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10 CFR 50.12
10 CFR 50.90

RS-05-103

October 3, 2005

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

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 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"

References:

(1) Letter from M. Chawla (USNRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Exemption from the Requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated August 8, 2001

(2) Letter from L. B. Marsh (USNRC) to K. W. Singer (Tennessee Valley Authority), "Sequoyah Nuclear Plant, Units 1 and 2, Exemption from the Requirements of 10 CFR Part 50, Appendix G," dated July 7, 2004

(3) Letter from C. Gratton (USNRC) to D. E. Grissette (Southern Nuclear Operating Company, Inc.), "Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Exemption and Amendments RE: Request to Revise Technical Specifications and Pressure Temperature Limits Report and Relocate the Cold Overpressure Protection System (COPS) Arming Temperature," dated March 28, 2005

(4) Technical Specification Task Force (TSTF) change traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," Revision 0

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment would (1) revise the TS 1.1, "Definitions," description of the Pressure

and Temperature Limits Report (PTLR) by deleting reference to specifications containing limits in the PTLR; (2) revise the administrative controls TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," by requiring the NRC approval documents to be identified by date and Topical Reports to be identified by number and title in accordance with Reference 4; and (3) add Westinghouse Electric Company, LLC, report WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," to the list of analytical methods provided in TS 5.6.6.

In support of the proposed amendment, and in accordance with 10 CFR 50.12, “Specific exemptions,” EGC is also requesting an exemption from 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.” Specifically, the requested exemption will allow the use of WCAP-16143, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,” dated November 2003 in 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).

EGC has received approval to use the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," which provides higher allowable pressure limits through the entire range of operating temperatures (i.e., Reference 1). The metal temperature requirement of the RPV closure head flange specified in 10 CFR 50, Appendix G; however, restricts the allowable pressure at temperatures below the reference nil-ductility transition temperature (RT_{NDT})+120°F. The continued operation with these P-T limits, without the relief provided by WCAP-16143, would unnecessarily restrict the P-T operating window for Byron Station Units, 1 and 2, and Braidwood Station, Units 1 and 2. The methodology in WCAP-16143 eliminates the minimum temperature requirement for the highly stressed region of the closure flange for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic pressure. EGC has determined that the use of WCAP-16143 will require NRC approval of an exemption to 10 CFR 50.60(a).

Similar exemption requests were previously granted for Sequoyah Nuclear Plant, Units 1 and 2 (i.e., Reference 2), and Vogtle Electric Generating Plant, Units 1 and 2 (i.e., Reference 3).

The attached amendment request and exemption request are subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachments 2-A and 2-B include the marked-up TS pages with the proposed changes indicated for Braidwood Station and Byron Station, respectively.

Attachments 3-A and 3-B include the associated revised TS pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

Attachment 4 provides the justification for the exemption request. As described in Attachment 1, EGC has concluded that for the reasons specified in the attachment, special circumstances, as defined in 10 CFR 50.12 exist; that the granting of the requested exemption will not present an undue risk to the health and safety of the public; and that the

granting of the requested exemption is consistent with the common defense and security. An environmental assessment of the requested exemption is provided in Attachment 4.

Attachment 5 provides a supporting affidavit signed by Westinghouse Electric Company LLC, the owner of proprietary information provided in Attachment 7. 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 in accordance with 10 CFR 2.390. Also enclosed are a Westinghouse authorization letter, CAW-05-2033, Proprietary Information Notice and Copyright Notice.

Attachment 6 provides a non-proprietary version of WCAP-16143. Attachment 7 provides the proprietary version of WCAP-16143.

The current P-T curves are valid until 17.6 effective full power years (EFPY) and 17.5 EFPY for Byron Station, Units 1 and 2, respectively, and until 16 EFPY for Braidwood Station, Units 1 and 2. Current estimates indicate that Byron Station, Unit 1, will reach 17.6 EFPY early in November 2006. The license amendment and exemption are desired prior to expiration of these curves to allow implementation of the new P-T curves. Therefore, EGC requests that approval of this license amendment and exemption request be granted prior to October 1, 2006, to allow sufficient time to develop and implement new P-T curves. Following NRC approval, the amendment will be implemented within 30 days. After license amendment implementation, the Braidwood Station and Byron Station PTLRs will be revised in a timeframe to assure issuance prior to exceeding each unit's current EFPY limit. The revised PTLRs will be submitted to the NRC in accordance with the requirements of TS 5.6.6.c.

The proposed amendment has been reviewed by the Braidwood Station and the 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.

EGC is notifying the State of Illinois of this application for a change to the TS by sending a copy of this letter and its attachments to the designated State Official.

Should you have any questions about this letter, please contact J. A. Bauer at (630) 657-2801.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 3rd day of October 2005.

Respectfully,

Keith R. Jury

Keith R. Jury
Director, Licensing and Regulatory Affairs

Attachment 1: Evaluation of Proposed Changes
Attachment 2-A: Markup of Proposed Technical Specifications Page Changes for Braidwood Station
Attachment 2-B: Markup of Proposed Technical Specifications Page Changes for Byron Station
Attachment 3-A: Revised Technical Specification Pages for Braidwood Station
Attachment 3-B: Revised Technical Specification Pages for Byron Station
Attachment 4: Justification for Exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation"
Attachment 5: Application for Withholding and Affidavit
Attachment 6: Non-proprietary Version of WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"
Attachment 7: Proprietary Version of WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"

ATTACHMENT 1
Evaluation of Proposed Changes

INDEX

1.0 DESCRIPTION

Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment would (1) revise the TS Section 1.1, "Definitions," description of the Pressure and Temperature Limits Report (PTLR) by deleting reference to specifications containing limits in the PTLR; (2) revise the administrative controls TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," by requiring the NRC approval documents to be identified by date and Topical Reports to be identified by number and title in accordance with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-419; "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," and (3) add Westinghouse Electric Company, LLC, WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," to the list of analytical methods provided in TS 5.6.6.

2.0 PROPOSED CHANGE

<u>TS Section 1.1</u>

TS Section 1.1 currently defines the PTLR as the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The proposed change deletes the reference to specifications containing limits in the PTLR. The TS Section 1.1 definition of PTLR has been revised to state:

> "The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6."

<u>TS 5.6.6</u>

TS 5.6.6.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, specifically those described in NRC letters dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," and August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2."

The proposed change reformats TS 5.6.6.b and adds WCAP-16143 as an analytical method approved by the NRC as follows:

"The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report,"

2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"

3. Westinghouse WCAP-16143, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," and

4. 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); and."

3.0 BACKGROUND

NRC Generic Letter 96-03, "Relocation of the Pressure and Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, allowed licensees to relocate the pressure temperature (P-T) limit curves and the Low Temperature Overpressure Protection (LTOP) System limits from the TS to a PTLR. The methodology used to determine the P-T limit curves and the LTOP System limit parameters is required to comply with the specific requirements of Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," be documented in an NRC approved topical report or an NRC approved plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment. The NRC approved the relocation of the Byron and Braidwood pressure and temperature limit curves and LTOP system limits in Reference 2.

TSTF-419 was submitted by the industry to remove duplication in the requirements of TS 5.6.6 and the definition of the PTLR, and to allow licensees to identify topical reports referenced in TS 5.6.6 by number and title only. A requirement was added in a reviewer's note to require that the complete citation in the PTLR for each topical report include the report number, title, revision, date, and any supplements. TSTF-419 received NRC approval on March 21, 2002.

EGC is also requesting an exemption from 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." Specifically, the requested exemption will allow the use of WCAP-16143 in calculating the reactor pressure vessel (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). The change in methodology must be approved by the NRC in a license amendment. This amendment

request, therefore, proposes to incorporate WCAP-16143 in the approved methodologies listed in TS 5.6.6.b.

4.0 TECHNICAL ANALYSIS

The TS Section 1.1 definition of PTLR identifies the specifications in which the pressure and temperature limits are addressed. TS 5.6.6.a requires that the individual specifications that address RCS pressure and temperature limits be referenced. The proposed changes to the definition eliminate duplication between the TS Section 1.1 definition of PTLR and TS 5.6.6. This change is consistent with TSTF-419.

The revision to ITS 5.6.6.b to allow the Topical Reports to be identified by number and title would allow EGC to use current Topical Reports to support limits in the PTLR without having to submit an amendment to the facility operating license each time the Topical Report is revised provided there is no change to the approved methodology. The PTLR would provide specific information identifying the particular Topical Reports used to determine the P-T limits or the LTOP System limits. TS 5.6.6.b provides the assurance that only the approved methodologies contained in the referenced Topical Reports will be used for the determination of the P-T limits or LTOP System limits. This proposed change is consistent with TSTF-419.

10 CFR 50, Appendix G, contains requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated reference nil-ductility transition temperature (RT_{NDT}) by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s, to ensure that appropriate margins would be maintained. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential RPV failure. Use of the methodology in WCAP-16143 eliminates the requirement that the minimum temperature of the highly stressed region of the closure flange exceed the unirradiated RT_{NDT} by 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic pressure.

The proposed methodology for the Byron Station and Braidwood Station P-T limits relies, in part, on American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1," which allows the use of K_{Ic} fracture toughness curve rather than the K_{Ia} curve. EGC received approval to use Code Case N-640 in Reference 3 and Reference 4. P-T limits developed using the K_{Ic} fracture toughness curve permit a much higher allowable pressure through the entire range of temperatures. The benefit is negated; however, at temperatures below RT_{NDT} +120°F because of the additional flange requirement of 10 CFR 50, Appendix G. 10 CFR 50, Appendix G, restricts the ability to raise the pressurizer power operated relief valve (PORV) setpoint as temperature rises by restricting the setpoint to the flange temperature limit rather than the ASME Code Case N-640 P-T limit.

Using the K_{Ic} fracture toughness, WCAP-16143 presents analyses that show that there is significant margin between the applied stress intensity factor at boltup and the material fracture toughness at cracks postulated to exist in the highest stress region of the closure head/flange region. The analyses also showed the boltup temperature requirement for Braidwood, Units 1 and 2, and Byron, Units 1 and 2, could be performed at 60°F or higher, easily justifying bolt up at ambient temperature. The results presented in WCAP-16143 demonstrate that the 10 CFR 50, Appendix G RPV closure head flange requirement can be eliminated and appropriate fracture margins are still maintained.

Applying WCAP-16143 methodology to the P-T curves will enhance overall plant safety by expanding the P-T operating window, especially in the region of low temperature operations. The two primary safety benefits that would be realized are a reduction in the potential challenges to the LTOP system and a reduction in the risk of damaging the reactor coolant pump seals. This will produce a significant improvement in plant safety by reducing the probability of an inadvertent reduction in reactor coolant inventory, and in easing the burden on the operators.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

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.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively.

The proposed amendment would (1) revise the TS 1.1, "Definitions," description of the Pressure and Temperature Limits Report (PTLR) by deleting reference to specifications containing limits in the PTLR; (2) revise the administrative controls TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," by requiring the NRC approval documents to be identified by date and Topical Reports to be identified by number and title in accordance with TSTF-419; and (3) add Westinghouse Electric

Company, LLC, WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," to the list of analytical methods provided in TS 5.6.6.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed TS changes by focusing on the three criteria set forth in 10 CFR 50.92 as discussed below:

Criteria

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

   Response: No.

   The proposed change to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P-T limits.

   The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine the pressure temperature (P-T) limits or Low Temperature Overpressure Protection (LTOP) System setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5.6.6.b requires that the analytical methods used to determine the P-T limits be those previously reviewed and approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59, "Changes, tests and experiments," and where required receive NRC review and approval.

   The use of WCAP-16143, following approval by the NRC, for generation of P-T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

   The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes 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 proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

   Based on the above discussion, the proposed changes do not involve an 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 proposed change to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P-T limits.

   The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P-T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5.6.6.b requires that the analytical methods used to determine the P-T limits be those previously reviewed and approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

   The use of WCAP-16143, following approval by the NRC, for generation of P-T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

   The proposed changes will allow the use of a new NRC-approved methodology for the calculation of P-T limits. However, the changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) and do not introduce a new mode of plant operation. Safety functions associated with P-T limits and LTOP setpoints will continue to function as previously assumed in accident analyses.

   Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed change to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P-T limits.

   The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P-T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5.6.6.b requires that the analytical methods used to determine the P-T limits be those previously reviewed and

approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The P-T limits provide assurance that the reactor pressure vessel is maintained. The use of WCAP-16143, following approval by the NRC, for generation of P-T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. Changes to setpoints at which protective actions are initiated that are allowed by the use of WCAP-16143 are evaluated in accordance with 10 CFR 50.59 and where required receive NRC review and approval. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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 as specified in 10 CFR 50.12, "Specific exemptions." EGC has determined that an exemption is required 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). The basis and justification for the exemption is provided as Attachment 4 to this submittal. Inclusion of WCAP-16143 in the TS following NRC approval continues to meet the requirements of 10 CFR 50.60 and 10 CFR 50.12.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change 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 would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for 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." Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter from G. F. Dick (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Issuance of Amendments," dated January 23, 1998

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

3. Letter from M. Chawla (USNRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Exemption from the Requirements of 10 CFR 50 Part 60 and Appendix G, for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated August 8, 2001

4. Letter from G. F. Dick (USNRC) to C. M. Crane (Exelon Generation Company, LLC), "Issuance of Amendments; Revised Pressure-Temperature Limits Methodology; Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated October 4, 2004

5. Technical Specification Task Force (TSTF) change traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," Revision 0

Attachment 2-A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

License Nos. NPF-72 and NPF-77

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”

Markup of Technical Specifications Pages

1.1-6
5.6-5

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates, and Power Operated Relief Valve (PORV) lift settings shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System";

the following documents:

1.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in NRC letter~~s~~ dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," ~~and~~ ¶

2. NRC letter dated

August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2"~~.~~ ~~and~~ ,

3. Westinghouse WCAP-16143, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," and

4. 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); and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Attachment 2-B

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

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"

Markup of Technical Specifications Pages

1.1-6
5.6-5

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates, and the pressurizer Power Operated Relief Valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Unit operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3586.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

5.6 Reporting Requirements

5.6.6 <u>Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)</u>

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates, and Power Operated Relief Valve (PORV) lift settings shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System";

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NRC letter~~s~~ dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," ~~and~~ ¶

2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2"~~.~~ ~~and~~ ,

3. Westinghouse WCAP-16143, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," and

4. 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); and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 <u>Post Accident Monitoring Report</u>

When a report is required by Condition C or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Attachment 3-A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457

License Nos. NPF-72 and NPF-77

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”

Revised Technical Specifications Pages

1.1-6
5.6-5

1.1 Definitions

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a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates, and Power Operated Relief Valve (PORV) lift settings shall be established and documented in the PTLR for the following:

   LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System";

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report;"

   2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"

   3. Westinghouse WCAP-16143, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," and

   4. 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); and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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Attachment 3-B

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

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"

Revised Technical Specifications Pages

1.1-6
5.6-5

1.1 Definitions

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b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report,"

2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50 Part 60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"

3. Westinghouse WCAP-16143, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," and

4. 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); and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Attachment 4

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

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"

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

Attachment 4

Justification for Exemption from 10 CFR 50.60

Specific Exemption Request

In accordance with 10 CFR 50.12, “Specific exemptions,” Exelon Generation Company, LLC (EGC) is requesting an exemption from 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.” Specifically, the requested exemption will allow the use of Westinghouse Electric Company Report, WCAP-16143, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,” in 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).

Basis For Exemption Request

The previous methodology used to generate the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Pressure Temperature Limits Report (PTLR), which contains the P-T curves and the low temperature overpressure protection (LTOP) system limits, is based on the methodology approved in a Safety Evaluation transmitted to EGC in a letter from R. A. Capra (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated January 21, 1998. This approved methodology includes the use of Westinghouse Electric Company Report, WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” Revision 1; American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code Case N-514, “Low Temperature Overpressure, Section XI, Division 1;” and ASME B&PV Code, Section XI, Appendix G, 1996 Addenda.

EGC received approval to use ASME B&PV Code Cases N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division1," and N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," in a letter from M. Chawla (USNRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Exemption from the Requirements of 10 CFR 50 Part 60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated August 8, 2001. EGC performed a change applying Code Cases N-640 and N-588 to the Byron Station and Braidwood Station PTLRs under 10 CFR 50.59, "Changes, tests, and experiments," to extend the applicability time of the Byron Station and Braidwood Station heat up and cool down curves by two effective full power years. The revised Byron Station and Braidwood Station PTLRs were submitted to the NRC in letters from S. E. Kuczynski (Exelon Generation Company, LLC) to USNRC, "Issuance of Reactor Coolant System Pressure and Temperature Limits Reports for Byron Station Units 1 and 2," dated November 12, 2004, and K. J. Polson (Exelon Generation Company, LLC) to USNRC, "Pressure and Temperature Limits Reports, Revision 3, Braidwood Station, Units 1 and 2," dated January 24, 2005, respectively.

Although Code Case N-640 provides higher allowable pressure limits through the entire range of operating temperatures, the metal temperature requirement of the RPV closure head flange specified in 10 CFR 50, Appendix G, restricts the allowable pressure at temperatures below the reference nil-ductility transition temperature (RT_{NDT})+120°F. The

continued operation with these P-T limits, without the relief provided by WCAP-16143, would unnecessarily restrict the P-T operating window for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. Approval of the requested exemption will allow the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, LTOP system setpoints to be established at 100% of the allowable pressure established by the P-T limit curve instead of the ASME B&PV Code Case N-514 setpoint of 110% of the allowable pressure.

10 CFR 50.12(a) Requirements

The requested exemption to allow use of WCAP-16143, in conjunction with ASME B&PV Code, Section XI, Appendix G, to determine the P-T limits meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the NRC may grant an exemption from requirements contained in 10 CFR 50 provided that the following is satisfied.

1. The requested exemption is authorized by law

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the NRC under 10 CFR 50.12.

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

   10 CFR 50, Appendix G, contains requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s, to ensure that appropriate margins would be maintained. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential RPV failure.

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

   Using the K_{Ic} fracture toughness, WCAP-16143 presents analyses that show that there is significant margin between the applied stress intensity factor at boltup and the material fracture toughness at cracks postulated to exist in the highest stress region of the closure head/flange region. The analyses also showed the boltup temperature

requirement for Braidwood, Units 1 and 2, and Byron, Units 1 and 2, could be performed at 60°F or higher, easily justifying bolt up at ambient temperature. The results presented in WCAP-16143 demonstrate that the 10 CFR 50, Appendix G RPV closure head flange requirement can be eliminated and appropriate fracture margins are still maintained.

Applying WCAP-16143 methodology to the P-T curves will enhance overall plant safety by expanding the P-T operating window, especially in the region of low temperature operations. The two primary safety benefits that would be realized are a reduction in the potential challenges to the LTOP system and a reduction in the risk of damaging the reactor coolant pump seals. This will produce a significant improvement in plant safety by reducing the probability of an inadvertent reduction in reactor coolant inventory, and in easing the burden on the operators.

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

   Implementation of this exemption request (i.e., WCAP-16143) will have no effect on the common defense and security.

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

   In accordance with 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) – Application of the regulation is not necessary to achieve the underlying purpose of the rule; and

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

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

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

   This requirement was originally based on concerns regarding the fracture margin in the closure flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of

hydrostatic test pressure were developed in the mid-1970s using the K_{Ia} fracture toughness to ensure that appropriate margins would be maintained.

WCAP-16143 presents analysis results using the K_{Ic} fracture toughness for the flange region which show that flange fracture margins are still maintained for boltup without the RT_{NDT}+120°F requirement. Use of the newly accepted K_{Ic} fracture toughness for flange boltup limits leads to the conclusion that the 10 CFR 50, Appendix G flange region requirement can be eliminated.

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

There are no known mechanisms of degradation for the RPV closure head flange region. Because the design fatigue usage factor for this region is less than 0.1, it may be concluded that flaws are unlikely to initiate in this region.

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

<u>10 CFR 50.12(a)(2)(iv)</u>

The RPV P-T operating window is defined by the P-T operating and test limits developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedures. EGC has received approval to use ASME B&PV Code Case N-640, which provides higher allowable pressure limits through the entire range of operating temperatures. The metal temperature requirement of the RPV closure head flange specified in 10 CFR 50, Appendix G; however, restricts the allowable pressure at temperatures below RT_{NDT}+120°F. The continued operation with these P-T limits, without the relief provided by WCAP-16143, would unnecessarily restrict the P-T operating window for Byron Station Units, 1 and 2, and Braidwood Station, Units 1 and 2. This operating window is defined by the area between the LTOP system setpoint and the minimum pressure for reactor coolant pump operation. The more restricted this space is, the greater the potential for inadvertent pressurizer PORV actuation or reactor coolant pump seal damage due to pump operation under conditions in which it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

The present methodology provides a more restrictive setpoint, which constitutes an unnecessary challenge that can be alleviated by application of WCAP-16143. Implementation of the P-T limits as allowed by WCAP-16143 does not reduce the margin of safety originally contemplated by either the NRC or ASME. Applying WCAP-16143 decreases the possibility of inadvertent pressurizer PORV actuation or reactor coolant pump seal damage and, thereby, is of benefit to the public health and safety.

Attachment 4

Justification for Exemption from 10 CFR 50.60

Environmental Assessment

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

The proposed action would grant an exemption from requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The requested exemption will specifically allow the use of Westinghouse Electric Company Report, WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," in calculating the 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).

The requested exemption is needed because utilization of WCAP-16143 will enhance overall plant safety by widening the P-T operating window, especially in the region of low temperature operations. The two primary safety benefits that would be realized are a reduction in the potential challenges to the low temperature overpressure protection system resulting in an inadvertent opening of a power operated relief valve and a reduction in the risk of damaging the reactor coolant pump seals due to pump operation under conditions in which it is difficult to maintain adequate seal differential pressure to ensure proper pump operation.

The principal alternative to the proposed action would be to deny the requested exemption and require adherence to the current 10 CFR 50.60 requirements. Denial of the exemption request would result in no change in environmental impacts.

Regarding alternative use of resources, granting the requested exemption will not involve the use of resources not previously considered in the Final Environmental Statements for Byron Station and Braidwood Station (i.e., NUREG-0848, "Final Environmental Statement related to the operation of Byron Station, Units 1 and 2," dated April 1982 and NUREG-1026, "Final Environmental Statement related to the operation of Braidwood Station, Units 1 and 2," dated June 1984).

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

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

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

Conclusion

Compliance with the specified requirement of 10 CFR 50.60(a) would unnecessarily restrict the P-T operating window and result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. Compliance with the 10 CFR 50, Appendix G RPV closure head flange limit would be inconsistent with the use of Code Case N-640. Code Case N-640 uses K_{Ic} fracture toughness methodology and the present RPV closure head flange limit is based on K_{Ia} fracture toughness methodology.

WCAP-16143 provides a valid basis for changing the RPV closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR 50, Appendix G requirement was issued. There is a further, positive benefit to the public health and safety by reducing potential inadvertent challenges to the pressurizer PORV and increasing the operating margin for the reactor coolant pump seals. Therefore, application of WCAP-16143 for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

Attachment 5

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

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"

Application for Withholding and Affidavit

Westinghouse

Westinghouse Electric Company
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U.S. Nuclear Regulatory Commission
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Our ref: CAW-05-2033

July 27, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16143-P, "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-05-2033 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 Corporation.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2033, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
J. J. DeBlasio

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. 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

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 29th day
of July, 2005

Lorraine M. Piplica
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, 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" 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 constitutes Westinghouse policy and provides 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.

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.

(iii) 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.

(iv) 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.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2" (Proprietary) dated September 2003, being transmitted by Exelon Corporation letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Beaver Valley Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Flow Induced vibration for EPU conditions.

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

(a) Provide documentation of the actual margins relative to the calculated stresses.

(b) Assist the customer in obtaining NRC approval by responding to NRC.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of Flow Induced Vibration (FIV) assessments and RCP Component Maximum Upset Stress Intensities at EPU conditions.

(b) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design and licensing a similar product.

(c) 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 calculations 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).

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Attachment 6

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

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"

Non-proprietary Version of WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"

Westinghouse Non-Proprietary Class 3

WCAP-16143-NP

November 2003

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

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16143-NP

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

Warren Bamford
K. Robert Hsu
Joseph F. Petsche

November 2003

Reviewer: ______________________
C. Y. Yang
Piping Analysis and Fracture Mechanics

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TABLE OF CONTENTS

1 INTRODUCTION

10 CFR Part 50, Appendix G contains requirements for pressure-temperature limits for the primary system, and requirements for the metal temperature of the closure head flange and vessel flange regions. The pressure-temperature limits are to be determined using the methodology of ASME Section XI, Appendix G, but the flange temperature requirements are specified in 10CFR50 Appendix G. This rule states that the metal temperature 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 N640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

Figure 1-1 illustrates the problem created by the flange requirements for a typical PWR heatup curve. It is easy to see that the heatup curve using K_{Ic} provides for a much higher allowable pressure through the entire range of temperatures. For this plant, however, the benefit is negated at temperatures below RT_{NDT} +120°F because of the flange requirement of 10 CFR Part 50, Appendix G. The flange requirement of 10 CFR 50 was originally developed using the K_{Ia} fracture toughness, and this report will show that use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated for Byron/Braidwood Units 1 and 2.

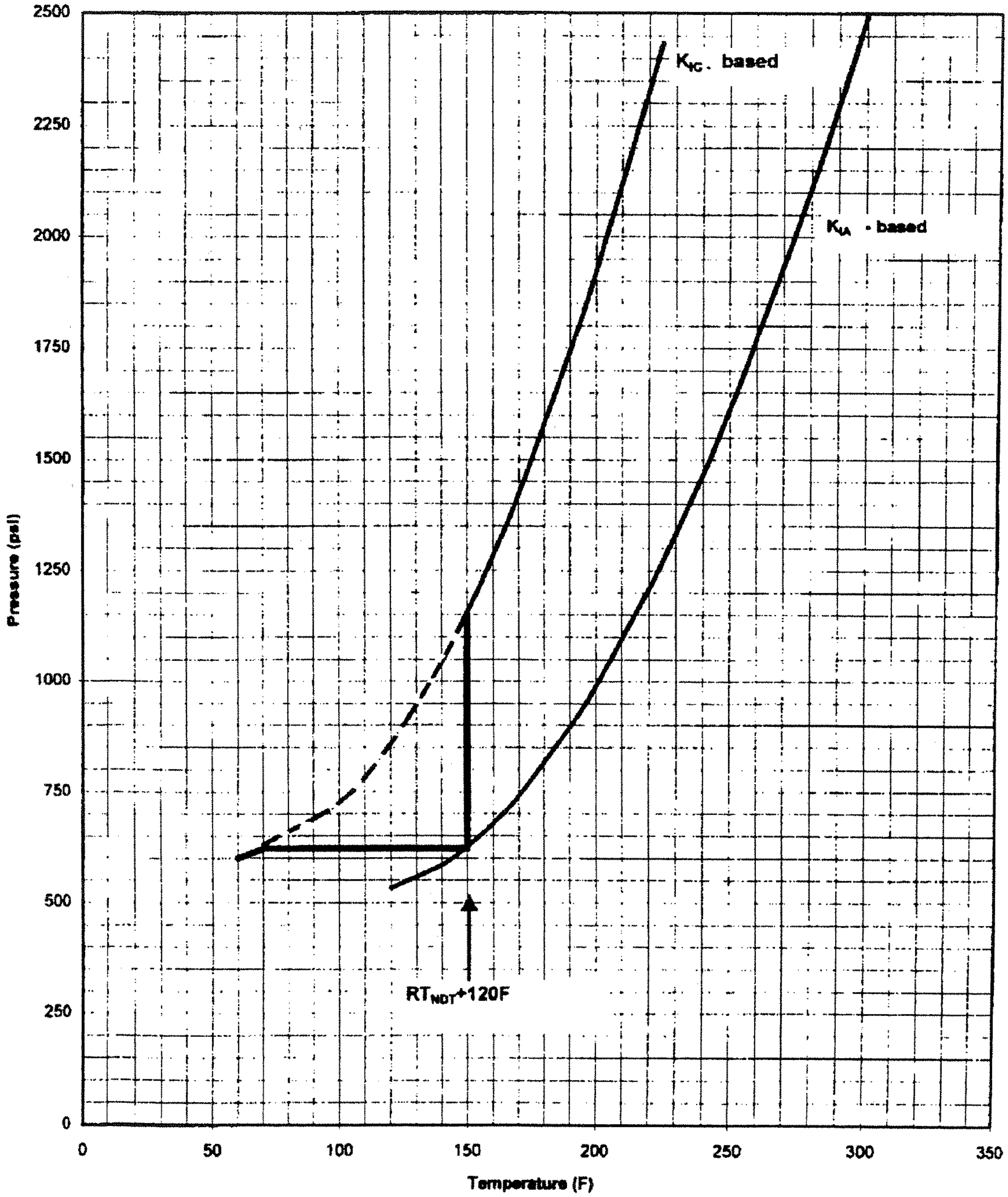


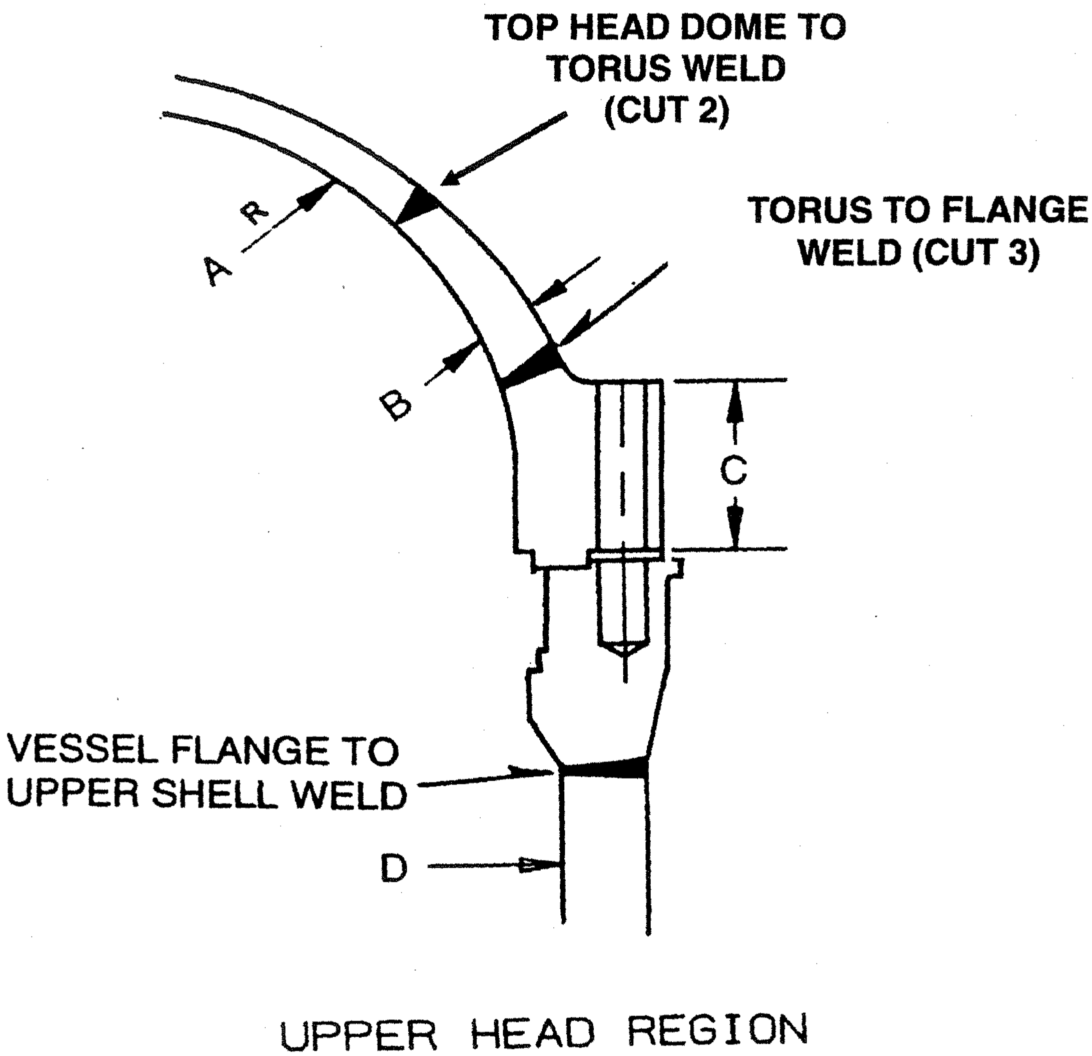
Figure 1-1 Illustration of the Impact of the Flange Requirement for a Typical PWR Plant

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:

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

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 $\phi = 0$, and this location was found to also be the point of maximum K_I for the cases considered here. The following expression is used for calculating K_I as a function of the angular location around the crack front (ϕ). The units of K_I are $\text{ksi}\sqrt{\text{in}}$.

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

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:

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

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

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

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 provides the background and technical basis 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 or determined from Charpy tests and drop weight tests [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.

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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:	$K_I = 24.84\ \text{ksi}\sqrt{\text{in}}$ (for a/t = 0.1)
End of Heatup:	$K_I = 49.58\ \text{ksi}\sqrt{\text{in}}$ (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 $K_{Ia} = 39.2\ \text{ksi}\sqrt{\text{in}}$ and $K_{Ic} = 50.9\ \text{ksi}\sqrt{\text{in}}$. 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 $\text{ksi}\sqrt{\text{in}}$, 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.

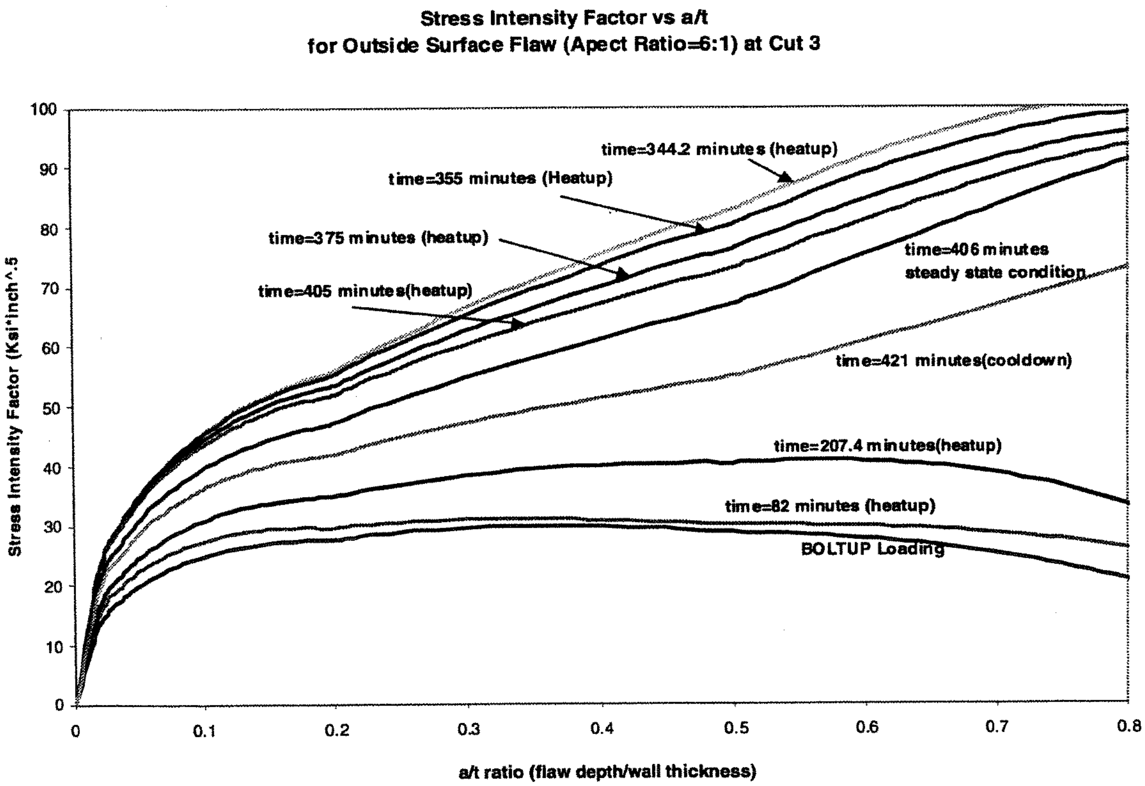

Figure 4-1 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Torus to Flange Region Weld for Byron/Braidwood Units 1 & 2)

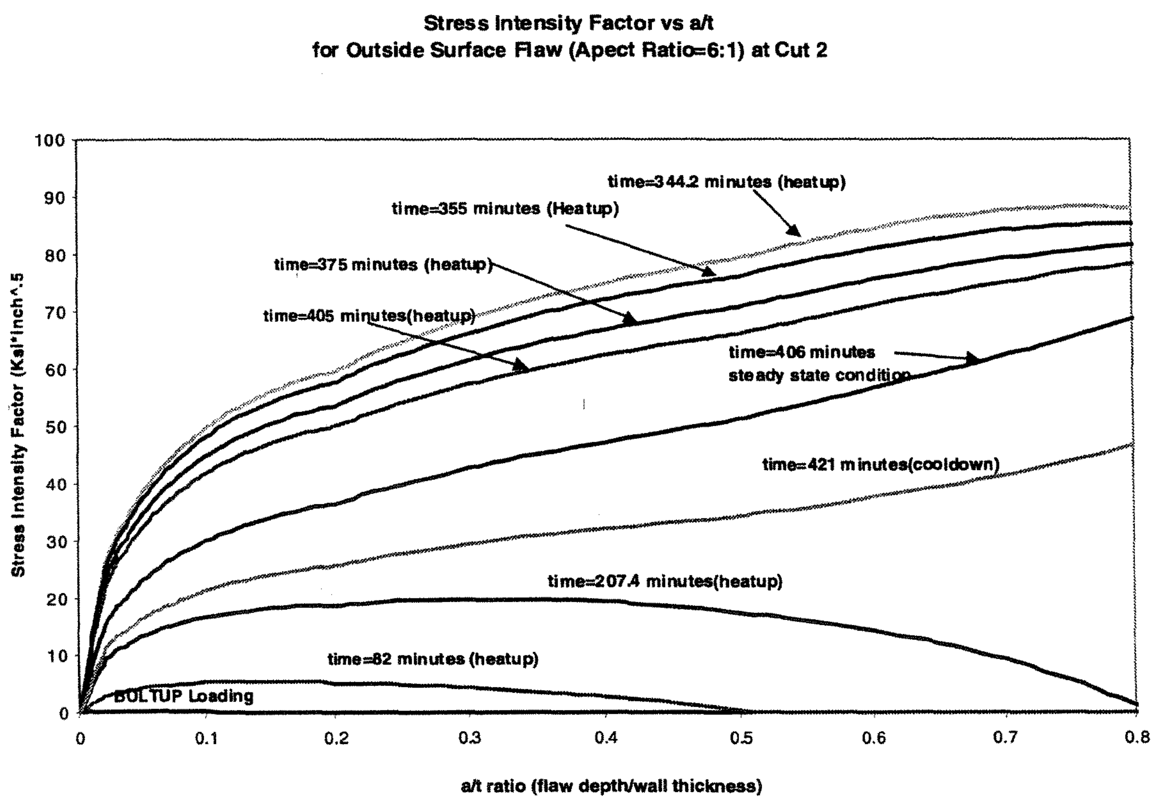

Figure 4-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Dome to Torus Region Weld for Byron/Braidwood Units 1 & 2)

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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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 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.

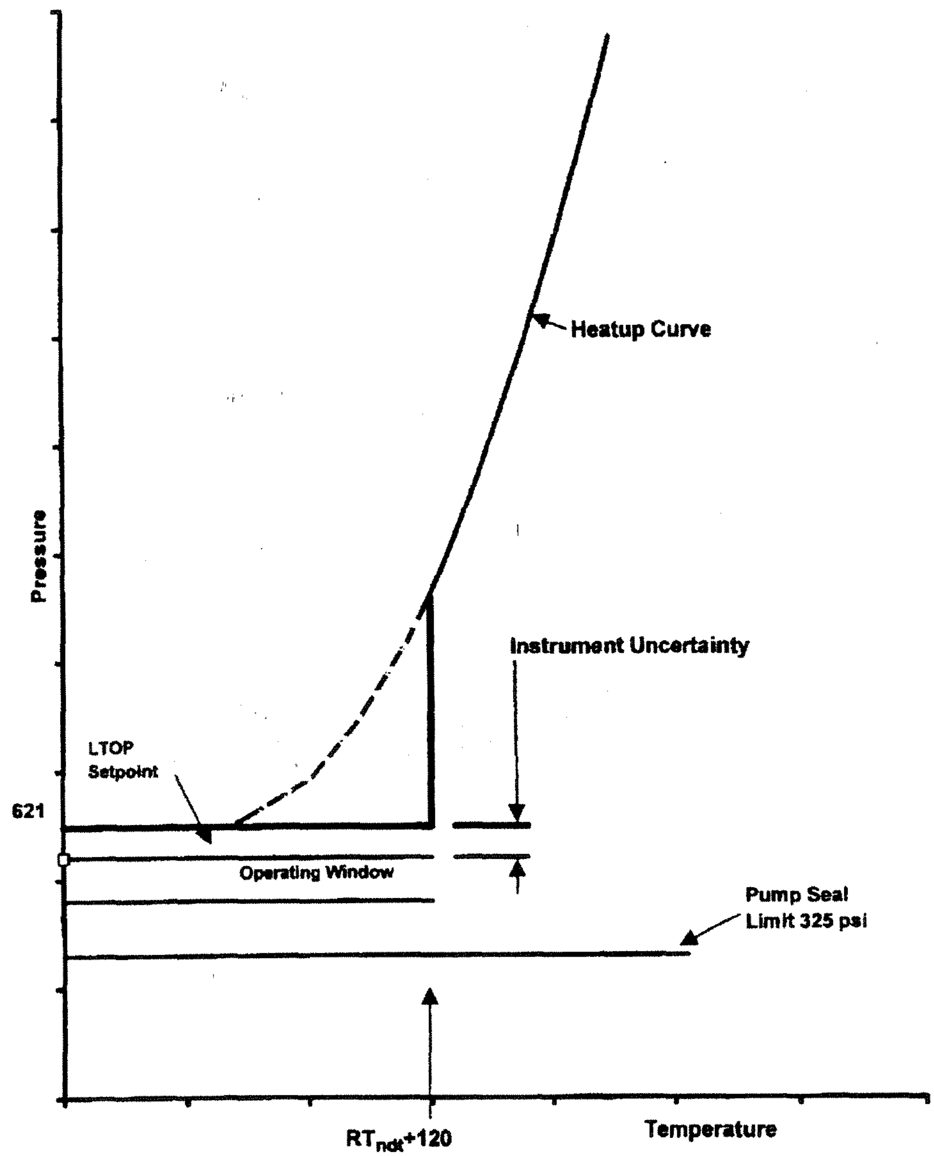

Figure 6-1 Illustration of the Flange Requirement and its Effect on the Operating Window for a Typical Heatup Curve

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 5P-5933 (using surv. capsule data)
LIMITING ART VALUES AT 12 EFPY: 1/4T, 70°F
3/4T, 60°F

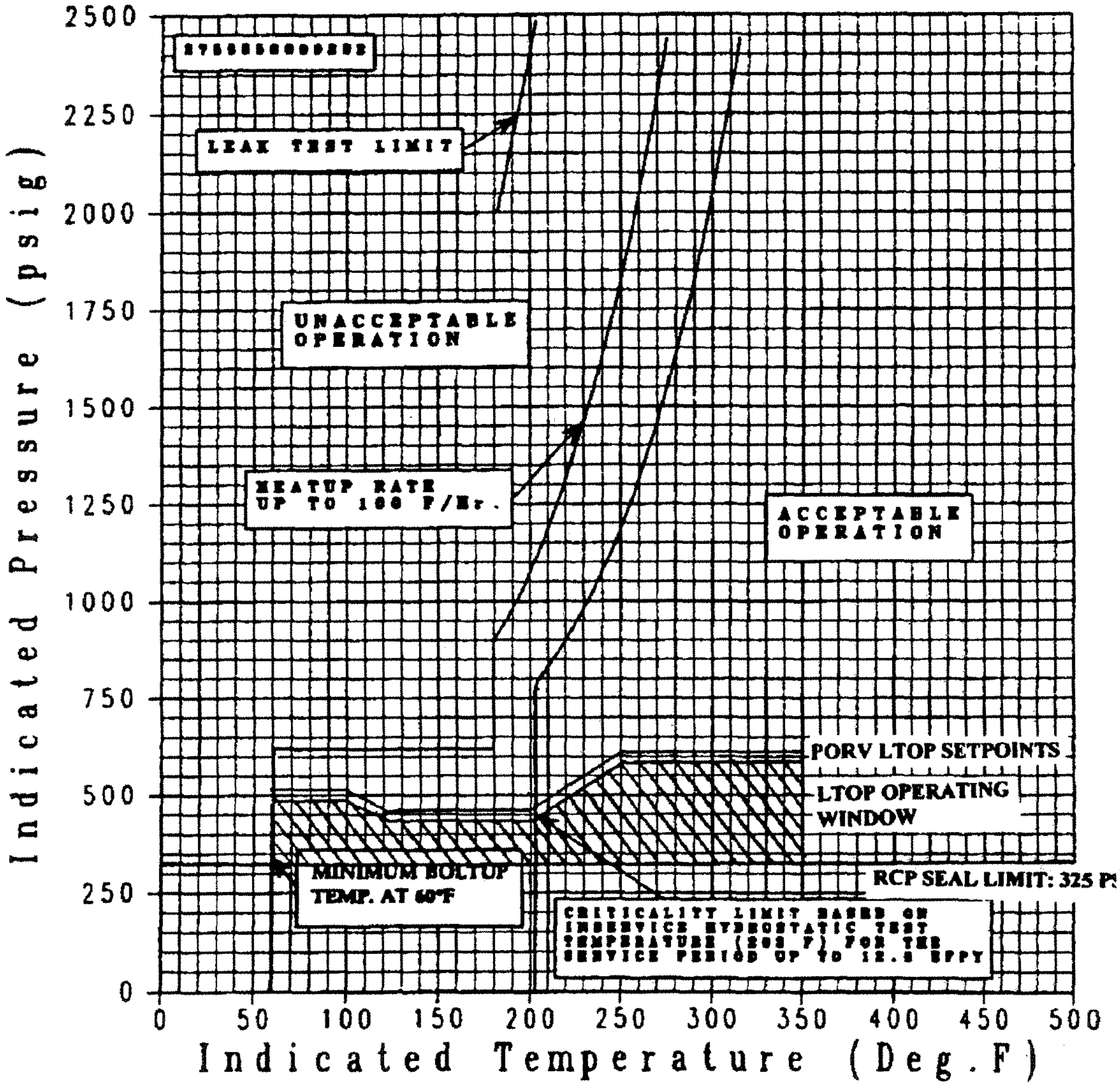

Figure 6-2 Illustration of the Actual Operating Window for Heatup of Byron Unit 1, a Low Copper Plant at 12 EFPY

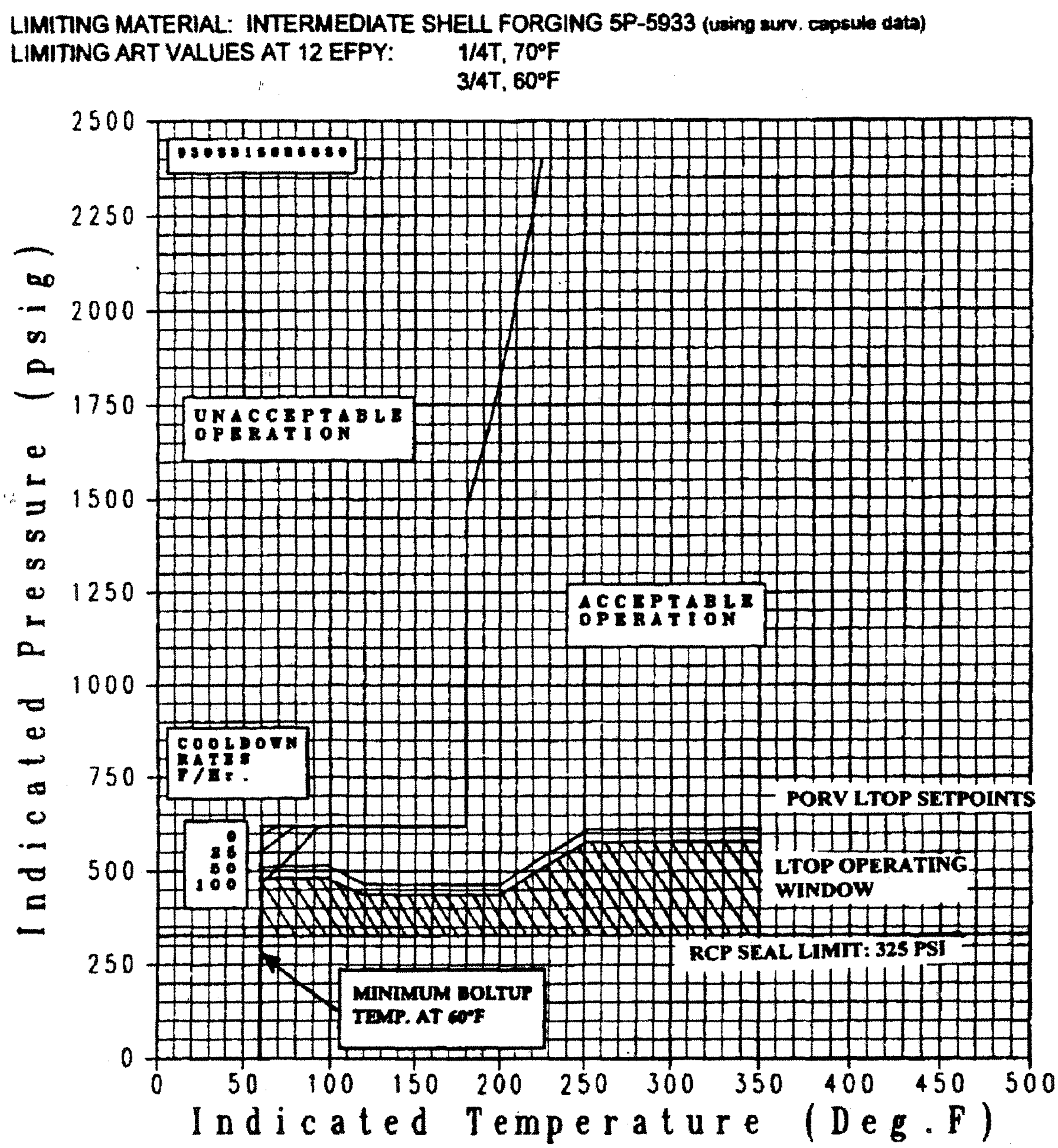

Figure 6-3 Illustration of the Actual Operating Window for Cooldown of Byron Unit 1, a Low Copper Plant at 12 EFPY

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APPENDIX A
REACTOR PRESSURE VESSEL INSPECTION RELIABILITY*

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EPRI

Charlotte NC

1 ABSTRACT

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

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3.2 COMBINED ID AND OD DETECTION

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

4 SIZING

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Figure 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 6 Sizing Error Surface Model

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Figure 7 Plan View of Sizing Error Surface Model

5 ACCEPTABILITY EVALUATION

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

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Figure 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

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APPENDIX C
STRESS DISTRIBUTIONS IN THE CLOSURE HEAD REGION

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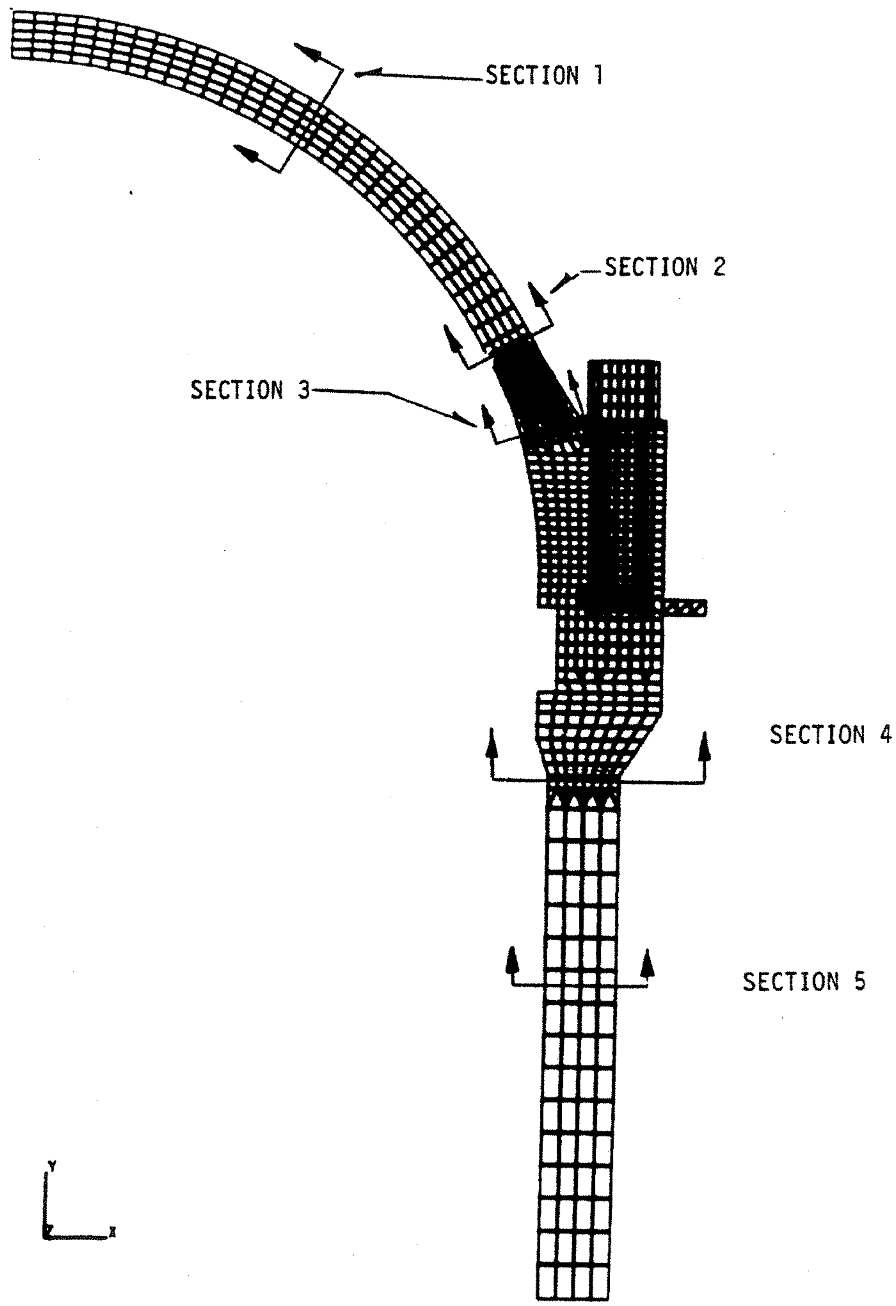

Figure C-1 Finite Element Model for Closure Head Region, Byron and Braidwood Units 1 and 2

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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 the magnetic particle method with no recordable indications.

The volumetric inspection, along with the surface inspection and the visual (VT-2) inspections performed every refuel outage provide reasonable assurance of 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 inspections have revealed no recordable indications and provide reasonable assurance of the continued structural integrity of this weld.

Volumetric exams. The approved inspection 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 side drilled

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.

Based upon review of past inspection data for the closure head flange region, it has been concluded that a 10% through-wall defect would have easily been detected and recorded.

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, RVHC-01)	Surface Exam (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 the 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.	This 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 exams 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, RVHC-01)	Surface Exam (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 the 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.	This examination was completed in April 2001on 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 exams 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 Exam (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 are based on the 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.	This 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 exams are 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 Exam (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 are based on the 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.	This 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 exams are 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.

Attachment 7

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. 50-454 and 50-455
License Nos. NPF-37 and NPF-66

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”

Proprietary Version of WCAP-16143, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2”