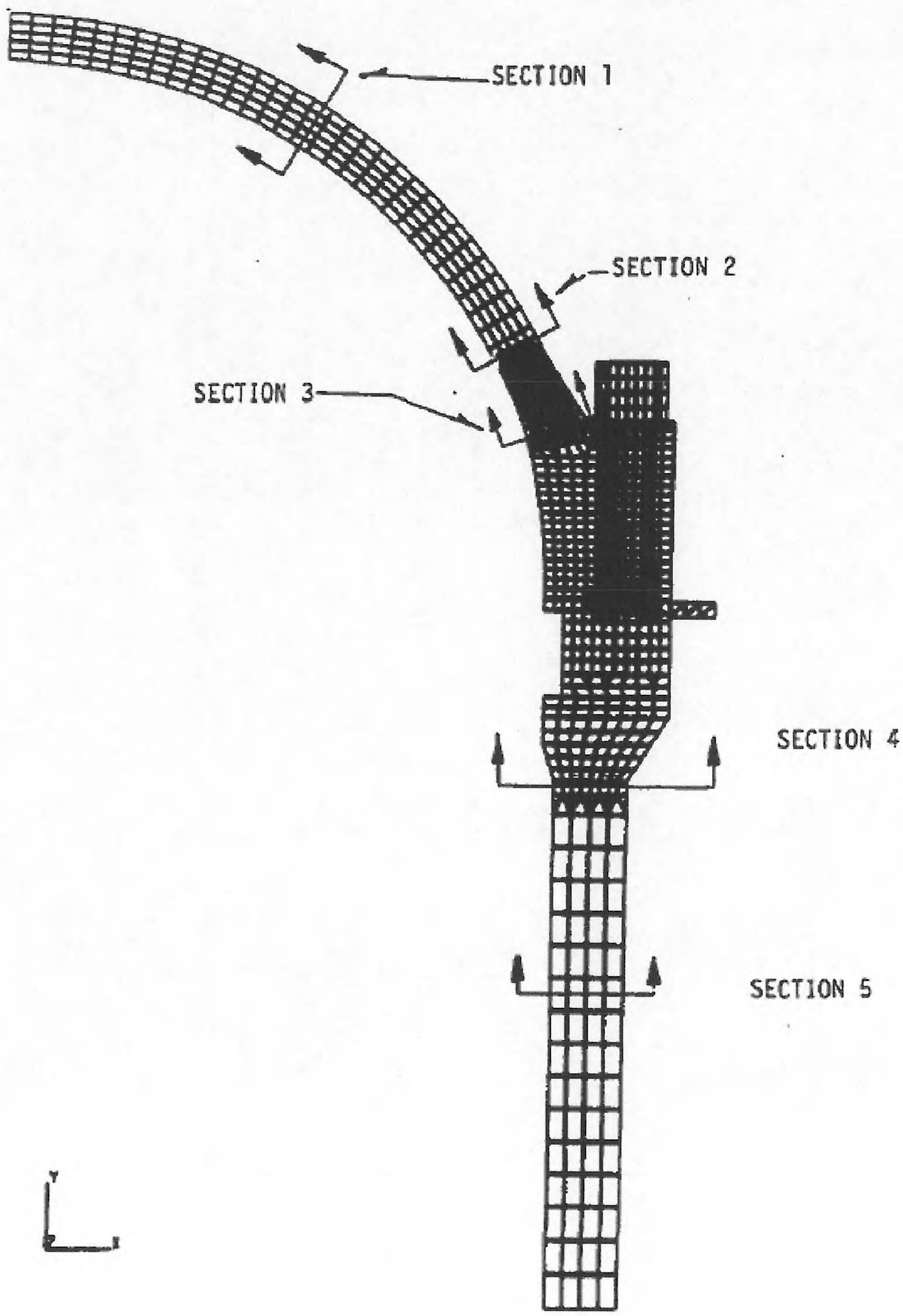

Figure C-1 Finite Element Model for Closure Head Region, Byron and Braidwood Units 1 and 2

Table C-1 Stress for Upper Head to Flange Transient Region (Cut 2)

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Table C-1 Stress for Upper Head to Flange Transient Region (Cut 2)

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Table C-1 Stress for Upper Head to Flange Transient Region (Cut 2)

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Table C-2 Stress for Upper Head to Flange Transition Region (Cut 3)

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Table C-2 Stress for Upper Head to Flange Transition Region (Cut 3)

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Table C-2 Stress for Upper Head to Flange Transition Region (Cut 3)

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APPENDIX D
FLANGE INSPECTION RESULTS: BYRON AND BRAIDWOOD PLANTS

These exams were performed using ASME Section XI and ASME Section V techniques and requirements. As required by Section XI and as listed in Tables D-1 and D-2, the head to flange weld was examined by both ultrasonic testing and the magnetic particle method with no recordable indications.

Volumetric exams. The approved ultrasonic testing examination procedure used was NDT-C-30 revision 7 which was in accordance with ASME Section XI 1989 Edition and in compliance with NRC Regulatory Guide 1.150. The calibration standards utilized were fabricated in accordance with ASME requirements and were actual material dropouts from the component. The examination sensitivity (both straight beam and angle beam) was established from signal responses from a side drilled hole.

Surface exams. The primary location of concern for the flange region is the outer surface, where the tensile stresses are the highest. This area has been inspected by the magnetic particle technique, which is very reliable.

The volumetric inspection, along with the surface inspection and the visual (VT-2) inspections performed every refueling outage, demonstrates the continued structural integrity of the Reactor Vessel Head to Flange weld. Furthermore, past First Interval inspections, Preservice inspections, ASME Section III construction inspections and every refueling outage VT-2 inspection have revealed no recordable indications and provide reasonable assurance of the continued structural integrity of this weld.

Table D-1 Byron Reactor Vessel Flange Examination History

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
Unit 1 Reactor Vessel Head	Flange to head weld (ISI# 1RC-01, RVCH-01)	Surface examination (magnetic particle) and Ultrasonic exams using 0, 45, and 60 degree scans.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from a 0.210 inch diameter side drilled hole.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 73% due to configuration and 3 integrally mounted lifting lugs, (Relief Request I2R-25).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in March 2002.
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 1 Reactor Vessel	Flange ligament (ISI # 1RC-01-R, FLG THREADS 01-54)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.437-inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-12.	No recordable indications.	The examination was completed in March 1999 on all 54 of the flange ligament areas. This was the second complete examination of these areas during the Unit's commercial operation.
	Flange to shell weld (ISI # 1RC-01-R, W-07)	Ultrasonic examination using automated technique with 0, 45, 60 and 70 degree shear and longitudinal wave scans.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.125 inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-4.	No recordable indications.	The examination was last performed in April 1996.

Table D-1 (cont.) **Byron Reactor Vessel Flange Examination History**

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
Unit 2 Reactor Vessel Head	Flange to head weld (ISI # 2RC-01, RVCH-01)	Surface examination (magnetic particle) and Ultrasonic exams using 0, 45, and 60 degree scans.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from a 0.210 inch diameter side drilled hole.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 73% due to configuration and 3 integrally mounted lifting lugs, (Relief Request I2R-25).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in September 2002.
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 2 Reactor Vessel	Flange ligament (ISI # 2RC-01-R, FLG THREADS 01-54)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.437-inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-12.	No recordable indications.	The examination was completed in April 2001 on all 54 of the flange ligament areas. This was the second complete examination of these areas during the Unit's commercial operation.
	Flange to shell weld (ISI # 2RC-01-R, W-07)	Ultrasonic examination using automated technique with 0, 45, 60 and 70-degree shear and longitudinal wave scans.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.125 inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-4.	No recordable indications.	The examination was last performed in April 1998.

Table D-2 Braidwood Reactor Vessel Flange Examination History

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
Unit 1 Reactor Vessel Head	Flange to head weld (ISI# 1RV-03-001)	Surface examination (magnetic particle) and Ultrasonic exams using 0, 30, 40, 45, and 60 degree scans.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from a 0.210 inch diameter side drilled hole.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 88% due to configuration and 3 integrally mounted lifting lugs, (see NRC approved Relief Request I2R-20).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in September 1998.
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 1 Reactor Vessel	Flange ligament (ISI # 1RV-02-038)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.437-inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-12.	No recordable indications.	The examination was completed in March 2000 on all 54 of the flange ligament areas. This was the second complete examination of these areas during the Unit's commercial operation.
	Flange to shell weld (ISI # 1RV-01-005)	Ultrasonic examination using automated technique with 45 and 70 degree shear and longitudinal wave scans.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.125 inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-4.	No recordable indications.	The examination was last performed in April 1997 using a technique that was demonstrated and qualified to the Performance Demonstration Initiative (PDI) Program.

Table D-2 (cont.) Braidwood Reactor Vessel Flange Examination History

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
Unit 2 Reactor Vessel Head	Flange to head weld (ISI # 2RV-03-001)	Surface examination (Magnetic particle) and Ultrasonic exams using 0, 30, 40, 45, and 60 degree scans.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from a 0.210 inch diameter side drilled hole.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 88% due to configuration and 3 integrally mounted lifting lugs, (see NRC approved Relief Request I2R-20).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in April 1999.
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 2 Reactor Vessel	Flange ligament (ISI # 2RV-02-038)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.437-inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-12.	No recordable indications.	The examination was completed in October 2000 on all 54 of the flange ligament areas. This was the second complete examination of these areas during the Unit's commercial operation.
	Flange to shell weld (ISI # 2RV-01-005)	Ultrasonic examination using automated technique with 45 and 70 degree shear and longitudinal wave scans.	The sensitivity of the ultrasonic exam is based on the signal responses from a 0.125 inch diameter side drilled hole.	Achieved 100% of the ASME Section XI coverage, Figure IWB-2500-4.	No recordable indications.	The examination was last performed in October 1997 using a technique that was demonstrated and qualified to the Performance Demonstration Initiative (PDI) Program.

APPENDIX E
REEVALUATION OF BYRON AND BRAIDWOOD REACTOR VESSEL CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS TO ACCOUNT FOR A MISSING CLOSURE STUD

E.1 BACKGROUND

The Byron and Braidwood closure heads were designed to use 54 closure studs to couple the closure head and vessel flanges and seal the main closure joint. However, Byron and Braidwood Units 2 each have one closure stud out of service (Reference E-4). Since only 53 studs are active, the load previously carried by the out-of-service stud will be distributed throughout the surrounding studs. This redistribution of load causes changes in the stresses in the reactor vessel closure head/vessel flange regions. Therefore, an evaluation is performed in this appendix to address the effect of the missing stud on the technical basis for elimination of the 10 CFR 50, Appendix G flange requirements from the P-T limits for Byron and Braidwood Units 2. Additionally, the elimination of the flange requirement from the Byron Unit 1 and Braidwood Unit 1 P-T limits will be evaluated assuming a hypothetical missing stud, as requested by the customer. The elimination of the flange requirements from the P-T limits of Byron and Braidwood Units 1 and 2 will also be re-affirmed for the case with all studs active based on the updated stress model. The updated stress model is based on a plant-specific three-dimensional finite element analysis.

E.2 TECHNICAL APPROACH

The evaluation presented in this appendix is intended to confirm that the elimination of the flange requirements from the P-T limit curves that was originally concluded in Revision 0 of this report remains valid for Byron and Braidwood Units 1 and 2, whether one closure head stud is missing or all closure head studs are in service. Fracture mechanics evaluations have been performed on the closure head geometry specific to these units using plant-specific three dimensional finite element analysis (FEA) models.

The existing stress analysis for the Byron and Braidwood Units 1 and 2 reactor vessel closure head/vessel flange was based on a two-dimensional axisymmetric FEA model of the reactor vessel closure head region of a Westinghouse standard 4-loop model with all 54 studs in place, as described in Appendix C of this report. For Revision 1 of this report, two new FEA models were developed for the evaluation of a missing closure head stud in Byron and Braidwood Units 1 and 2. Both models are three-dimensional. The first FEA model was generated to evaluate the stresses near the missing closure head stud. For completeness, the second model was generated as a baseline case with all 54 studs present for comparison with the original 2-D FEA model developed in Revision 0 of the WCAP report. The stresses determined using the second model are representative of the stresses sufficiently far away from a missing stud. The stresses determined based on the second model are also representative of a unit that does not have any studs out of service. The finite element models are described in greater detail in Appendix F.

The FEA model with one closure head stud missing is shown in Figure E-1 and the FEA model with no missing closure head studs is shown in Figure E-2. The boltup, heatup, and cooldown stresses were generated for several different through wall cuts in the vicinity of the closure head flange region for each model. The cuts were taken at five planes around the circumference of the reactor vessel, as shown in

Figure E-3. The same cuts were taken in both models. Each plane either goes through the center of a stud (including the missing stud) or directly between two studs. The cuts were taken on the previously mentioned planes at several elevations throughout the closure head and flange region. The locations of the cuts are shown in Figure E-4, with the location of the two limiting cuts (3 and 35) identified. The highest stressed location in the closure head and vessel flange region is on the outside surface of the head, just above the bolting flange. Therefore, the fracture evaluation has assumed a postulated flaw on the outside surface in the vicinity of the closure head to flange weld.

The goal of the evaluation is to compare the structural integrity margins of the closure head during the boltup, plant heatup and plant cooldown processes, considering one missing stud and all studs present in Byron and Braidwood Units 1 and 2. This comparison is used to demonstrate that the 10 CFR 50, Appendix G flange requirement is not necessary and can be eliminated from the P-T limits for Byron and Braidwood Units 1 and 2.

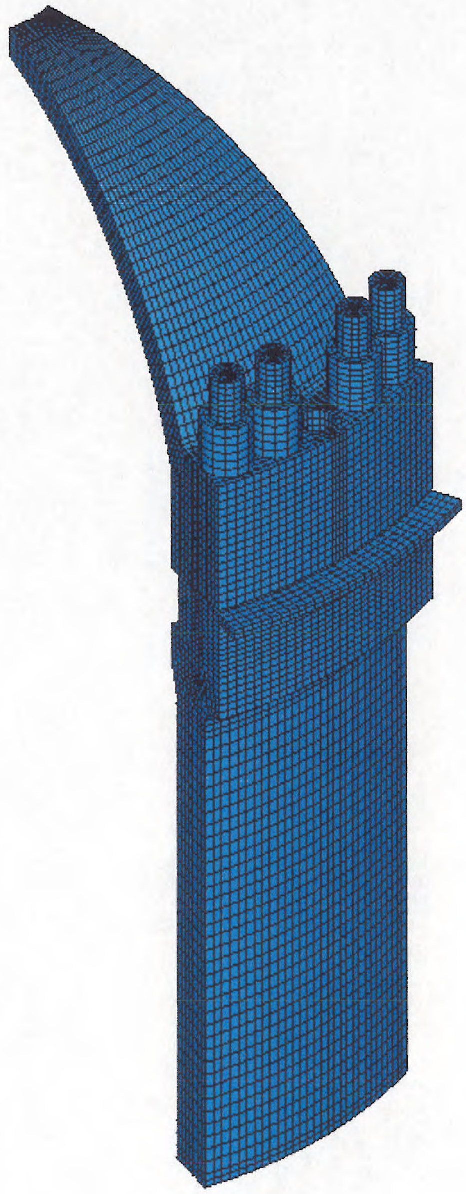


Figure E-1 Geometry of the Upper Head/Flange Region of the Byron and Braidwood Reactor Vessels with One Missing Closure Head Stud

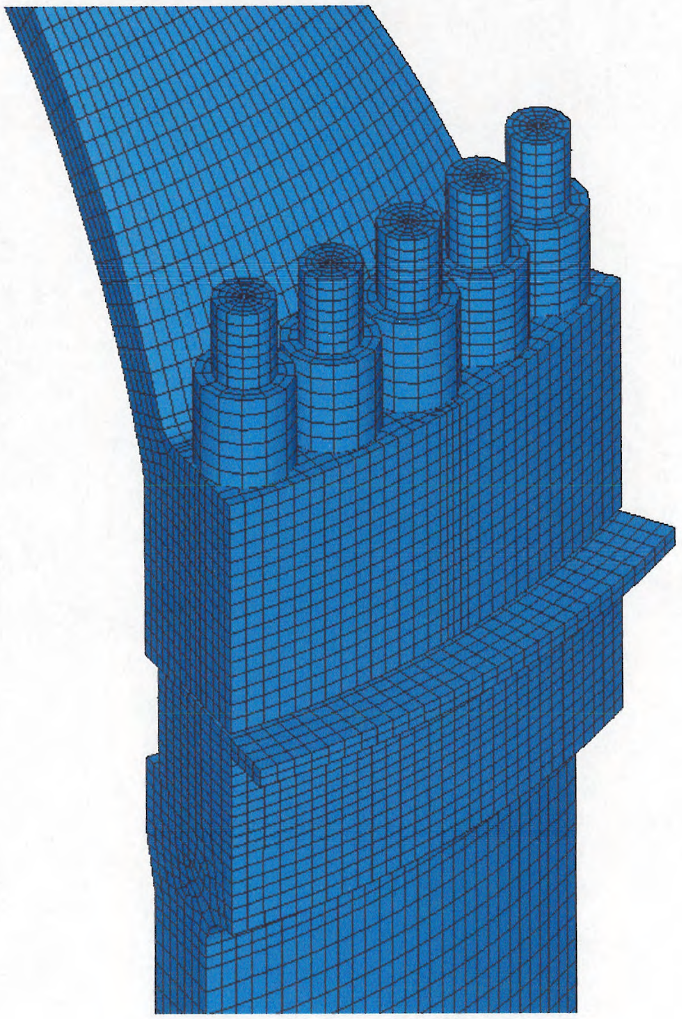
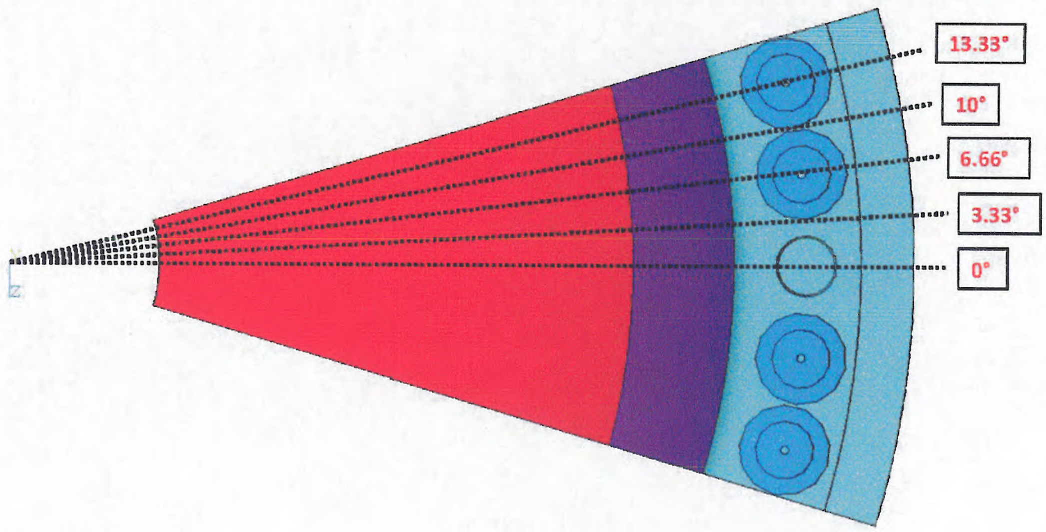


Figure E-2 Geometry of the Upper Head/Flange Region of the Byron and Braidwood Reactor Vessels with No Missing Closure Head Stud



0° - Cuts 1 through 8
6.66° - Cuts 17 through 24
13.33° - Cuts 33 through 40
3.33° - Cuts 9 through 16
10° - Cuts 25 through 32

Figure E-3 Upper Head/Flange Region Stress Cut Planes
(Missing Stud Model Shown)

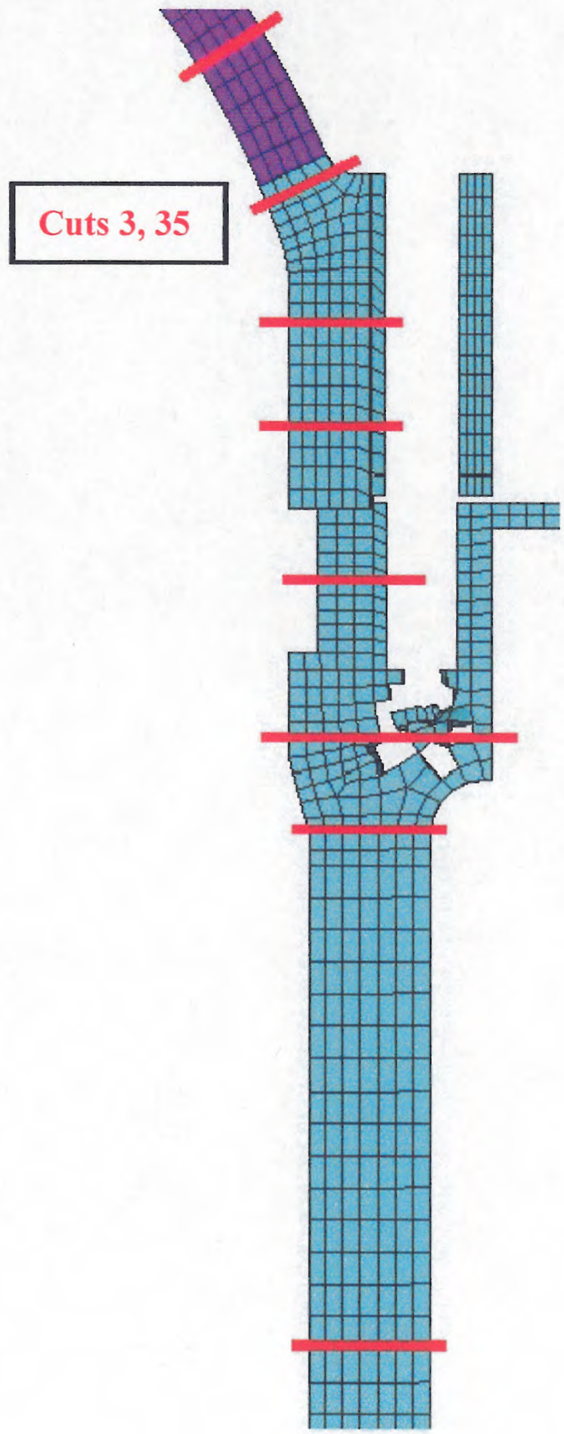


Figure E-4 Upper Head/Flange Region Stress Cuts

E.3 FRACTURE MECHANICS ANALYSIS

The fracture mechanics evaluation was performed using the approach suggested in ASME Section XI, Appendix G (Reference E-1). A semi-elliptic surface flaw was postulated to exist in the highest stressed region, which is at the outside surface of the closure head at the flange to head weld region shown in Figure E-4.

E.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). In most cases, the stress intensity factor for the fracture mechanics calculations utilized a representation of the actual stress profile rather than a linearization. The stress profile was represented by a 4th order polynomial:

$$\sigma = \sigma_0 + \sigma_1 \left(\frac{x}{t}\right) + \sigma_2 \left(\frac{x}{t}\right)^2 + \sigma_3 \left(\frac{x}{t}\right)^3 + \sigma_4 \left(\frac{x}{t}\right)^4 \qquad \text{(E-1)}$$

where:

x	=	the coordinate distance into the wall, in.
t	=	thickness of the wall, in.
σ	=	stress perpendicular to the plane of the crack, ksi
σ_i	=	coefficients of the curve fit (i = 0, 1, 2, 3, 4)

For the surface flaw with length six times its depth, the stress intensity factor expression of API-579 (Reference E-2) was used. The stress intensity factor K_I can be calculated anywhere along the crack front. The point of maximum crack depth is represented by φ =90, 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 as a function of the angular location around the crack front (φ). The units of K_I are ksi$\sqrt{\text{in}}$.

$$K_I = \left[\frac{\pi a}{Q}\right]^{0.5} \sum_{j=0}^{4} G_j\ (a/c, a/t, t/R, \varphi)\ \sigma_j \left(\frac{a}{t}\right)^j \qquad \text{(E-2)}$$

The boundary correction factors G_0, G_1, G_2, G_3,and G_4 are obtained by the procedure outlined in Reference E-2. The dimension "a" is the crack depth, "c" is the crack half length, "t" is the wall thickness, "R" is the inside radius, and "Q" is the flaw shape factor, which can be approximated by $Q = 1 + 1.464\ (a/c)^{1.65}$.

It should be noted that the stress intensity factors calculated based on API-579 (Reference E-2) are very similar to the stress intensity factors calculated in Section 3.0 of this report using Raju/Newman.

E.3.2 Fracture Toughness

The values of RT_{NDT} for the closure head/flange regions of the Byron and Braidwood units are shown in Table 3-1. The limiting RT_{NDT} value for Byron Unit 1 is [ ]^{a,c,e}; whereas, the limiting RT_{NDT} value for Byron Unit 2 and Braidwood Units 1 and 2 is [ ]^{a,c,e}. These limiting values are used for the integrity evaluations discussed in Section E.5.

The RT_{NDT} values are used in the determination of the fracture toughness of the material, as discussed in Section 3 of the main body of this report. 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 (Reference E-3). This value is consistent with general practice in such evaluations and it was used in Revision 0 of this report.

E.4 STRESS EVALUATION

The first step in evaluation of the closure head/flange region is to examine the stresses. The stresses that 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 boltup as well as during the entire heatup and cooldown process are summarized in Appendix F. Boltup is the key condition to review here, in comparison with the heatup and cooldown operation, since the flange requirement applies to boltup conditions. No other transients result in stresses in this region at low temperatures.

Boltup, heatup, and cooldown stresses were determined based on two three-dimensional FEA models of the Byron and Braidwood closure head/vessel flange region. The first model accounts for a missing stud while the second model has all studs present. The results from the finite element stress analyses in Appendix F demonstrate that for the missing stud condition, the boltup stresses in the head flange, which is the most limiting location for the evaluation of boltup requirements, were reduced in the vicinity of the missing stud. The boltup stresses in the head flange transition near the missing stud are reduced because there is less total preload on the bolted joint. The stresses remain largely unaffected away from the missing stud. Therefore, the results from the finite element stress analysis at the head flange demonstrated that, for the missing stud condition, the boltup stresses were less than or equal to the boltup stresses with all 54 studs in place. This is a key conclusion, because it demonstrates that the limiting case for boltup is with all studs in place.

During the course of the FEA evaluation it was discovered that the boltup stresses at the closure head flange using the three-dimensional FEA model with all studs are slightly more limiting than the original stresses shown in Table 4-1 of this report. The newer analysis is based on a fully three-dimensional, plant specific finite element model, while the original evaluation in Revision 0 of this report utilized a generic two-dimensional finite element model. The integrity of the reactor vessel closure head/vessel flange region was re-evaluated using the newer three-dimensional FEA model stresses to reconfirm the validity of the original conclusion that the flange requirements of 10 CFR 50, Appendix G can be eliminated from the P-T limit curves.

Tables E-1 and E-2 provide a comparison of the boltup stresses at the limiting locations for the all studs intact condition and the missing stud condition, respectively. It can be seen that the stresses at boltup are mostly bending, with a very small membrane stress. Based on Tables E-1 and E-2, it is observed that the boltup stresses for the missing stud case are lower at the outside surface than the case with all studs. The outside surface during the boltup condition is the limiting location in the fracture mechanics evaluation. The heatup and cooldown stresses are shown in Appendix F of this report.

Table E-1 Stress Distributions at the Closure Flange Region using Three-Dimensional FEA Model – All Studs Present

Distance (x/t)	Limiting Axial Boltup Stress (Cut 3) (ksi)	Limiting Hoop Boltup Stress (Cut 3) (ksi)
0 (ID)	-22.012	0.795
0.1	-17.217	2.341
0.2	-12.527	3.858
0.3	-8.534	5.285
0.4	-4.573	6.700
0.5	-0.609	8.137
0.6	3.356	9.573
0.7	8.402	11.158
0.8	13.510	12.759
0.9	21.622	14.714
1.0 (OD)	29.980	16.715

Table E-2 Stress Distributions at the Closure Flange Region using Three-Dimensional FEA Model – One Missing Stud

Distance (x/t)	Limiting Axial Boltup Stress (Cut 35) (ksi)	Limiting Hoop Boltup Stress (Cut 35) (ksi)
0 (ID)	-18.145	0.328
0.1	-14.123	1.630
0.2	-10.114	2.929
0.3	-6.751	4.130
0.4	-3.392	5.329
0.5	-0.017	6.543
0.6	3.358	7.757
0.7	7.708	9.106
0.8	12.066	10.456
0.9	19.203	12.136
1.0 (OD)	26.368	13.820

E.5 SUMMARY AND CONCLUSIONS

The original evaluation in Revision 0 of this report concluded that the 10 CFR 50, Appendix G flange requirement could be eliminated. The purpose of this revision is to confirm that the flange requirements can be eliminated from the P-T limit curves for Byron and Braidwood Units 1 and 2 using a fracture mechanics evaluation based on plant specific three-dimensional finite element models.

To investigate the impact of a stud out of service, two different three-dimensional finite element models were used to determine stresses for the boltup, heatup, and cooldown conditions. One model has all studs intact while the other model represents the case with a missing stud. It was determined that the highest stresses in the reactor vessel closure head/vessel flange region were those near the flange to torus weld region. The results from the finite element stress analysis at the flange to torus weld region demonstrated that, with the missing stud, the boltup stresses were less than the boltup stresses with all 54 studs in place at the region of concern.

Additionally, the stresses from the three-dimensional finite element model with all 54 studs intact were determined to be slightly higher than the stresses used in Revision 0 of this report. The new three-dimensional finite element model stresses with all 54 studs in place were therefore used to reaffirm the flange requirements for Byron and Braidwood Units 1 and 2 with all studs intact.

Fracture mechanics evaluations were performed using the stress results based on the two separate models considering one missing closure stud as well as the case with all studs intact. Again, boltup is the limiting condition to consider for the fracture mechanics evaluation since the flange requirement applies to boltup conditions. Additionally, the fracture mechanics results are less limiting for the heatup and cooldown transients as compared to the boltup condition because the margin between the stress intensity factors and the material fracture toughness increases due to higher temperatures during the heatup and cooldown conditions. The conclusions for the case with the missing stud are summarized in Section E.5.1. Similarly, the conclusions for the case with all studs intact are summarized in Section E.5.2.

E.5.1 Missing Stud Results

The fracture mechanics evaluation contained in Revision 0 of this report determined that boltup could occur at 60°F and the 10 CFR 50, Appendix G flange requirement was not necessary to take into account during the normal heatup or cooldown process for Byron and Braidwood Units 1 and 2. To reaffirm this conclusion accounting for a single missing stud, stress intensity factors were determined as a function of postulated flaw depth for all cut locations using the three-dimensional FEA stresses based on a model with a single missing stud. It was determined that the limiting location for the fracture mechanics evaluation is at the closure head to flange region (Cut 35) for boltup conditions and the limiting flaw configuration is for a postulated outside surface circumferential flaw.

The circumferential outside surface flaw stress intensity factors for several times during heatup and cooldown, as well as boltup, are shown in Figure E-5 for Cut 35. A factor of 2 was added to the stress intensity factors, as required by ASME Section XI, Appendix G, and the modified stress intensity factors were compared to the material fracture toughness calculated at the boltup temperature. The boltup temperature for a typical PWR is 60°F. For Byron Unit 2 and Braidwood Units 1 and 2, considering that the limiting RT_{NDT} is [ ]^{a,c,e} (from Table 3-1 for Byron Unit 2 and Braidwood Units 1 and 2), the

ASME K_{Ic} reference toughness value is [$]^{a,c,e}$. Based on the fracture mechanics evaluation, it was determined that a flaw larger than 0.675 inch (a/t = 0.10) is needed before the flange integrity margins would be compromised. For Byron Unit 1, considering that the limiting RT_{NDT} is [$]^{a,c,e}$ (from Table 3-1 for Byron Unit 1), the ASME K_{Ic} reference toughness value is [$]^{a,c,e}$. Based on the fracture mechanics evaluation, it was determined that a flaw as large as 0.43 inch (a/t = 0.065) is needed before the flange integrity margins would be compromised.

The flaw size of 0.43 inch was based on the structural integrity of the vessel flange according to the fracture toughness of the material (K_{Ic}) per Appendix G of Section XI of the ASME Code. Determining fracture toughness of a material using K_{Ic} is known to be conservative since K_{Ic} is the lower bound curve of fracture toughness test data of representative material. Based on Appendix A, and Figures 5-1 and 5-2, the probability of detecting and rejecting a 0.43 inch flaw in the material is approximately [$]^{a,c,e}$. This flaw size is therefore well within the detection capability of inservice examinations. Additionally, there are no recordable indications found at this region to date for Byron and Braidwood Units 1 and 2. Therefore, since there is a [$]^{a,c,e}$ probability of detecting the flaw, and no indications have ever been detected in the limiting region, the postulated flaw size of 0.43 inch is acceptable in the fracture mechanics evaluation.

The stress analysis and subsequent fracture mechanics evaluation for the missing stud case determined that for the boltup condition, the all studs intact case is more limiting. Additionally, the results of the missing stud case are in agreement with the conclusions determined in Revision 0 of this report. Specifically, the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves. **Thus, the conclusions reached in Revision 0 of this report remain valid for Byron and Braidwood Units 1 and 2 with the missing closure stud.**

E.5.2 All Studs Intact Results

The all studs intact case was developed based on a plant specific three-dimensional finite element model as a baseline case to compare with the original two-dimensional FEA model in Revision 0 of the WCAP report. Due to improved 3-D finite element modeling capability along with the use of plant specific geometry, the baseline FEM provided slightly higher boltup stresses as compared to the original stress analysis that was performed in Revision 0. Thus, the purpose for performing this case was to reaffirm the original conclusion of this report that boltup could occur at 60°F and the 10 CFR 50, Appendix G flange requirement can be eliminated from the P-T limit curves for Byron and Braidwood Units 1 and 2.

The limiting location for the fracture mechanics evaluation is still at the closure head to flange region for boltup conditions and the limiting flaw configuration is for a postulated outside surface circumferential flaw. The stress intensity factors for several time steps during heatup and cooldown, as well as boltup, are shown in Figure E-6 at Cut 3, which is the limiting location in the region for boltup. A factor of 2 was added to the stress intensity factors, as required by ASME Section XI, Appendix G, and the modified stress intensity factors were compared to the material fracture toughness calculated at the boltup temperature. The boltup temperature for a typical PWR is 60°F; therefore, considering that the limiting RT_{NDT} is [$]^{a,c,e}$ (from Table 3-1), the ASME K_{Ic} reference toughness value is [$]^{a,c,e}$. Based on the fracture mechanics evaluation, it was determined that a flaw as large as 0.32 inch (a/t = 0.047) is needed before the flange integrity margin would be compromised.

A flaw size of 0.32 inch is determined to be acceptable for the continued elimination of the flange requirement in the P-T limit curves. The flaw size of 0.32 inch was based on the structural integrity of the vessel flange according to the fracture toughness of the material (K_{Ic}) per Appendix G of Section XI of the ASME Code. Determining fracture toughness of a material using K_{Ic} is known to be conservative since K_{Ic} is the lower bound curve of fracture toughness test data of representative material. Based on Appendix A, and Figures 5-1 and 5-2, the probability of detecting and rejecting a 0.32 inch flaw in the material is approximately [ ]^{a,c,e}. This flaw size is therefore well within the detection capability of inservice examinations. Additionally, there are no recordable indications found at this region to date for Byron and Braidwood Units 1 and 2. Therefore, since there is a [ ]^{a,c,e} probability of detecting the flaw, and no indications have ever been detected in the limiting region, the postulated flaw size of 0.32 inch is acceptable in the fracture mechanics evaluation.

Thus, the FEA analysis with all 54 studs is representative and limiting for Byron and Braidwood Units 1 and 2, even with a closure head stud out of service, and provides acceptable fracture mechanics results for boltup, heatup, and cooldown conditions to demonstrate structural integrity of the reactor vessel. **Therefore, the conclusions determined in Revision 0 of this report have been confirmed, and the flange requirements of 10 CFR 50, Appendix G remain eliminated from the P-T limit curves for the Byron and Braidwood Units 1 and 2.**

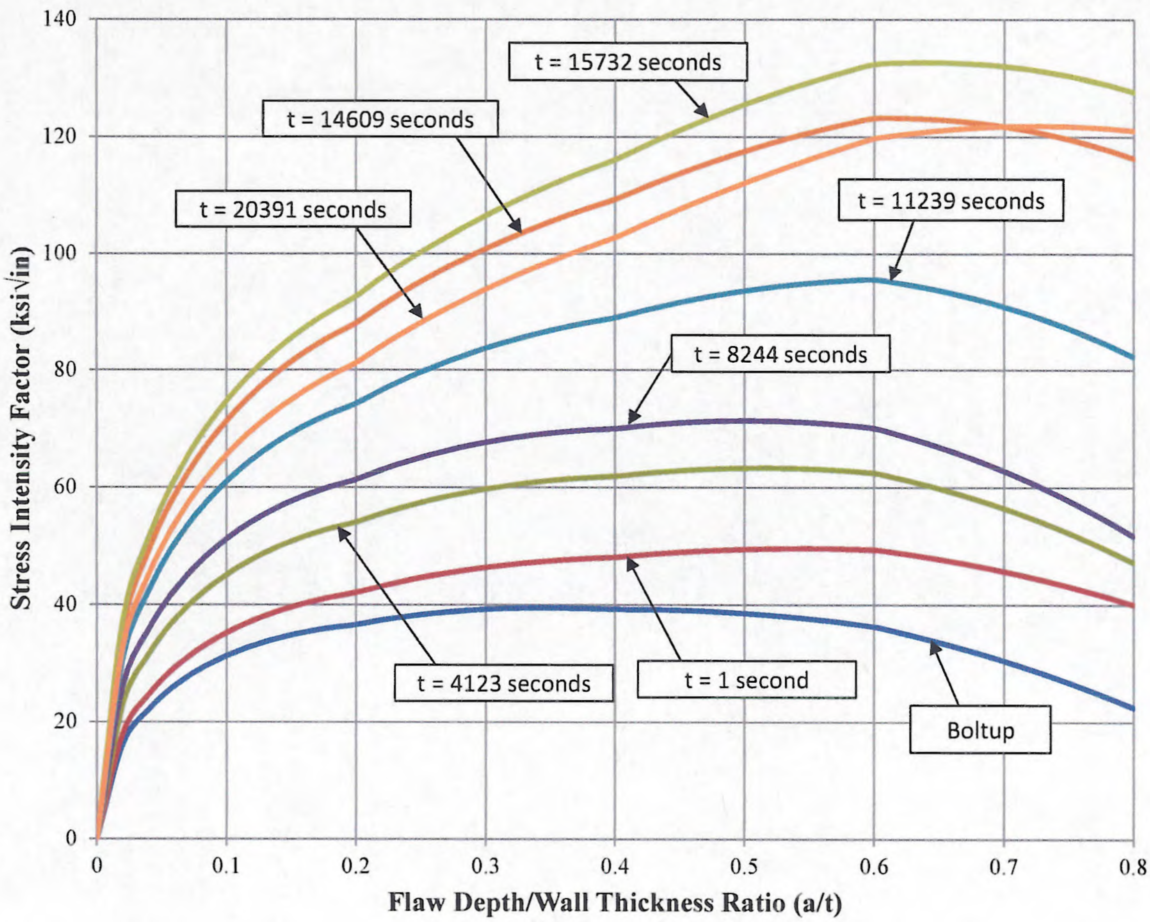


Figure E-5 **Crack Driving Force as a Function of Flaw Size: Circumferential Outside Surface Flaw in the Torus to Flange Region Weld with One Missing Stud (Flaw Length/Flaw Depth =6)**

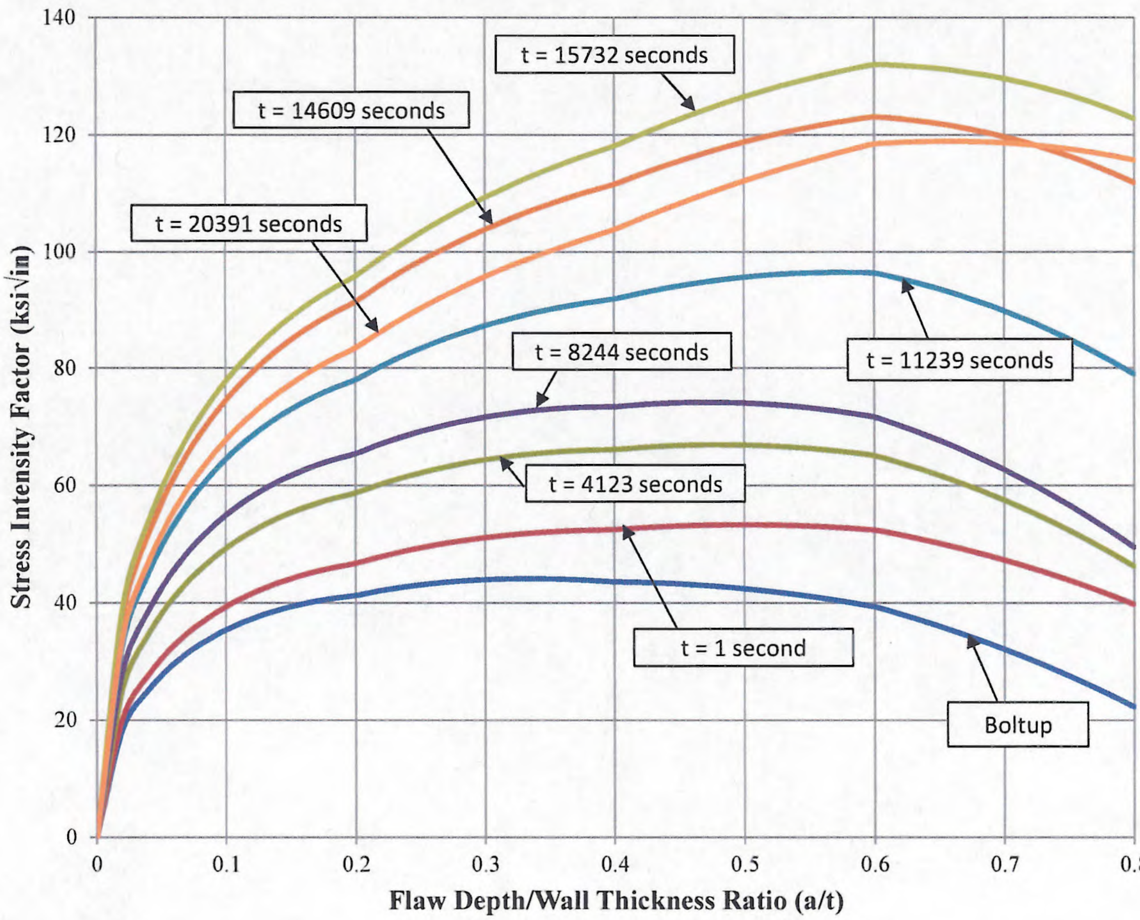


Figure E-6 Crack Driving Force as a Function of Flaw Size: Circumferential Outside Surface Flaw in the Torus to Flange Region Weld with All Studs Intact (Flaw Length/Flaw Depth =6)

E.6 REFERENCES

E-1 Appendix G to the 1998 through 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1.

E-2 American Petroleum Institute, API 579-1/ASME FFS-1 (API 579 Second Edition), "Fitness-For-Service," June 2007.

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

E-4 Design Information Transmittal DIT-BRW-2014-0059/BYR14-051, Revision 0, Braidwood Unit 2 and Byron Unit 2, September 2014.

APPENDIX F

STRESS DISTRIBUTIONS IN THE CLOSURE HEAD REGION FOR PLANT SPECIFIC THREE-DIMENSIONAL FINITE ELEMENT MODELS

F.1 STRESS ANALYSIS

[

]a,c,e

F.1.1 Mechanical Boundary Conditions

[

]a,c,e

F.1.2 Thermal Boundary Conditions

[

]a,c,e

[

]a,c,e

F.1.3 Stress Results

[

]a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a,c,e

Table F-1 Stress for Upper Head to Flange Transition Region from Three-Dimensional Finite Element Model

a.c.e
