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10 CFR 50.90

RS-04-075

May 21, 2004

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: Request for a License Amendment to Incorporate Approved Pressure and Temperature Limits Report (PTLR) Methodology into Technical Specifications

Reference: Letter from U. S. NRC 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

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 revise TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," by adding the referenced letter as an acceptable method for determining reactor pressure vessel (RPV) pressure temperature (P-T) limits. The NRC approved the use of Code Cases N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," and N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1," for Braidwood Station, Units 1 and 2 and Byron Stations, Units 1 and 2 in the referenced letter.

The attached amendment request is 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.

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

Attachments 4-A through 4-D include reports developed by Westinghouse Electric Company (Westinghouse) that contain revised P-T limit curves for Braidwood Station and Byron Station. These reports are provided to demonstrate the results of applying Code Cases N-588 and N-640 to the currently approved methodology for developing the P-T limits. These reports also apply an additional allowance regarding the RPV closure flange, as discussed in Attachment 1, that has not been approved by the NRC. NRC approval of the reports is not being requested.

Attachment 5 provides a comparison of the 22 effective full power years (EFPY) data points in Attachments 4-A through 4-D to the data points of the existing P-T limit curves for Braidwood and Byron Stations. The 22 EFPY data points have been modified to restore the minimum flange temperature requirement of 10 CFR Appendix G Table 1 for each of the Braidwood and Byron Units.

Attachments 6-A and 6-B include P-T limit curves with revised applicability dates for Braidwood and Byron Stations, respectively. NRC approval of these curves is not being requested, since P-T limit curves are not part of the Braidwood and Byron Station TS.

The current Byron Station, Unit 1 P-T limit curves will reach the limiting reactor vessel neutron fluence on approximately November 3, 2004. Thus, we request approval of the proposed amendment by October 22, 2004. Once approved, the amendment will be implemented in a timeframe to accommodate Byron Station, Unit 1.

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 concerning this letter, please contact D. J. Chrzanowski at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 21, 2004

Kenneth A. Ainger
Kenneth A. Ainger
Manager, Licensing

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: Typed Pages for Technical Specification Changes for Braidwood Station
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Attachment 4-A: WCAP-15364, Revision 2, Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation
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Attachment 4-C: WCAP-15391, Revision 1, Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation
Attachment 4-D: WCAP-15392, Revision 2, Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation
Attachment 5: Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Attachment 6-A: Revised P-T Limit Curves for Braidwood Station, Units 1 and 2
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ATTACHMENT 1
Evaluation of Proposed Changes

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1.0 DESCRIPTION

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 revise TS 5.6.6, "Pressure Temperature Limits Report," by adding the referenced letter to the list of approved methods for determining reactor pressure vessel (RPV) pressure temperature (P-T) limits. The referenced letter documents NRC approval of the use of American Society of Mechanical Engineers (ASME) Code Cases N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," and N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1," for Braidwood and Byron Stations.

By adding this acknowledgement to our approved methodology, EGC would have the basis to perform a change to the Braidwood and Byron Station PTLRs under 10 CFR 50.59, "Changes, tests, and experiments," to extend the applicability time of the current Braidwood and Byron heat up and cool down curves by two effective full power years (EFPY). This would provide a reasonable limit that would allow continued safe operation while accommodating a review and approval schedule for a future licensing action. The future submittal request would provide a completely updated set of heat-up and cooldown curves and low temperature overpressure protection setpoints based on the application of ASME Code Case N-640, ASME Code Case N-588 and, an exemption to 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.c (minimum flange temperature requirements) as contained in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirement Evaluation for Byron/Braidwood Units 1 and 2. This revised methodology would provide for Braidwood and Byron Station Pressure and Temperature Limit Reports that would be effective for 32 EFPY.

We request approval of the proposed amendment by October 22, 2004. Once approved, the amendment will be implemented in a timeframe necessary to accommodate the current Byron Station, Unit 1 Pressure and Temperature Limit Report expiration date.

2.0 PROPOSED CHANGE

TS Section 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," part b, states the following.

> 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 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

The proposed change revises TS 5.6.6.b to state the following.

> 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 Part 50 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2"; and

3.0 BACKGROUND

Braidwood Station and Byron Station TS have adopted the provisions of NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," to include the pressure-temperature (P-T) limit curves in a PTLR controlled by the licensee. TS 5.6.6 requires that the P-T limits be developed using the specific NRC-approved methodology described in TS 5.6.6.b.

The current Braidwood and Byron Station P-T limit curves were developed using the methodology described in the January 21, 1998, letter referenced in TS 5.6.6.b, which involves the use of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1996 Addenda, the use of Code Case N-514, and use of the methodology in the 1996 ASME Code Section XI, Appendix G, Article G-2000.

The current Braidwood and Byron Station P-T limit curves are applicable for RPV exposures less than or equal to 14 effective full power years (EFPY) for Braidwood Station Units 1 and 2, 15.6 EFPY for Byron Station, Unit 1, and 15.5 EFPY for Byron Station, Unit 2. The first unit to reach the limiting exposure for the P-T curves is Byron Station, Unit 1, which is expected to reach 15.6 EFPY on approximately November 3, 2004.

In the referenced letter, the NRC granted Braidwood and Byron Stations an exemption from 10 CFR 50, Appendix G, "Fracture Toughness Requirements," to allow application of Code Cases N-640 and N-588 to determine the P-T limits. Code Case N-588 allows postulation of a circumferentially-oriented reference flaw as the limiting flaw in an RPV circumferential weld in lieu of an axially-oriented flaw required by the 1995 Edition (1996 Addenda) of ASME Section XI, Appendix G. Code Case N-640 allows use of the plane strain fracture toughness (K_{Ic}) curve in lieu of the crack arrest fracture toughness (K_{Ia}) curve as the lower bound for fracture toughness.

Braidwood and Byron Station TS Section 5.6.6 requires that the methods used to develop the P-T limit curves be those specifically listed in TS 5.6.6.b. Thus, the TS require revision to include the approval granted in the referenced letter.

4.0 TECHNICAL ANALYSIS

Use of Code Cases N-588 and N-640 has been determined to be acceptable for Braidwood and Byron Stations, as documented in the referenced letter. Thus, the addition of these methods to TS 5.6.6 represents an essentially administrative change to the TS.

Westinghouse Electric Company (Westinghouse) developed P-T limit curves for Braidwood and Byron Stations using the methods described in the referenced letter and, additionally, applying Code Cases N-588 and N-640. Westinghouse developed the curves using updated RPV fluence values developed from RPV material surveillance data. These curves are provided for information as Westinghouse reports for each unit at Braidwood and Byron Station in Attachments 4-A through 4-D. These reports are provided to demonstrate the results of applying Code Cases N-588 and N-640 to the currently approved methodology for developing the P-T limits. NRC approval of the reports is not being requested, since P-T limit curves are not part of the Braidwood and Byron Station TS.

To support future licensing actions, the P-T limit curves in Attachments 4-A though 4-D were developed by eliminating the requirement of 10 CFR 50 Appendix G (Table 1, Note 2) that the minimum temperature of the highly stressed region of the closure flange exceed the unirradiated reference nil-ductility transition temperature (RT_{NDT}) by 120°F for normal operation when the RPV pressure exceeds 20 percent of the pre-service hydrostatic pressure. The NRC has not approved the elimination of the requirement of Table 1, Note 2 for Braidwood and Byron Stations, and EGC is not requesting approval of the elimination of this requirement in these proposed changes. Attachment 5 provides a comparison of the 22 Effective Full Power Years (EFPY) data points in Attachments 4-A through 4-D to the data points of the existing P-T limit curves for Braidwood and Byron Stations. The 22 EFPY data points have been modified to restore the minimum flange temperature requirement of 10 CFR Appendix G Table 1 for each of the Braidwood and Byron Units.

Upon NRC approval of the proposed change to add the referenced letter to TS 5.6.6.b, Braidwood and Byron Stations will revise the current P-T limit curves by extending the current P-T limit curves an additional two EFPY, as provided in Attachments 6-A and 6-B, respectively. The P-T limit curves in Attachments 6-A and 6-B are applicable until RPV exposures of 16 effective EFPY for Braidwood Station Units 1 and 2, 17.6 EFPY for Byron Station, Unit 1, and 17.5 EFPY for Byron Station, Unit 2. These curves provide more conservative P-T limits (i.e., higher required minimum temperature for a given pressure) than the curves in Attachments 4-A through 4-D for 22 EFPY. Thus, the P-T limit curves provided in Attachments 6-A and 6-B are acceptable for an additional two EFPY. These curves are being provided for information and NRC approval of these P-T limit curves is not being requested. Braidwood and Byron Stations will submit the revised P-T limit curves as part of the PTLR in accordance with the requirements of TS 5.6.6.c.

In a future licensing action, EGC plans to request NRC approval to eliminate the requirement of 10 CFR 50 Appendix G Table 1, Note 2, for Braidwood and Byron Stations. This licensing action, once approved, would then allow use of the P-T limit curves contained in Attachments 4-A through 4-D.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

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 revise TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," by adding a reference to the use of Code Cases N-640 and N-588 as acceptable methods for determining reactor pressure vessel (RPV) pressure temperature (P-T) limits. The NRC has previously approved the use of Code Cases N-640 and N-588 for Braidwood and Byron Stations.

Criteria

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.

1. **The proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The use of Code Cases N-588 and N-640 has been approved for Braidwood and Byron Stations. The use of P-T limits based on these Code Cases will continue to ensure that the RPV integrity is maintained under all conditions. Thus there is no increase in the probability or consequences of an accident previously evaluated.

2. **The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The proposed change does not involve the use or installation of new equipment. No equipment will be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed change will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

   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. **The proposed TS change does not involve a significant reduction in a margin of safety.**

   The P-T limits provide assurance that RPV integrity is maintained. The use of Code Cases N-588 and N-640 has been previously approved by the NRC for Braidwood and Byron Stations and will continue to ensure that RPV integrity is maintained. Thus, there is no reduction in the margin of safety.

Based on the above discussions, it has been determined that the requested TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration under the standards 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 pressure boundary must meet the requirements of 10 CFR 50, Appendix G. 10 CFR 50.60(b) allows alternatives to the requirements as specified in 10 CFR 50.12, "Specific exemptions." In the referenced letter, the NRC granted an exemption to Braidwood and Byron Stations allowing the use of Code Cases N-588 and N-640 in lieu of the requirements of 10 CFR 50 Appendix G. Thus, inclusion of the NRC approval in the Braidwood and Byron Stations TS continues to meet the requirements of 10 CFR 50.60 and 10 CFR 50.12.

6.0 ENVIRONMENTAL CONSIDERATION

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

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

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

The proposed change does not result in an increase in power level, does not increase the production nor alter the flow path or method of disposal, of radioactive waste or byproducts; thus, there will be no change in the amounts of radiological effluents released offsite.

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

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

The proposed change will not result in any changes to the configuration of the facility. The proposed change will not cause a change in the level of controls or methodology used for the processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed amendment result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

7.0 REFERENCE

Letter from U. S. NRC 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

ATTACHMENT 2-A

Markup of Proposed Technical Specifications Page Changes

BRAIDWOOD STATION

<u>REVISED TS PAGE</u>

5.6-5

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";

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 Insert

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.

Insert

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

ATTACHMENT 2-B

Markup of Proposed Technical Specification Page Changes

BYRON STATION

<u>REVISED TS PAGE</u>

5.6-5

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";

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
S 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
Insert

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.

Insert

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

ATTACHMENT 3-A

Typed Pages

for

Technical Specification Changes

BRAIDWOOD STATION

REVISED TS PAGES

5.6-5

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";

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 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 Part 50 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2"; 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 3-B

Typed Pages

for

Technical Specification Changes

BYRON STATION

<u>REVISED TS PAGES</u>

5.6-5

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";

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 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 10CFR Part 50 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2"; 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-A

WCAP-15364, Revision 2

Braidwood Unit 1
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Non-Proprietary Class 3

WCAP-15364
Revision 2

November 2003

Braidwood Unit 1
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-15364, Revision 2

Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

November 2003

Approved:
J. A. Gresham, Manager
Reactor Component Design & Analysis

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: J. H. Ledger

Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially affects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15364 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15364 Rev. 0.

Note that only the heatup curves and associated data point tables in section 5 have changed. The cooldown curves and data points remain valid and were not changed.

Revision 2:

This revision was completed to incorporate an updated reference for the flange elimination from the PT limit curves. In the previous revision, WCAP-15315 was provided as justification for the flange notch elimination and it was changed to WCAP-16143-P in this revision. In addition to this change, the thermal stress intensity factors for the highest heatup and cooldown rates were added to this report in Appendix A.

EXECUTIVE SUMMARY

The purpose of this report is to document the generation of pressure-temperature limit curves for Braidwood Unit 1 for normal operation at 16, 22 and 32 EFPY using the latest methodologies (i.e. WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-16143-P). Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values. The 1/4T and 3/4T values are summarized in Tables 4-16 through 4-21 and were calculated using the circumferential weld WF-562 (Heat 442011) (limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 5P-7016 (limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all EFPY values evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-6.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Braidwood Station Unit No. 1 reactor pressure vessel (Specifically from lower shell forging [49D867/49C813]-1-1). Also supplied was a weldment made with non-copper coated weld wire using the automatic sub-arc welding process (Weld wire heat # 442011 Linde 80 flux, lot number 8061, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company (now known as Exelon Nuclear) contracted Westinghouse to analyze surveillance capsule W from the Braidwood Unit 1 reactor vessel. As a part of this analysis and the current uprating program, Westinghouse is to generated new heatup and cooldown curves for 16, 22 and 32 EFPY. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10 CFR Part 50[3] (WCAP-16143-P[4], "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,"
- ASME Code Case N-588[5] (where applicable), and
- Methodology of the 1996 version of ASME Section XI, Appendix G[6].
- The P-T Curves will be developed WITHOUT margins or instrumentation errors.

Circumferential Flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and ASME Code Case N-640 (K_{Ic} methodology) will be used to develop the P-T Curves for the limiting circumferential weld material. Axial Flaw methodology, 1996 Appendix G to ASME Section XI and ASME Code Case N-640 (K_{Ic} methodology) will be used to develop P-T Curves for the limiting forging/base metal material. In addition, the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (WCAP-16143-P) will be employed. The final P-T Curves will be developed by generating a P-T Curve based on the most limiting circumferential weld ART value P-T Curve and/or the P-T Curve based on the most limiting axial flaw material ART value. Hence the final P-T Curves will be a composite curve based of the limiting data from the two curves (ie. Circ flaw or axial flaw case).

The purpose of this report is to present the calculations and the development of the Braidwood Unit 1 heatup and cooldown curves for 16, 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of Exelon, the surveillance weld data from the Braidwood Unit 1 and Braidwood Unit 2 surveillance programs has been integrated. The Braidwood Unit 1 surveillance weld metal was fabricated from the same weld wire heat as the Braidwood Unit 2 surveillance weld metal (Heat No. 442011). This weld metal is in the upper to lower shell gird weld seam WF-562 of both Units. Per WCAP-15366[7], all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640 of Appendix G of the ASME Code, Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T-RT_{NDT})]} \qquad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steels.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Code Case N-640 of Appendix G of the ASME Code as follows:

$$C * K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress
K_{It} = stress intensity factor caused by the thermal gradients
K_{Ic} = function of temperature relative to the RT_{NDT} of the material
C = 2.0 for Level A and Level B service limits
C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the K_I corresponding to membrane tension for the postulated defect is:

$$K_{Im} = M_m * (pR_i \div t) \qquad (3)$$

Where M_m for an inside surface is given by:

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

$M_m = 0.926 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and

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

Similarly, M_m for an outside surface flaw is given by:

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

$M_m = 0.893 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and

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

Where:

Ri = vessel inner radius,
t = vessel wall thickness, and
p = internal pressure,

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where M_b is two-thirds of M_m

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \quad (4)$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \quad (5)$$

where:

HU = the heatup rate in °F/hr.

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

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{6}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{7}$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{8}$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3 through 8 were added to the OPERLIM computer program, which is the Westinghouse computer program used to generate pressure-temperature limit curves. No other changes were made to the OPERLIM computer program with regard to the pressure-temperature curve calculation methodology. Hence, the pressure-temperature curve methodology described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) remains valid for the generation of the pressure-temperature curves documented in this report with the exceptions described above.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of a postulated flaw at the ¼T and ¾T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the ¼T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{IC} at the ¼T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IC} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the ¼T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a ¼T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IC} for the ¼T crack during heatup is lower than the K_{IC} for the ¼T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IC} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the ¼T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a ¼T flaw located at the ¼T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Code Case N-588 for Circumferential Weld Flaws:

In 1997, ASME Section XI, Appendix G was revised to add methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The earlier ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{Im} = M_m \times (pR_i / t)$$

where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Similarly, M_m for an outside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G contains the 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 (3106 psig), which is 621 psig for the Braidwood Unit 1 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, 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.

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 Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1".

The discussion given in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," concluded that the integrity of the closure head/vessel flange region is not a concern for Byron or Braidwood, therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta\, RT_{NDT} + Margin \qquad (9)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta\, RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (10)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (11)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively)[10] is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 10 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections and the results are presented in Section 6 of WCAP-15316[11]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 4-2 through 4-4, herein, contains the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluence values used to calculate the ART values for all beltline materials in the Braidwood Unit 1 reactor vessel for 16, 22 and 32 EFPY. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-5.

Ratio Procedure and Temperature Adjustment:

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12th & 13th of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT, measured}$ to T_{plant} is made as follows:
>
> $$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT, measured} + 1.0*(T_{capsule} - T_{plant}) \quad (12)$$

The irradiation temperatures of the Braidwood Unit 1 & 2 reactor vessels are as follows:

TABLE 4-1
Irradiation Temperatures of the Braidwood Unit 1 & 2 Reactor Vessels

Fuel Cycle	Braidwood 1	Capsule	Braidwood 2	Capsule
1	557°F	U	557°F	U
2	551°F	--	551°F	--
3	551°F	--	551°F	--
4	551°F	X	551°F	X
5	551-554°F	--	551°F *	--
6	554°F	--	551°F *	--
7	554°F *	W	550°F **	W
Average Temp.	**553°F**		**552°F**	

* These values are from ComEd and are documented in BRW-DIT-97-321[15].
** This value is from ComEd and is documented in BRW-DIT-2000-0010[16].

Capsules U and X were exposed to the same average operating temperature of 553°F. Capsule W was exposed to the same average operating temperature of 553°F for 5 of 7 cycles. For cycles 6 and 7, the Braidwood Unit 1 capsule W saw only a 1°F increase in average operating temperature. Thus, the average inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest. Hence, no temperature adjustments are required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-6 through 4-9. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, all available surveillance data for Braidwood Unit 1 is credible. This assessment also calculated the vessel weld (including temperature and chemistry adjustment) chemistry factor using Braidwood Unit 1 stand alone data and it was determined to be 33.5°F. Position 2.1 chemistry factors are calculated in Table 4-15.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Braidwood Unit 1, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows: $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$.

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Braidwood Unit 1, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I.

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 10.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values at 16 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E > 1.0 MeV)	1/4T (n/cm^2, E > 1.0 MeV)	3/4T (n/cm^2, E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	1.03×10^{19}	6.19×10^{18}	2.23×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	1.03×10^{19}	6.19×10^{18}	2.23×10^{18}
Nozzle Shell Forging 5P-7016	3.07×10^{18}	1.84×10^{18}	6.65×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.00×10^{19}	6.00×10^{18}	2.17×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	3.07×10^{18}	1.84×10^{18}	6.65×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-3

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E > 1.0 MeV)	1/4T (n/cm^2, E > 1.0 MeV)	3/4T (n/cm^2, E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Nozzle Shell Forging 5P-7016	4.20×10^{18}	2.52×10^{18}	9.09×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.37×10^{19}	8.23×10^{18}	2.97×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	4.20×10^{18}	2.52×10^{18}	9.09×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-4

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E > 1.0 MeV)	1/4T (n/cm^2, E > 1.0 MeV)	3/4T (n/cm^2, E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	2.05×10^{19}	1.23×10^{19}	4.44×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	2.05×10^{19}	1.23×10^{19}	4.44×10^{18}
Nozzle Shell Forging 5P-7016	6.08×10^{18}	3.65×10^{18}	1.32×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.99×10^{19}	1.19×10^{19}	4.31×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	6.08×10^{18}	3.65×10^{18}	1.32×10^{18}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-5
Calculated Integrated Neutron Exposure of the Braidwood Unit 1
Surveillance Capsules Tested to Date

Capsule	Fluence
U	3.87×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.24×10^{19} n/cm^2, (E > 1.0 meV)
W	2.09×10^{19} n/cm^2, (E > 1.0 MeV)

Contained in Table 4-6 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[12], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-6
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[a]
Intermediate Shell Forging [49D867/49C813]-1-1 (Tangential Orientation)	U	5.78°F
	X	38.23°F
	W	24.14°F
Intermediate Shell Forging [49D867/49C813]-1-1 (Axial Orientation)	U	0.0°F[b]
	X	28.75°F
	W	37.11°F
Surveillance Program Weld Metal (Heat # 442011)	U	17.06°F
	X	30.15°F
	W	49.68°F
Heat Affected Zone	U	56.31°F
	X	92.8°F
	W	84.86°F

Notes:

(a) From capsule W analysis results Calculated using the measured Charpy data plotted using CVGRAPH, Verision 4.1[11].

(b) Actual value for ΔRT_{NDT} is -16.07. This physically should not occur, therefore for conservatism (i.e. higher Chemistry Factor) a value of zero will be reported.

Table 4-7 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 1 base materials in the beltline region. Table 4-8 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 1 surveillance weld material, while Table 4-9 presents the overall best estimate average for that heat of weld. Table 4-10 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-10 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-12. Table 4-11 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-12.

TABLE 4-7
Calculation of the Best Estimate Cu and Ni Weight Percent for the
Braidwood Unit 1 Forging Materials

	Intermediate Shell Forging [49D383/49C344]-1-1 MK 24-2		**Lower Shell Forging [49D867/49C813]-1-1 MK 24-3**	
Reference	**Cu %**	**Ni %**	**Cu %**	**Ni %**
13	0.05	0.73	---	---
11	---	---	0.03	0.73
11 (Charpy EL-6)	---	---	0.052	0.746
11 (Charpy ET-57)	---	---	0.046	0.736
Charpy ET-31[a]	---	---	0.045	0.687
Charpy EL-38[a]	---	---	0.049	0.725
Charpy ET-40[a]	---	---	0.047	0.721
Charpy EL-42[a]	---	---	0.053	0.809
Best Estimate Average[b]	**0.05**	**0.73**	**0.05**	**0.74**

Notes:
(a) Charpy Specimen From Capsule W of Braidwood Unit 1 (Ref 11).
(b) The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-8
Calculation of the Average Cu and Ni Weight Percent for the Braidwood Unit 1 Surveillance Weld Material Only (Heat # 442011)

Reference	Weight % Copper	Weight % Nickel
WCAP-14824, Rev.2[(a)]	0.032	0.671
Charpy EW-32[(b)]	0.032	0.674
Charpy EW-33[(b)]	0.032	0.660
Charpy EW-34[(b)]	0.030	0.661
Charpy EW-42[(b)]	0.033	0.690
Surveillance Weld Average	**0.03**	**0.67**

Notes:

(a) This is the average of 29 data points. Therefore, to average with the new Charpy data, multiply the average copper and nickel values from Ref. 17 by 29, add four new points and divide by 33.
(b) Charpy specimen from Capsule W of Braidwood Unit 1 (Ref. 11).

TABLE 4-9

Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Units 1 & 2 Weld Material (Using Braidwood 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.028	0.63
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.03	0.65
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.04	0.67
Braidwood Unit 1 Surv. Data Ave.[a]	Table 4-8	0.032	0.671
Braidwood Unit 2 Surv. Data Ave.[b]	WCAP-15369	0.034	0.710
BEST ESTIMATE AVERAGE[c]		**0.03**	**0.67**

Notes:

(a) The weld material in the Braidwood Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld (Weld seam WF-562, Wire Heat No. 442011, Flux Type Linde 80, Flux Lot No. 8061). The weld wire is type Linde MnMoNi (Low Cu-P).

(b) The Braidwood Unit 2 surveillance weld is representative of that used in the Braidwood Unit 1 reactor vessel core region girth seam (WF-562) heat number 442011, with a Linde 80 type flux, lot number 8061 (i.e. The Braidwood Unit 2 surveillance weld was fabricated with Linde 80 type flux, Lot number 0344).

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-10
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}[(a)]
Closure Head Flange 2030-V-1	0.11	0.67	-20
Vessel Flange 122N357VA1	---	0.77	-10
Nozzle Shell Forging 5P7016	0.04	0.73	10
Intermediate Shell Forging [49D383/49C344]-1-1 [Table 4-7]	0.05	0.73	-30
Lower Shell Forging [49D867/49C813]-1-1 [Table 4-7]	0.05	0.74	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011) [Table 4-9]	0.03	0.67	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.04	0.46	-25
Surveillance Weld Braidwood Unit 1 (Heat 442011) [Table 4-9]	0.03	0.67	---
Surveillance Weld Braidwood Unit 2 (Heat 442011) [Table 4-9]	0.03	0.71	---

Notes:

(a) The Initial RT_{NDT} values for the forgings and welds are based on measured data.

TABLE 4-11
Calculation of Chemistry Factors for Braidwood Unit 1 using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c] (°F)	$FF*\Delta RT_{NDT}$ (°F)	FF^2
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.387	0.737	5.78	4.26	0.543
	X	1.24	1.060	38.23	40.52	1.124
	W	2.09	1.201	24.14	28.99	1.442
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.387	0.737	0.0[e]	0.0	0.543
	X	1.24	1.060	28.75	30.48	1.124
	W	2.09	1.201	37.11	44.57	1.442
				SUM:	148.82	6.218
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (148.82°F) \div (6.218) = 23.9°F$					
Braidwood Unit 1 Surveillance Weld Metal	U	0.387	0.737	17.06[d]	12.57	0.543
	X	1.24	1.060	30.15[d]	31.96	1.124
	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2 Surveillance Weld Metal	U	0.40	0.746	0.0[d,f]	0.0	0.557
	X	1.23	1.058	26.3[d]	27.83	1.119
	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19°F) \div (6.273) = 25.7°F$					

Notes:
(a) f = Calculated fluence from the Braidwood Unit 1 capsule W dosimetry analysis results[11] and the Calulated fluence from the Braidwood Unit 2 capsule W analysis[14], (x 10^{19} n/cm^2, E > 1.0 MeV).
(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.
(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 1 taken from Ref. 11 and for Braidwood Unit 2 taken from Ref. 14.
(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted. The chemistry factors of the surveillance welds and the vessel weld is 41°F, hence, the ratio factor is 1.0. In addition, the average inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest. Hence, no temperature adjustments are required.
(e) Actual value of ΔRT_{NDT} is -16.07. This physically should not occur, therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.
(f) Measured value is –0.58. 0 is assumed for conservatism.

Contained in Table 4-12 is a summary of the calculated chemistry factors of the Braidwood Unit 1 beltline materials based on the Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1 methodology.

TABLE 4-12
Summary of the Braidwood Unit 1 Beltline Material Chemistry Factor Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell Forging [49D383/49C344]-1-1	31.0°F	---
Lower Shell Forging [49D867/49C813]-1-1	31.0°F	23.9°F
Nozzle Shell Forging 5P7016	26.0°F	---
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	54.0°F	---
Braidwood Unit 1 Surveillance Program Weld Metal	41.0°F	---
Braidwood Unit 1 & 2 Surveillance Program Weld Metal	41.0°F	---

Contained in Tables 4-13 through 4-15 is a summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Braidwood Unit 1 reactor vessel beltline materials for 16, 22 and 32 EFPY.

TABLE 4-13
Calculation of the ¼T and ¾T Fluence Factor Values used for the Generation of the 16 EPFY Heatup/Cooldown Curves

Material	¼T F (n/cm^2, E > 1.0 MeV)	¼T FF	¾T f (n/cm^2, E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	6.19×10^{18}	0.866	2.23×10^{18}	0.596
Lower Shell Forging [49D687/49C813]-1-1	6.19×10^{18}	0.866	2.23×10^{18}	0.596
Nozzle Shell Forging 5P-7016	1.84×10^{18}	0.550	6.65×10^{17}	0.340
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	6.00×10^{18}	0.857	2.17×10^{18}	0.589
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	1.84×10^{18}	0.550	6.65×10^{17}	0.340

TABLE 4-14
Calculation of the ¼T and ¾T Fluence Factor Values
used for the Generation of the 22 EFPY Heatup/Cooldown Curves

Material	¼T F (n/cm^2, E > 1.0 MeV)	¼T FF	¾T f (n/cm^2, E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Lower Shell Forging [49D687/49C813]-1-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Nozzle Shell Forging 5P-7016	2.52×10^{18}	0.626	9.09×10^{17}	0.398
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	8.23×10^{18}	0.945	2.97×10^{18}	0.668
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	2.52×10^{18}	0.626	9.09×10^{17}	0.398

TABLE 4-15
Calculation of the ¼T and ¾T Fluence Factor Values
used for the Generation of the 32 EFPY Heatup/Cooldown Curves

Material	¼T F (n/cm^2, E > 1.0 MeV)	¼T FF	¾T f (n/cm^2, E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	1.23×10^{19}	1.058	4.44×10^{18}	0.774
Lower Shell Forging [49D687/49C813]-1-1	1.23×10^{19}	1.058	4.44×10^{18}	0.774
Nozzle Shell Forging 5P-7016	3.65×10^{18}	0.722	1.32×10^{18}	0.475
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.19×10^{19}	1.049	4.31×10^{18}	0.766
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	3.65×10^{18}	0.722	1.32×10^{18}	0.475

Contained in Tables 4-16 and 4-21 are the calculations of the ART values used for the generation of the 16, 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-16
Calculation of the ART Values for the ¼T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 16[b] EFPY (x 10^{19})	¼-t f (x 10^{19})	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.03	0.619	0.866	-30	26.8	0	13.4	26.8	24
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.03	0.619	0.866	-20	26.8	0	13.4	26.8	34
Lower Shell Forging → using S/C Data				23.9	1.03	0.619	0.866	-20	20.7	0	10.4	20.7	21
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.00	0.600	0.857	40	35.1	0	17.0	34.0	109
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.00	0.600	0.857	40	22.0	0	11.0	22.0	84
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.307	0.184	0.550	10	14.3	0	7.2	14.3	39
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.307	0.184	0.550	-25	29.7	0	14.9	29.7	34

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible.

TABLE 4-17

Calculation of the ART Values for the ¾T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 16[b] EFPY ($x 10^{19}$)	3/4-t f ($x 10^{19}$)	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.03	0.223	0.596	-30	18.5	0	9.3	18.5	7
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.03	0.223	0.596	-20	18.5	0	9.3	18.5	17
Lower Shell Forging → using S/C Data				23.9	1.03	0.223	0.596	-20	14.2	0	7.1	14.2	8
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.00	0.217	0.589	40	24.1	0	12.1	24.1	88
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.00	0.217	0.589	40	15.1	0	7.6	15.1	70
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.307	0.067	0.340	10	8.8	0	4.4	8.8	28
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.307	0.067	0.340	-25	18.4	0	9.2	18.4	12

NOTES:

(a) Initial RT_{NDT} values are measured values.

(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(d) ΔRT_{NDT} = CF * FF

(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).

(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible.

TABLE 4-18

Calculation of ART Values for the ¼T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 22[b] EFPY ($x 10^{19}$)	¼-t f ($x 10^{19}$)	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.41	0.847	0.953	-30	29.5	0	14.8	29.5	29
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.41	0.847	0.953	-20	29.5	0	14.8	29.5	39
Lower Shell Forging → using S/C Data				23.9	1.41	0.847	0.953	-20	22.8	0	11.4	22.8	26
Inter. To Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.37	0.823	0.945	40	38.7	0	19.4	38.7	117
Inter. To Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.37	0.823	0.945	40	24.3	0	12.2	24.3	89
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.420	0.252	0.626	10	16.3	0	8.2	16.3	43
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.420	0.252	0.626	-25	33.8	0	16.9	33.8	43

NOTES:

(a) Initial RT_{NDT} values are measured values.

(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(d) ΔRT_{NDT} = CF * FF

(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).

(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-19
Calculation of ART Values for the ¾T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	F @ 22[b] EFPY ($x\ 10^{19}$)	3/4-t f ($x\ 10^{19}$)	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.41	0.305	0.675	-30	20.9	0	10.5	20.9	12
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.41	0.305	0.675	-20	20.9	0	10.5	20.9	22
Lower Shell Forging → using S/C Data				23.9	1.41	0.305	0.675	-20	16.1	0	8.1	16.1	12
Inter. To Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.37	0.297	0.668	40	27.4	0	13.7	27.4	95
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.37	0.297	0.668	40	17.2	0	8.6	17.2	74
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.420	0.091	0.398	10	10.3	0	5.2	10.3	31
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.420	0.091	0.398	-25	21.5	0	10.8	21.5	18

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-20
Calculation of ART Values for the ¼T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 32[b] EFPY ($x 10^{19}$)	¼-t f ($x 10^{19}$)	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	2.05	1.23	1.058	-30	32.8	0	16.4	32.8	36
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	2.05	1.23	1.058	-20	32.8	0	16.4	32.8	46
Lower Shell Forging → using S/C Data				23.9	2.05	1.23	1.058	-20	25.3	0	12.7	25.3	31
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.99	1.19	1.049	40	43.0	0	21.5	43.0	126
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.99	1.19	1.049	40	27.0	0	13.5	27.0	94
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.608	0.365	0.722	10	18.8	0	9.4	18.8	48
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.608	0.365	0.722	-25	39.0	0	19.5	39.0	53

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-21
Calculation of ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 32[b] EFPY ($x 10^{19}$)	3/4-t f ($x 10^{19}$)	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	2.05	0.444	0.774	-30	24.0	0	12.0	24.0	18
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	2.05	0.444	0.774	-20	24.0	0	12.0	24.0	28
Lower Shell Forging → using S/C Data				23.9	2.05	0.444	0.774	-20	18.5	0	9.3	18.5	17
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.99	0.431	0.766	40	31.4	0	15.7	31.4	103
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.99	0.431	0.766	40	19.7	0	9.9	19.7	79
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.608	0.132	0.475	10	12.4	0	6.2	12.4	35
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.608	0.132	0.475	-25	25.7	0	12.9	25.7	26

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

The intermediate to lower shell circumferential weld and the nozzle shell forging are the limiting beltline material for all heatup and cooldown curves to be generated. The ART value associated with this material will be used in all three sets of curves. The circumferential weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw). The ART associated with the limiting axial material must also be considered to determine if this case would be more conservative or overlap the circumferential flaw curves. Contained in Table 4-22 is a summary of the limiting ARTs to be used in the generation of the Braidwood Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-22
Summary of the Limiting ART Values used in Generation of the Braidwood Unit 1 Reactor Vessel Heatup and Cooldown Curves

EFPY	¼ T Limiting ART	¾ T Limiting ART
Limiting Circumferential Material (Weld Seam WF-562)		
16	84	70
22	89	74
32	94	79
Limiting Axial Material (Nozzle Shell Forging 5P-7016)		
16	39	28
22	43	31
32	48	35

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-6 present the 16, 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology, Code Case N-640 and Code Case N-588. The heatup and cooldown curves generated for the limiting circumferential material (Weld Seam WF-562) utilizing Code Case N-588 and the 1996 Appendix G methodology are less conservative than the heatup and cooldown curves generated for the axial flaw case even though the ART values is higher. This is true throughout the entire temperature range, including the criticality curve. Hence, the heatup and cooldown curves presented in this report were generated utilizing the limiting axial material ART (Nozzle Shell Forging 5P-7016), Code Case N-640 and the 1996 ASME Appendix G methodology.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-6. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1, 5-3 and 5-5 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640 and Appendix G to Section XI of the ASME Code[6] as follows:

$$1.5K_{Im} < K_{Ic} \qquad (13)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02(T - RTNDT)]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 16 EFPY is 100°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 22 EFPY is 103°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 32 EFPY is 108°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The approximately vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-6 define all of the above limits for ensuring prevention of nonductile failure for the Braidwood Unit 1 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-6 are presented in Tables 5-1 through 5-3, respectively.

Additionally, Westinghouse Engineering has reviewed the minimum boltup temperature requirements for the Braidwood Unit 1 reactor pressure vessel. According to Paragraph G-2222 of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, the reactor vessel may be bolted up and pressurized to 20 percent of the initial hydrostatic test pressure at the initial RT_{NDT} of the material stressed by the boltup. Therefore, since the most limiting initial RT_{NDT} value is -10°F (vessel flange), the reactor vessel can be bolted up at this temperature.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 16 EFPY: ¼T, 39°F
¾T, 28°F

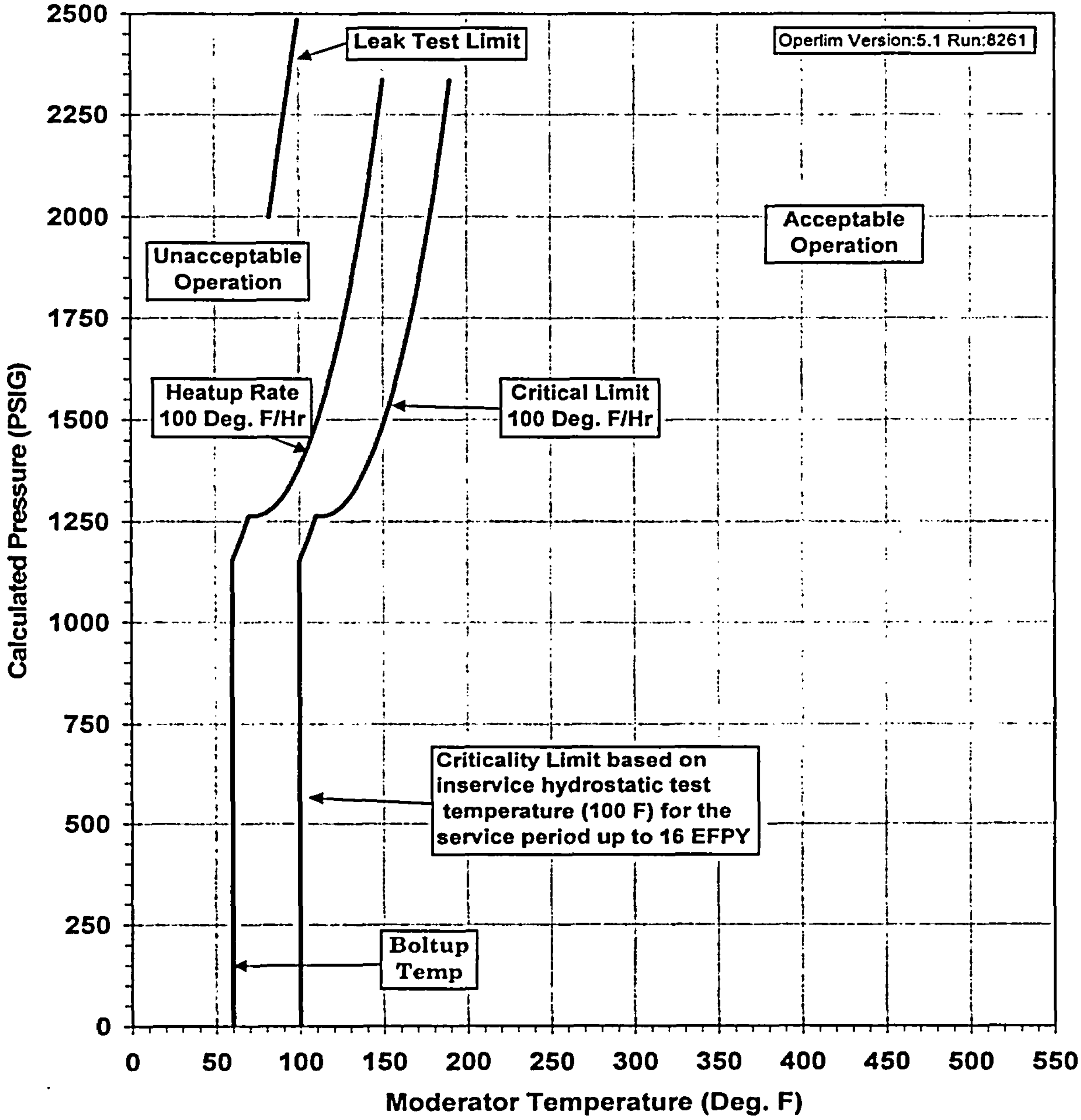

FIGURE 5-1 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 16 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 16 EFPY: ¼T, 39°F
¾T, 28°F

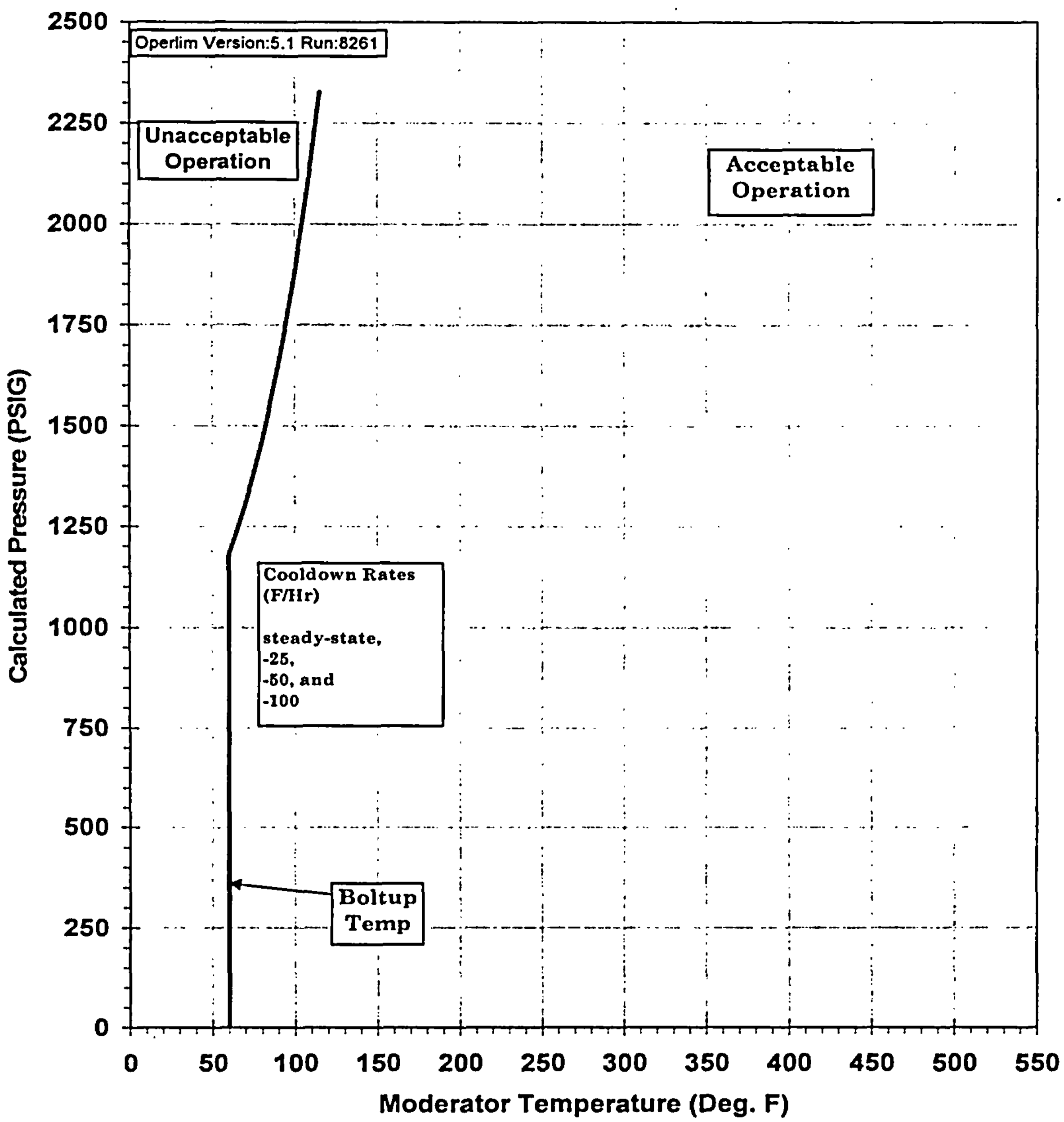

FIGURE 5-2 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 16 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

TABLE 5-1
Braidwood Unit 1 Heatup and Cooldown Data Points for the 16 EFPY Curves
Using 1996 Appendix G and Code Case N-640 Methodology
(Without Margins for Instrumentation Errors)

Run = 8261		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1177	60	1177	60	1177	60	1177
65	1237	65	1237	65	1237	65	1237
70	1304	70	1304	70	1304	70	1304
75	1377	75	1377	75	1377	75	1377
80	1459	80	1459	80	1459	80	1459
85	1549	85	1549	85	1549	85	1549
90	1648	90	1648	90	1648	90	1648
95	1758	95	1758	95	1758	95	1758
100	1880	100	1880	100	1880	100	1880
105	2014	105	2014	105	2014	105	2014
110	2162	110	2162	110	2162	110	2162
115	2326	115	2326	115	2326	115	2326

Run = 8261	Heatup Curves					
	100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
	60	0	100	0	82	2000
	60	1151	100	1151	99	2485
	65	1205	105	1205		
	70	1263	110	1263		
	75	1263	115	1263		
	80	1271	120	1271		
	85	1286	125	1286		
	90	1310	130	1310		
	95	1342	135	1342		
	100	1382	140	1382		
	105	1431	145	1431		
	110	1488	150	1488		
	115	1555	155	1555		
	120	1631	160	1631		
	125	1717	165	1717		
	130	1814	170	1814		
	135	1924	175	1924		
	140	2046	180	2046		
	145	2183	185	2183		
	150	2335	190	2335		

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 22 EFPY: ¼T, 43°F
¾T, 31°F

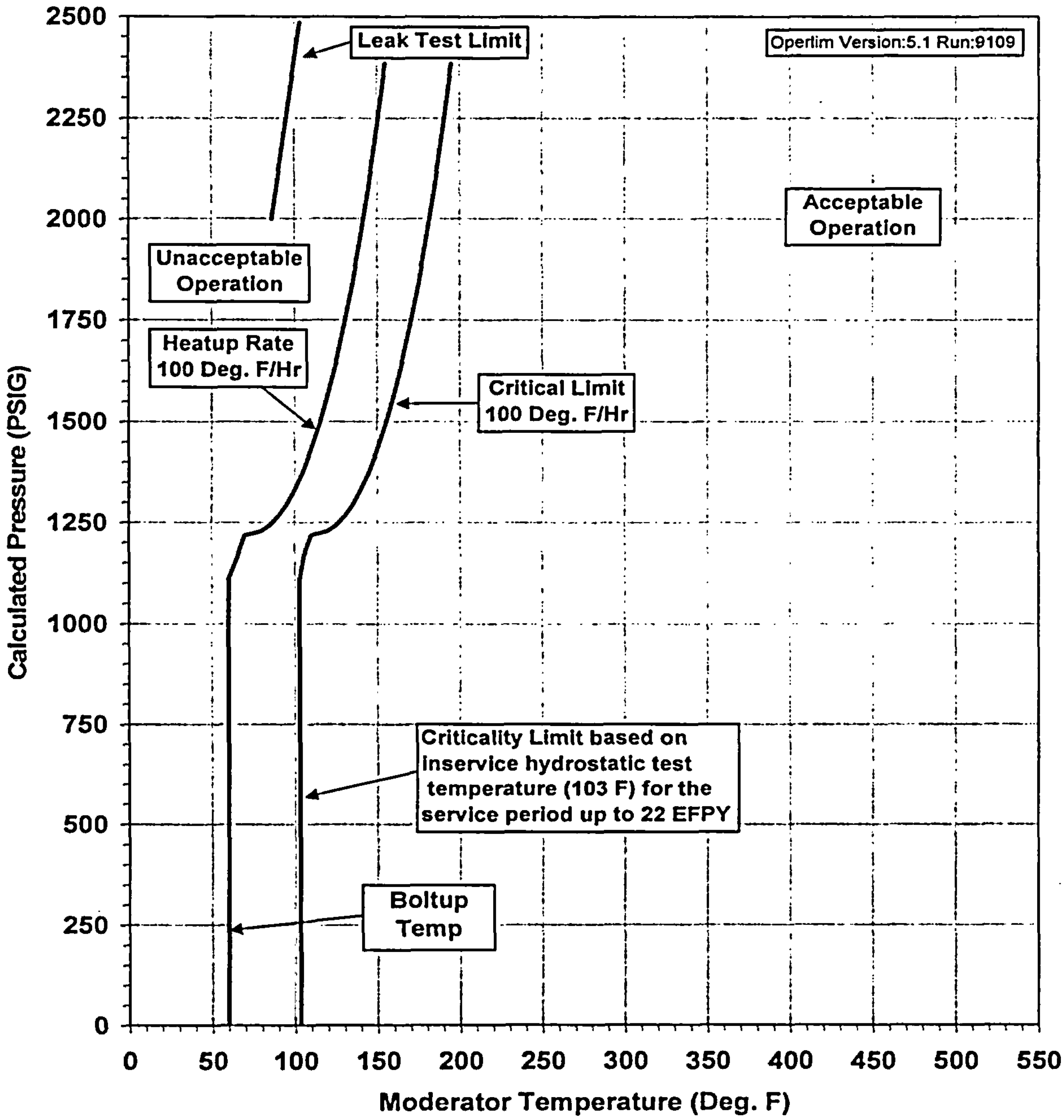

FIGURE 5-3 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 22 EFPY: ¼T, 43°F
¾T, 31°F

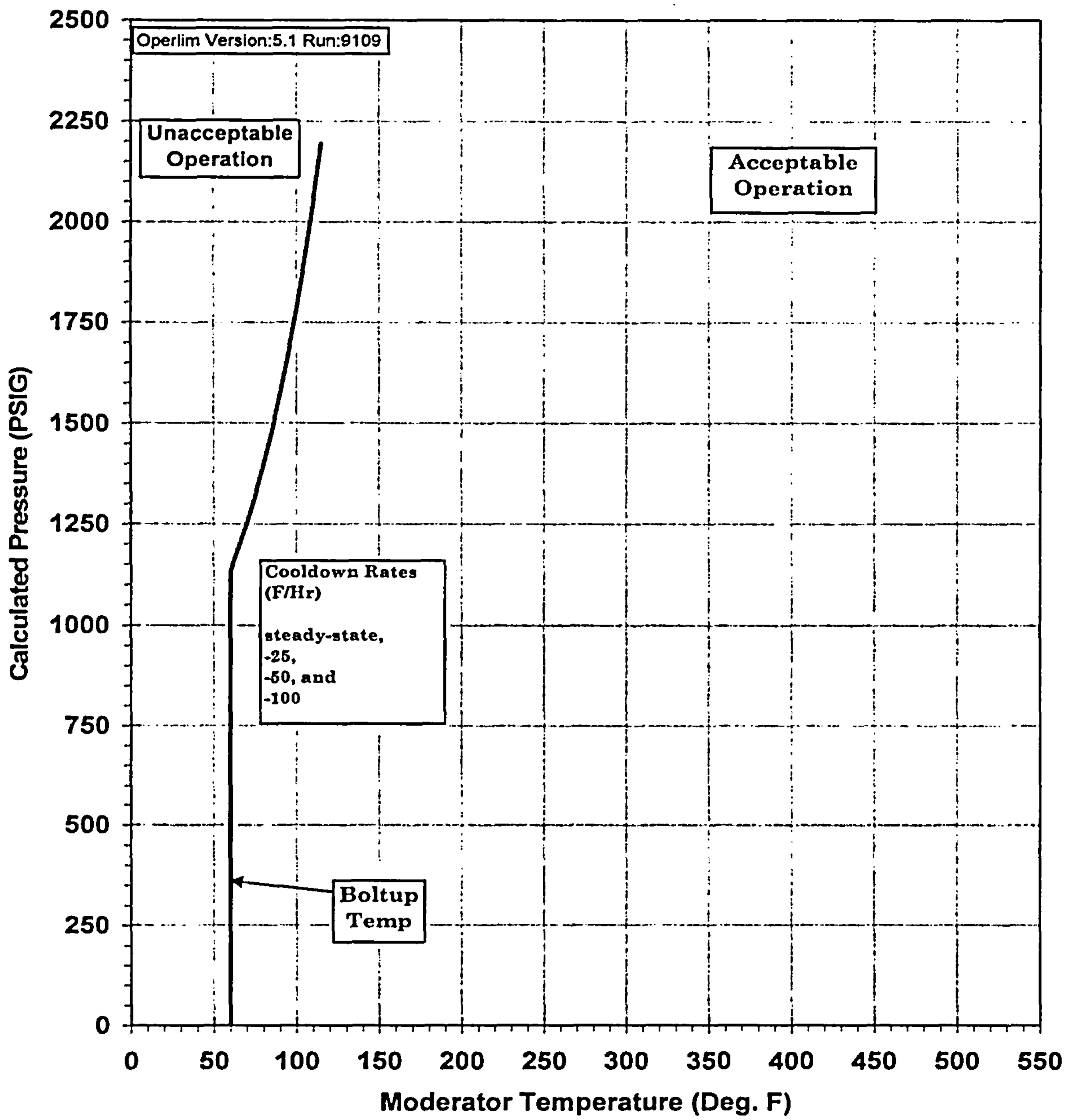

FIGURE 5-4 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

TABLE 5-2
Braidwood Unit 1 Heatup and Cooldown Curve Data Points for the 22 EFPY Curves
Using 1996 Appendix G and Code Case N-640 Methodology
(Without Margins for Instrumentation Errors)

Run = 9109		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1133	60	1133	60	1133	60	1133
65	1188	65	1188	65	1188	65	1188
70	1250	70	1250	70	1250	70	1250
75	1318	75	1318	75	1318	75	1318
80	1393	80	1393	80	1393	80	1393
85	1476	85	1476	85	1476	85	1476
90	1568	90	1568	90	1568	90	1568
95	1669	95	1669	95	1669	95	1669
100	1781	100	1781	100	1781	100	1781
105	1905	105	1905	105	1905	105	1905
110	2042	110	2042	110	2042	110	2042
115	2194	115	2194	115	2194	115	2194
120	2361	120	2361	120	2361	120	2361

Run = 9109		Heatup Curves					
		100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
		Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
		60	0	103	0	86	2000
		60	1110	103	1110	103	2485
		65	1163	105	1163		
		70	1218	110	1218		
		75	1222	115	1222		
		80	1229	120	1229		
		85	1243	125	1243		
		90	1264	130	1264		
		95	1294	135	1294		
		100	1331	140	1331		
		105	1376	145	1376		
		110	1430	150	1430		
		115	1492	155	1492		
		120	1563	160	1563		
		125	1644	165	1644		
		130	1736	170	1736		
		135	1838	175	1838		
		140	1954	180	1954		
		145	2082	185	2082		
		150	2225	190	2225		
		155	2384	195	2384		

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: ¼T, 48°F
¾T, 35°F

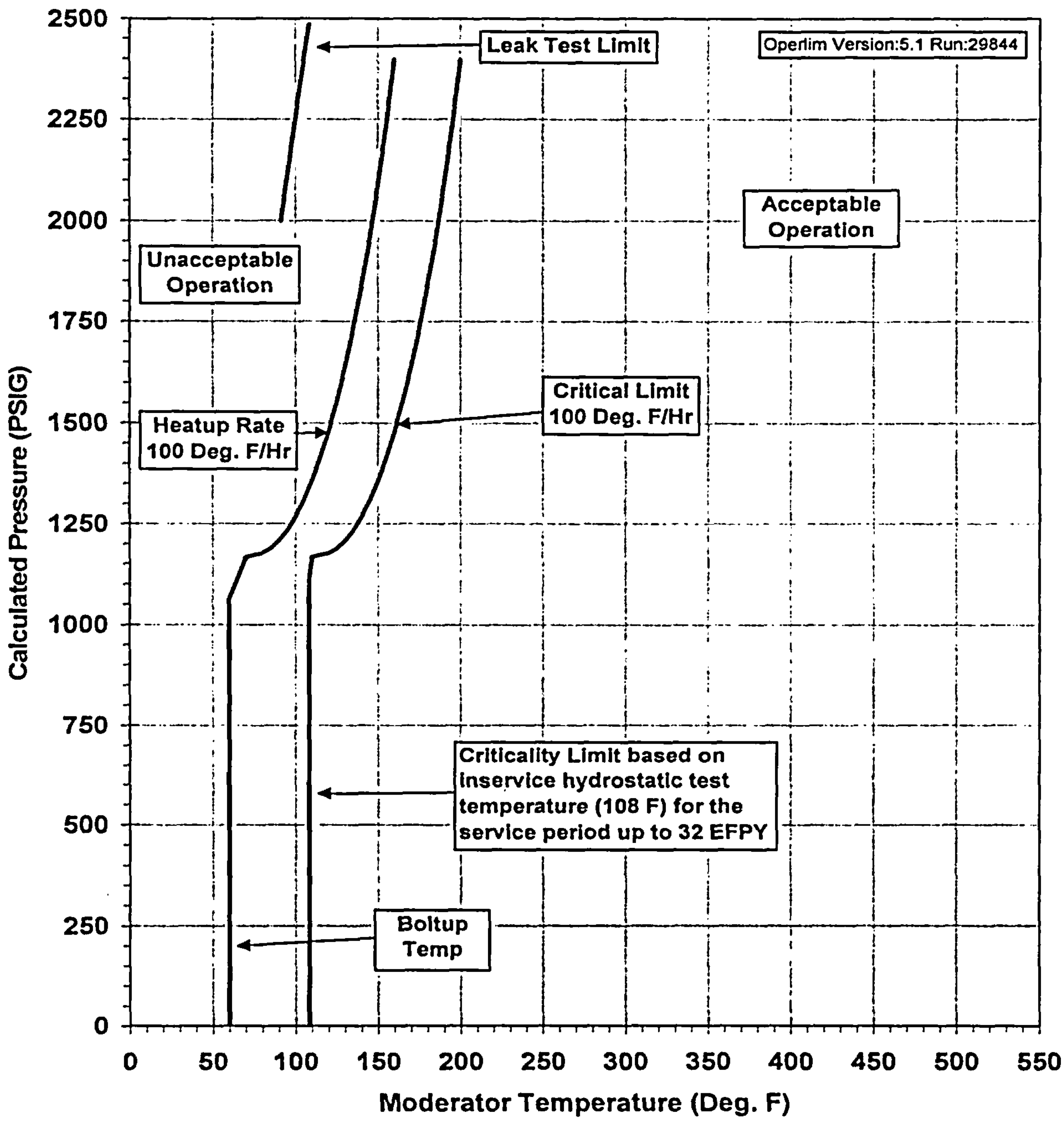

FIGURE 5-5 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: 1/4T, 48°F
3/4T, 35°F

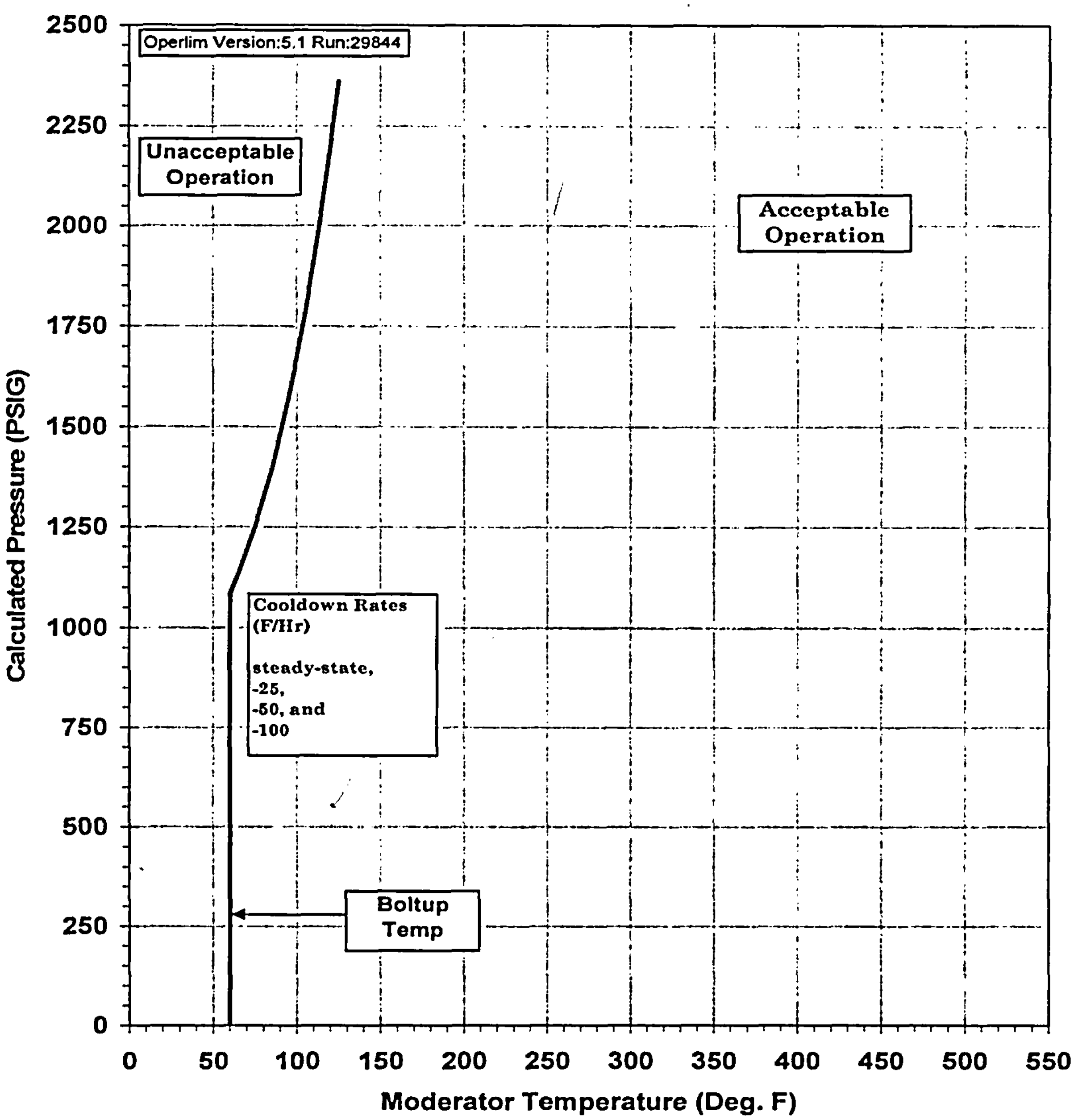

FIGURE 5-6 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

TABLE 5-3
Braidwood Unit 1 Heatup and Cooldown Curve Data Points for the 32 EFPY Curves
Using 1996 Appendix G and Code Case N-640 Methodology
(Without Margins for Instrumentation Errors)

Run = 29844		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1082	60	1078	60	1079	60	1079
65	1133	65	1133	65	1133	65	1133
70	1188	70	1188	70	1188	70	1188
75	1250	75	1250	75	1250	75	1250
80	1318	80	1318	80	1318	80	1318
85	1393	85	1393	85	1393	85	1393
90	1476	90	1476	90	1476	90	1476
95	1568	95	1568	95	1568	95	1568
100	1669	100	1669	100	1669	100	1669
105	1781	105	1781	105	1781	105	1781
110	1905	110	1905	110	1905	110	1905
115	2042	115	2042	115	2042	115	2042
120	2194	120	2194	120	2194	120	2194
125	2361	125	2361	125	2361	125	2361

Run =29844	Heatup Curves					
	100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
	60	0	108	0	91	2000
	60	1064	108	1114	108	2485
	65	1114	110	1166		
	70	1166	115	1172		
	75	1172	120	1176		
	80	1176	125	1188		
	85	1188	130	1207		
	90	1207	135	1234		
	95	1234	140	1267		
	100	1267	145	1308		
	105	1308	150	1357		
	110	1357	155	1414		
	115	1414	160	1479		
	120	1479	165	1554		
	125	1554	170	1638		
	130	1638	175	1732		
	135	1732	180	1838		
	140	1838	185	1956		
	145	1956	190	2088		
	150	2088	195	2235		
	155	2235	200	2397		
	160	2397				

6 REFERENCES

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2. Cases of ASME Boiler and Pressure Vessel Code, Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", Approved March 1999.

3. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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6. ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1995.

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15 Nuclear Design Information Transmittal NDIT No. BRW-DIT-97-321, Rev. 0.

16 Nuclear Design Information Transmittal NDIT No. BRW-DIT-2000-0010.

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APPENDIX A

Thermal Stress Intensity Factors (K_{It})

TABLE A1

K_{It} Values for 100°F/hr Heatup Curve (16 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
PT Curves are Limited by the ¼ T Location up to 65°F and ¾ T Limited for the Remainder of the Curve				
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183

Vessel Radius to the ¼T and ¾T Locations are as follows:

- 1/4T Radius = 88.750"
- 3/4T Radius = 93.000"

TABLE A2
K_{It} Values for 100°F/hr Cooldown Curve (16 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: All Cooldown Rates are limited by the Steady State Condition.

TABLE A3
K_{It} Values for 100°F/hr Heatup Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
PT Curves are Limited by the ¼ T Location up to 70°F and ¾ T Limited for the Remainder of the Curve				
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799

TABLE A4

K_{It} Values for 100°F/hr Cooldown Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: All Cooldown Rates are limited by the Steady State Condition.

TABLE A5
K_{It} Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
PT Curves are Limited by the ¼ T Location up to 70°F and ¾ T Limited for the Remainder of the Curve				
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277

TABLE A6
K_{It} Values for 100°F/hr Cooldown Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: All Cooldown Rates are limited by the Steady State Condition except for 60°F, where the 25°F/hr rate controls the higher rates.

ATTACHMENT 4-B

WCAP-15373, Revision 2

Braidwood Unit 2
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Non-Proprietary Class 3

WCAP-15373
Revision 2

November 2003

Braidwood Unit 2
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15373, Revision 2

Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

November 2003

Approved:
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PREFACE

This report has been technically reviewed and verified by:

Reviewer: J.H. Ledger

Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15364 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15364 Rev. 0.

Note that only the heatup curves and associated data point tables in section 5 have changed. The cooldown curves and data points remain valid and were not changed.

Revision 2:

This revision was completed to incorporate an updated reference for the flange elimination from the PT limit curves. In the previous revision, WCAP-15315 was provided as justification for the flange notch elimination and it was changed to WCAP-16143-P in this revision. In addition to this change, the thermal stress intensity factors for the highest heatup and cooldown rates were added to this report in Appendix A.

EXECUTIVE SUMMARY

The purpose of this report is to generate pressure-temperature limit curves for Braidwood Unit 2 for normal operation at 12, 16, 22 and 32 EFPY using the Following methodologies: WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-16143-P. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART). The 1/4T and 3/4T values are summarized in Tables 4-18 through 4-25 and were calculated using the circumferential weld WF-562, Heat 442011 (The limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 5P-7056 (The limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-8.

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1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Braidwood Station Unit No. 2 reactor pressure vessel (Specifically from lower shell forging 50D102-1/50C97-1). Also supplied was a weldment made with weld wire heat # 442011 Linde 80 flux, lot number 0344, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company (now known as Exelon Nuclear) contracted Westinghouse to analyze surveillance capsule W from the Braidwood Unit 2 reactor vessel and perform an evaluation for a 5% Uprating. As a part of these analysis Westinghouse generated new heatup and cooldown curves for 12, 16, 22 and 32 EFPY. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10 CFR Part 50[3] per WCAP-16143-P, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"[4],
- ASME Code Case N-588[5] (where applicable),
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G,[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, two sets of PT Curves will be generated. Set one will consist of the circumferential flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and the K_{Ic} methodology (ASME Code Case N-640) for the limiting circumferential weld material. Set two will consist of the 1996 Appendix G to ASME Section XI and the K_{Ic} methodology (ASME Code Case N-640) for the limiting forging/base metal material. Both sets of curves will used the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-16143-P). The final PT curves to be presented herein will be the most limiting set of curves. If the situation arises where portions of each set of curves are limiting, then composite curves will be generated that are based on the most limiting data (i.e. Circ. Flaw or Axial Flaw Case).

The purpose of this report is to present the calculations and the development of the Commonwealth Edison Company Braidwood Unit 2 heatup and cooldown curves for 12, 16, 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Commonwealth Edison Company, the surveillance weld data from the Braidwood Unit 1 and Unit 2 surveillance programs has been integrated. Note that Braidwood Unit 1 surveillance weld is identical to the surveillance weld (Heat No. 442011) at Braidwood Unit 2. Per WCAP-15368[7], all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \tag{1}$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2 and SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \tag{2}$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ic}	=	function of temperature relative to the RT_{NDT} of the material
C	=	2.0 for Level A and Level B service limits
C	=	1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \qquad (3)$$

where, M_m for an inside surface flaw is given by:

$M_m = 1.85$ for $\sqrt{t} < 2$,
$M_m = 0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m = 3.21$ for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

$M_m = 1.77$ for $\sqrt{t} < 2$,
$M_m = 0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m = 3.09$ for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

$K_{Ib} = M_b$ * Maximum Stress, where M_b is two-thirds of M_m

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

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

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

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (4)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Code Case N-588; Circumferential Welds:

In 1997, ASME Section XI, Appendix G was revised to add methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The earlier ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds.

In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of -2120 is

$$K_{Im} = M_m \times (pR_i / t)$$

where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Similarly, M_m for an outside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note again, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] addresses 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 preservice hydrostatic test pressure (3106 psi), which is 621 psig for Braidwood Unit 2 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, 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 pressure limitation of 20 percent of the hydrotest pressure were developed using the K_{IA} fracture toughness from the mid 1970's.

Improved knowledge of the 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 development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," concluded that the integrity of the closure head/vessel flange region is not a concern for Byron or Braidwood, therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta\ RT_{NDT} + Margin \tag{7}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta\ RT_{NDT} = CF * f^{(0.28-0.10\log f)} \tag{8}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \tag{9}$$

where x inches (vessel beltline thickness is 8.5 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections and the results are presented in Section 6 of WCAP-15369[10] and WCAP-15316, Rev. 1[11]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-2 through 4-5, herein, contain the calculated peak vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Braidwood Unit 2 reactor vessel. Additionally, the Braidwood Unit 2 calculated surveillance capsule fluence values are presented in Table 4-6.

<u>Ratio Procedure and Temperature Adjustment:</u>

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2, Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the measured ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted measured ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12th and 13th of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradition damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT, measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\,measured} + 1.0*(T_{capsule} - T_{plant}) \qquad (10)$$

Per Reference 12, page C-5, following are the average operating inlet temperatures of the Braidwood Unit 1 and 2 reactor vessels along with when the capsules were removed.

TABLE 4-1
Irradiation Temperature of the Braidwood Unit 1 and 2 Reactor Vessels

Fuel Cycle	Braidwood 1	Capsule	Braidwood 2	Capsule
1	557°F	U	557°F	U
2	551°F	--	551°F	--
3	551°F	--	551°F	--
4	551°F	X	551°F	X
5	551-554°F	--	551°F*	--
6	554°F	--	551°F*	--
7	554°F*	W	550°F**	W
Average Temp.	**553°F**		**552°F**	

* Information provided by ComEd in NDIT No. BRW-DIT-97-321[13].
** Information provided by ComEd in NDIT No. BRW-DIT-2000-0010[14].

Capsules U and X were exposed to the same average operating temperature of 553°F. Capsule W was exposed to the same average operating temperature of 553°F for 5 of the 7 cycles. For cycles 6 and 7, capsule W saw only a difference of 1°F in average operating temperature. Thus, the inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest and no temperature adjustments are required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-8 through 4-10. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, the surveillance weld data for Braidwood Unit 2 is credible while the surveillance forging material is non-credible. In addition, Reference 7 also shows that the Table chemistry factor is non-conservative and the surveillance chemistry factor should be used with a full margin term. This assessment also calculated the vessel weld (including temperature and chemistry adjustment) chemistry factor using Braidwood Unit 2 stand alone data and Braidwood Unit 1 and 2 combined data. The chemistry factor was determined to be 25.7°F. Position 2.1 chemistry factors are calculated in Table 4-12.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Braidwood Unit 2, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:
$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$.

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Braidwood Unit 2, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I .

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 12 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	Surface[b] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	7.43×10^{18}	4.46×10^{18}	1.61×10^{18}
Lower Shell Forging 50D102-1/50C97-1	7.43×10^{18}	4.46×10^{18}	1.61×10^{18}
Nozzle Shell Forging 5P-7056	2.16×10^{18}	1.30×10^{18}	4.68×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	7.20×10^{18}	4.32×10^{18}	1.56×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.16×10^{18}	1.30×10^{18}	4.68×10^{17}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-3
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 16 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	**Surface[b]** (n/cm^2, E>1.0 MeV)	**¼ T** (n/cm^2, E>1.0 MeV)	**¾ T** (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	9.87×10^{18}	5.93×10^{18}	2.14×10^{18}
Lower Shell Forging 50D102-1/50C97-1	9.87×10^{18}	5.93×10^{18}	2.14×10^{18}
Nozzle Shell Forging 5P-7056	2.86×10^{18}	1.72×10^{18}	6.19×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	9.54×10^{18}	5.73×10^{18}	2.07×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.86×10^{18}	1.72×10^{18}	6.19×10^{17}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-4
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	**Surface[b]** (n/cm^2, E>1.0 MeV)	**¼ T** (n/cm^2, E>1.0 MeV)	**¾ T** (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	1.35×10^{19}	8.11×10^{18}	2.92×10^{18}
Lower Shell Forging 50D102-1/50C97-1	1.35×10^{19}	8.11×10^{18}	2.92×10^{18}
Nozzle Shell Forging 5P-7056	3.91×10^{18}	2.35×10^{18}	8.47×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.31×10^{19}	7.87×10^{18}	2.84×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	3.91×10^{18}	2.35×10^{18}	8.47×10^{17}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-5

Summary of the Peak Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[(a)]	Surface[(b)] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	1.96×10^{19}	1.18×10^{19}	4.24×10^{18}
Lower Shell Forging 50D102-1/50C97-1	1.96×10^{19}	1.18×10^{19}	4.24×10^{18}
Nozzle Shell Forging 5P-7056	5.67×10^{18}	3.40×10^{18}	1.23×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.89×10^{19}	1.13×10^{19}	4.09×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	5.67×10^{18}	3.40×10^{18}	1.23×10^{18}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-6
Calculated Integrated Neutron Exposure of the Braidwood Unit 2 Surveillance Capsules Tested to Date

Capsule	Fluence
U	4.00×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.23×10^{19} n/cm^2, (E > 1.0 MeV)
W	2.25×10^{19} n/cm^2, (E > 1.0 MeV)

Notes:
(a) Documented in WCAP-15369[10].

Contained in Table 4-7 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[15], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-7
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[a]
Lower Shell Forging	U	-9.73
50D102-1/50C97-1	X	-9.42
(Tangential Orientation)	W	4.53
Lower Shell Forging	U	-0.13
50D102-1/50C97-1	X	33.94
(Axial Orientation)	W	33.2
Surveillance Program	U	-0.58
Weld Metal	X	26.3
(Heat # 442011)	W	23.9
Heat Affected Zone	U	-34.45
	X	-9.54
	W	4.03

Notes:

(a) From capsule W analysis results [10].

Table 4-8 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 2 base materials in the beltline region. Table 4-9 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 2 surveillance weld material, while Table 4-10 presents the overall best estimate average for that heat of weld. Table 4-11 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-11 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-13. Table 4-12 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-13.

TABLE 4-8

Calculation of the Best Estimate Cu and Ni Weight Percent for the Braidwood Unit 2 Forging Materials

Reference	Intermediate Shell Forging 49D963-1/49C904-1		Lower Shell Forging 50D102-1/50C97-1	
	Cu %	Ni %	Cu %	Ni %
WCAP-11188[16]	0.03	0.71	---	---
WCAP-11188[16]	---	---	0.06	0.75
WCAP-11188[16]	---	---	0.057	0.77
Charpy FL-6[17]	---	---	0.049	0.745
Ref. 18			0.056	0.804
Charpy FL-33[10]	---	---	0.065	0.680
Charpy FL-42[10]	---	---	0.065	0.690
Charpy FT-43[10]	---	---	0.060	0.795
Charpy FT-45[10]	---	---	0.066	0.840
Best Estimate Average[a]	**0.03**	**0.71**	**0.06**	**0.76**

Notes:
(a) The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-9

Calculation of the Average Cu and Ni Weight Percent for the Braidwood Unit 2 Surveillance Weld Material Only (Heat # 442011)

Reference	Weight % Copper	Weight % Nickel
WCAP-14824, Rev.2[a]	0.033	0.708
Charpy FW-38[b]	0.044	0.703
Charpy FW-34[b]	0.036	0.774
Charpy FW-41[b]	0.037	0.747
Charpy FW-43[b]	0.042	0.670
Surveillance Weld Average[c]	**0.03**	**0.71**

Notes:
(a) This is the average of 32 data points.
(b) Charpy Specimens From Capsule W of Braidwood Unit 2 (Ref. 10).
(c) The Surveillance weld average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-10
Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Units 1 & 2 Weld Material (Using Braidwood 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W WQ: BAW-2261	Ref. 12	0.028	0.63
B&W WQ: BAW-2261	Ref. 12	0.03	0.65
B&W WQ: BAW-2261	Ref. 12	0.04	0.67
Braidwood Unit 2 Surv. Data Ave.[a]	Table 4-5	0.034	0.710
Braidwood Unit 1 Surv. Data Ave.[b]	Ref. 11	0.032	0.671
BEST ESTIMATE AVERAGE		**0.03[c]**	**0.67[c]**

Notes:

(a) The weld material in the Braidwood Unit 2 surveillance program was made of the same wire and flux type as the reactor vessel inter. to lower shell girth seam weld (Weld seam WF-562, Wire Heat No. 442011, Flux Type Linde 80, Flux Lot No. 8061). The surveillance weld flux lot # is 0344.

(b) The Braidwood Unit 1 surveillance weld is identical to that used in the Braidwood Unit 2 reactor vessel core region girth seam (WF-562) heat number 442011, with a Linde 80 type flux, lot number 8061.

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-11
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[(a)]**
Closure Head Flange 3P6566/5P7547/4P6986	---	0.75	20
Vessel Flange 124P455	0.07	0.70	20
Nozzle Shell Forging 5P7056	0.04	0.90	30
Intermediate Shell Forging 49D963-1/49C904-1	0.03	0.71	-30
Lower Shell Forging 50D102-1 / 50C97-1	0.06	0.76	-30
Inter. to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	0.03	0.67	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.04	0.46	-25
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03, 0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

TABLE 4-12

Calculation of Chemistry Factors for Braidwood Unit 2 using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	0.0[e]	0	1.119
(Tangential)	W	2.25	1.220	4.53	5.53	1.488
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	33.94	35.91	1.119
(Axial)	W	2.25	1.220	33.2	40.50	1.488
				SUM:	81.94	6.328
	CF = Σ(FF * RT_{NDT}) ÷ Σ(FF^2) = (81.94) ÷ (6.328) = 12.9°F					
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Metal	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d, e]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Metal	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	CF = Σ(FF * RT_{NDT}) ÷ Σ(FF^2) = (161.19) ÷ (6.273) = 25.7°F					

Notes:

(a) f = Calculated fluence from the Braidwood Unit 2 capsule W dosimetry analysis results [10],the Braidwood Unit 1 calculated fluences are from capsule W analysis[11], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 2 taken from WCAP-15369[10]. Braidwood Unit 1 values are from WCAP-15316 Rev 1 (Ref. 11).

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted (i.e. the ratio factor is 1.0).

(e) Actual values of ΔRT_{NDT} are -9.73 (Cap U Tang.), -9.42 (Cap. X Tang.), -0.13 (Cap. U Axial), -0.58 (Cap U Weld). This physically should not occur; therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.

TABLE 4-13
Summary of the Braidwood Unit 2 Reactor Vessel Beltline Material Chemistry Factors
Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell Forging 49D963-1/49C904-1	20.0°F	---
Lower Shell Forging 50D102-1/50C97-1	37.0°F	12.9°F
Nozzle Shell Forging 5P7056	26.0°F	---
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	54.0°F	---
Braidwood Unit 1 & 2 Surveillance Program Weld Metal	41.0°F	---

Contained in Tables 4-14 through 4-17 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Braidwood Unit 2 reactor vessel beltline materials for 12, 16, 22 and 32 EFPY.

TABLE 4-14
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 12 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	4.46×10^{18}	0.775	1.61×10^{18}	0.519
Lower Shell Forging 50D102-1/50C97-1	4.46×10^{18}	0.775	1.61×10^{18}	0.519
Nozzle Shell Forging 5P-7056	1.30×10^{18}	0.471	4.68×10^{17}	0.282
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	4.32×10^{18}	0.767	1.56×10^{18}	0.512
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	1.30×10^{18}	0.471	4.68×10^{17}	0.282

TABLE 4-15
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 16 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	5.93×10^{18}	0.854	2.14×10^{18}	0.586
Lower Shell Forging 50D102-1/50C97-1	5.93×10^{18}	0.854	2.14×10^{18}	0.586
Nozzle Shell Forging 5P-7056	1.72×10^{18}	0.534	6.19×10^{17}	0.328
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	5.73×10^{18}	0.844	2.07×10^{18}	0.578
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	1.72×10^{18}	0.534	6.19×10^{17}	0.328

TABLE 4-16
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	8.11×10^{18}	0.941	2.92×10^{18}	0.663
Lower Shell Forging 50D102-1/50C97-1	8.11×10^{18}	0.941	2.92×10^{18}	0.663
Nozzle Shell Forging 5P-7056	2.35×10^{18}	0.609	8.47×10^{17}	0.384
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	7.87×10^{18}	0.933	2.84×10^{18}	0.656
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.35×10^{18}	0.609	8.47×10^{17}	0.384

TABLE 4-17
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	1.18×10^{19}	1.046	4.24×10^{18}	0.762
Lower Shell Forging 50D102-1/50C97-1	1.18×10^{19}	1.046	4.24×10^{18}	0.762
Nozzle Shell Forging 5P-7056	3.40×10^{18}	0.703	1.23×10^{18}	0.460
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.13×10^{19}	1.034	4.09×10^{18}	0.752
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	3.40×10^{18}	0.703	1.23×10^{18}	0.460

Contained in Tables 4-18 through 4-25 are the calculations of the ART values used for the generation of the 12, 16, 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-18

Calculation of the ART Values for the 1/4T Location @ 12 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 12[a] EFPY	¼-t f (x 10^{19})	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.743	0.446	0.775	-30	15.5	0	7.75	15.5	1
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.743	0.446	0.775	-30	28.7	0	14.35	28.7	27
Lower shell Forging → using S/C Data				12.9	0.743	0.446	0.775	-30	10.0	0	17.0	34.0	14
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.720	0.432	0.767	40	31.4	0	15.7	31.4	103
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.720	0.432	0.767	40	19.7	0	9.85	19.7	79
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.216	0.130	0.471	30	12.2	0	6.1	12.2	54
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.216	0.130	0.471	-25	25.4	0	12.7	25.4	26

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-19

Calculation of the ART Values for the 3/4T Location @ 12 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 12[a] EFPY	¾-t f (x 10^{19})	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.743	0.161	0.519	-30	10.4	0	5.2	10.4	-9
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.743	0.161	0.519	-30	19.2	0	9.6	19.2	8
Lower shell Forging → using S/C Data				12.9	0.743	0.161	0.519	-30	6.7	0	17.0	34.0	11
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.720	0.156	0.512	40	21.0	0	10.5	21.0	82
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.720	0.156	0.512	40	13.2	0	6.6	13.2	66
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.216	0.0468	0.282	30	7.3	0	3.65	7.3	45
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.216	0.0468	0.282	-25	15.2	0	7.6	15.2	5

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-20

Calculation of the ART Values for the 1/4T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 16[(a)] EFPY	¼-t f (x 10^{19})	¼-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.987	0.593	0.854	-30	17.1	0	8.55	17.1	4
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.987	0.593	0.854	-30	31.6	0	15.8	31.6	33
Lower shell Forging → using S/C Data				12.9	0.987	0.593	0.854	-30	11.0	0	17.0	34.0	15
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.954	0.573	0.844	40	34.6	0	17.0	34.0	109
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.954	0.573	0.844	40	21.7	0	10.85	21.7	83
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.286	0.172	0.534	30	13.9	0	6.69	13.9	58
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.286	0.172	0.534	-25	28.8	0	14.4	28.8	33

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-21

Calculation of the ART Values for the 3/4T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 16[a] EFPY	¾-t f ($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.987	0.214	0.586	-30	11.7	0	5.85	11.7	-7
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.987	0.214	0.586	-30	21.7	0	10.85	21.7	13
Lower shell Forging → using S/C Data				12.9	0.987	0.214	0.586	-30	7.6	0	17.0	34.0	12
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.954	0.207	0.578	40	23.7	0	11.85	23.7	87
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.954	0.207	0.578	40	14.9	0	7.45	14.9	70
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.286	0.0619	0.328	30	8.5	0	4.25	8.5	47
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.286	0.0619	0.328	-25	17.7	0	8.85	17.7	10

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-22

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY	¼-t f (x 10^{19})	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.35	0.811	0.941	-30	18.8	0	9.4	18.8	8
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.35	0.811	0.941	-30	34.8	0	17.0	34.0	39
Lower shell Forging → using S/C Data				12.9	1.35	0.811	0.941	-30	12.1	0	17.0	34.0	16
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.31	0.787	0.933	40	38.3	0	19.15	38.3	117
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.31	0.787	0.933	40	24.0	0	12.0	24.0	88
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.391	0.235	0.609	30	15.8	0	7.9	15.8	62
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.391	0.235	0.609	-25	32.9	0	16.45	32.9	41

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-23
Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY	¾-t f (x 10^{19})	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.35	0.292	0.663	-30	13.3	0	6.65	13.3	-3
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.35	0.292	0.663	-30	24.5	0	12.25	24.5	19
Lower shell Forging → using S/C Data				12.9	1.35	0.292	0.663	-30	8.6	0	17.0	34.0	13
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.31	0.284	0.656	40	26.9	0	13.45	26.9	94
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.31	0.284	0.656	40	16.9	0	8.45	16.9	74
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.391	0.0847	0.384	30	10.0	0	5.0	10.0	50
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.391	0.0847	0.384	-25	20.7	0	10.35	20.7	16

NOTES:
(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).
(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(c) ΔRT_{NDT} = CF * FF
(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-24

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF(d)	f@32(a) EFPY	¼-t f ($x 10^{19}$)	¼-t FF	I	ΔRT_{NDT}(c)	σ_I	σ_Δ	M	ART(b)
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.96	1.18	1.046	-30	20.9	0	10.45	20.9	12
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.96	1.18	1.046	-30	38.7	0	17.0	34.0	43
Lower shell Forging → using S/C Data				12.9	1.96	1.18	1.046	-30	13.5	0	17.0	34.0	18
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.89	1.13	1.034	40	42.4	0	21.2	42.4	125
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.89	1.13	1.034	40	26.6	0	13.3	26.6	93
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.567	0.340	0.703	30	18.3	0	9.15	18.3	67
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.567	0.340	0.703	-25	38.0	0	19.0	38.0	51

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-25

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY	¾-t f ($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.96	0.424	0.762	-30	15.2	0	7.6	15.2	0
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.96	0.424	0.762	-30	28.2	0	14.1	28.2	26
Lower shell Forging → using S/C Data				12.9	1.96	0.424	0.762	-30	9.8	0	17.0	34.0	14
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.89	0.409	0.752	40	30.8	0	15.4	30.8	102
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.89	0.409	0.752	40	19.3	0	9.65	19.3	79
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.567	0.123	0.460	30	12.0	0	6.0	12.0	54
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.567	0.123	0.460	-25	24.8	0	12.4	24.8	25

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

The girth weld WF-562 and the nozzle shell forging 5P-7056 are the limiting beltline materials for all heatup and cooldown curves to be generated. The ART value associated with these materials will be used in all four sets of curves. The girth weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw). The ART associated with the limiting axial material must also be considered to determine if this case would be more conservative or overlap the circ. flaw curves. Contained in Tables 4-26 through 4-29 is a summary of the limiting ARTs to be used in the generation of the Braidwood Unit 2 reactor vessel heatup and cooldown curves.

TABLE 4-26
Summary of Adjusted Reference Temperature (ART) values at the1/4T and 3/4T Locations for 12 EFPY

Material	**12 EFPY**	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging 49D963-1/49C904-1	1	-9
Lower Shell Forging 50D102-1/50C97-1	27	8
- Using Surveillance Data	14	11
Circumferential Weld WF-562	103	82
- Using Surveillance Data	79(a)	66(a)
Nozzle Shell Forging 5P-7056	54(b)	45(b)
Circumferential Weld WF-645	26	5

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-1 and 5-2. See note (b).

(b) These ART values, using the '96 App. G Methodology, produced a more conservative curve (heatup and cooldown) with no overlap versus those curves generated with the circ. flaw ART values and Code Case N-588 Methodology.

TABLE 4-27

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 16 EFPY

Material	16 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	4	-7
Lower Shell Forging 50D102-1/50C97-1	33	13
- Using Surveillance Data	15	12
Circumferential Weld WF-562	109	87
- Using Surveillance Data	83(a)	70(a)
Nozzle Shell Forging 5P-7056	58(b)	47(b)
Circumferential Weld WF-645	33	10

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-3 and 5-4. See note (b).

(b) These ART values, using the '96 App. G Methodology, produced a more conservative curve (heatup and cooldown) with no overlap versus those curves generated with the circ. flaw ART values and Code Case N-588 Methodology.

TABLE 4-28

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 22 EFPY

Material	22 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	8	-3
Lower Shell Forging 50D102-1/50C97-1	39	19
- Using Surveillance Data	16	13
Circumferential Weld WF-562	117	94
- Using Surveillance Data	88[a]	74[a]
Nozzle Shell Forging 5P-7056	62[b]	50[b]
Circumferential Weld WF-645	41	16

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-5 and 5-6. See note (b).

(b) These ART values, using the '96 App. G Methodology, produced a more conservative curve (heatup and cooldown) with no overlap versus those curves generated with the circ. flaw ART values and Code Case N-588 Methodology.

TABLE 4-29

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 32 EFPY

Material	32 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	12	0
Lower Shell Forging 50D102-1/50C97-1	43	26
- Using Surveillance Data	18	14
Circumferential Weld WF-562	125	102
- Using Surveillance Data	93[a]	79[a]
Nozzle Shell Forging 5P-7056	67[b]	54[b]
Circumferential Weld WF-645	51	25

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-7 and 5-8. See note (b).

(b) These ART values, using the '96 App. G Methodology, produced a more conservative curve (heatup and cooldown) with no overlap versus those curves generated with the circ. flaw ART values and Code Case N-588 Methodology.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-8 present the 12, 16, 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6] and Code Case N-588[5], respectively. The heatup and cooldown curves that are presented herein are actually curves generated using the 1996 App. G methodology with the *lower axial flaw ART value*. The reason is these curves are more conservative than the curves generated using Code Case N-588 methodology with the *higher circ. flaw ART value*. This is true throughout the entire temperature range, including criticality.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-8. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1, 5-3, 5-5 and 5-7 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Braidwood Unit 2 reactor vessel at 12, 16, 22 and 32 EFPY are 114°F, 118°F, 122°F and 127°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-8 define all of the above limits for ensuring prevention of nonductile failure for the Braidwood Unit 2 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-8 are presented in Tables 5-1 through 5-8, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 12 EFPY: 1/4T, 79°F (N-588) & 54°F ('96 App. G)
3/4T, 66°F (N-588) & 45°F ('96 App. G)

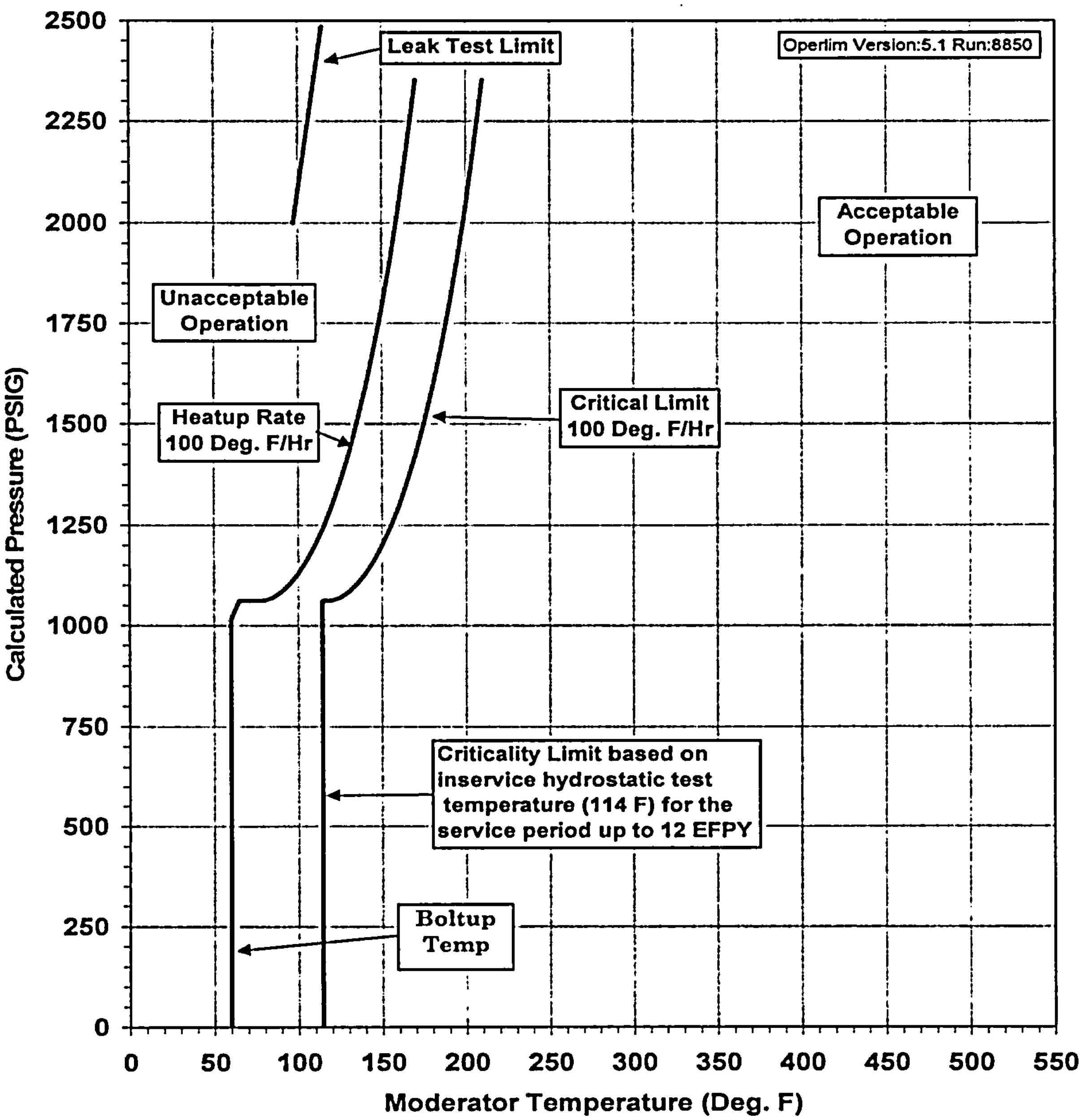

FIGURE 5-1 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 12 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 12 EFPY: 1/4T, 79°F (N-588) & 54°F ('96 App. G)
3/4T, 66°F (N-588) & 45°F ('96 App. G)

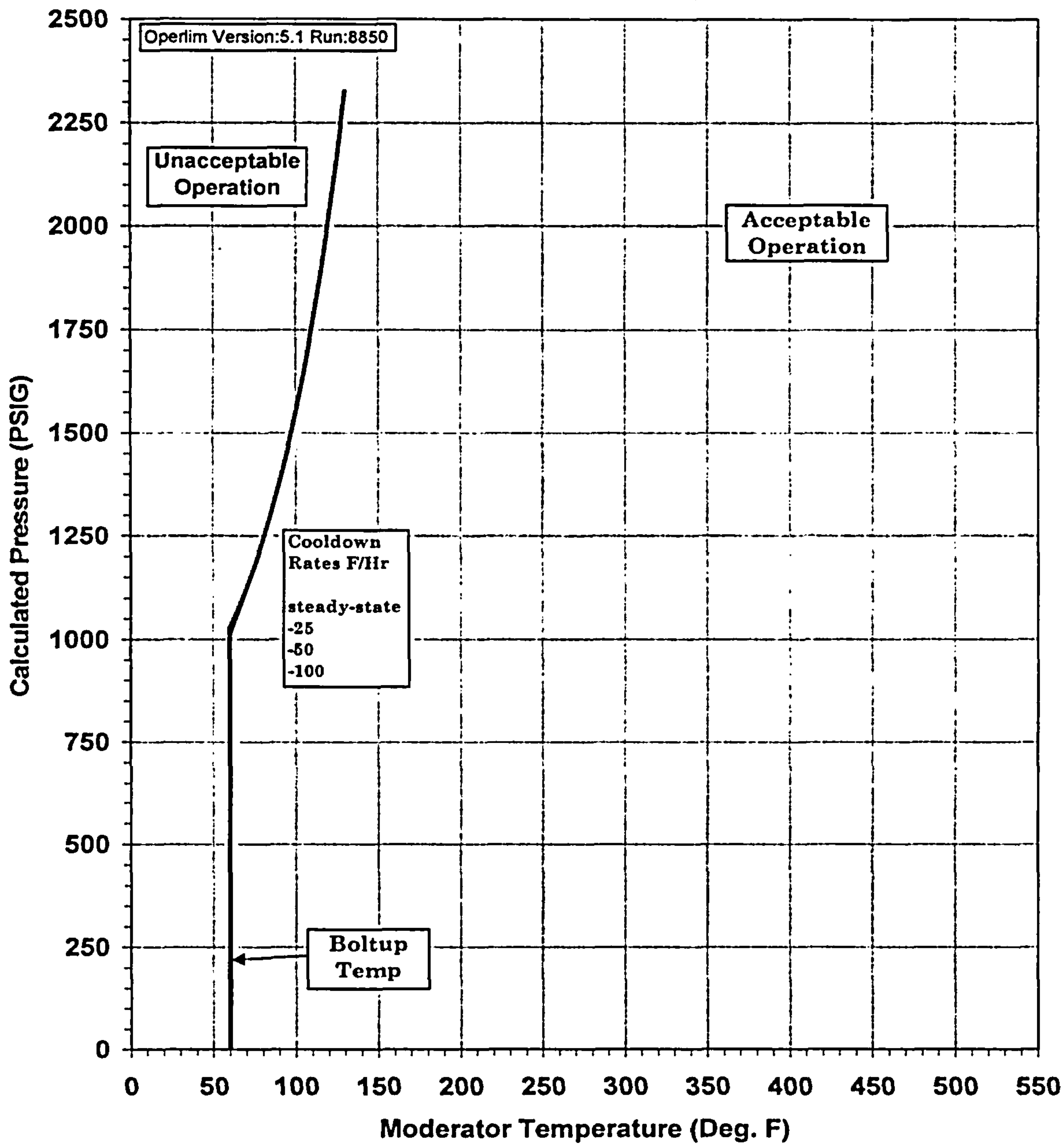

FIGURE 5-2 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 12 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

TABLE 5-1
Braidwood Unit 2 Heatup Data at 12 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	114	0	97	2000
60	1014	114	1061	114	2485
65	1061	115	1061		
70	1061	120	1062		
75	1061	125	1069		
80	1062	130	1083		
85	1069	135	1103		
90	1083	140	1128		
95	1103	145	1161		
100	1128	150	1199		
105	1161	155	1244		
110	1199	160	1297		
115	1244	165	1357		
120	1297	170	1425		
125	1357	175	1501		
130	1425	180	1587		
135	1501	185	1683		
140	1587	190	1791		
145	1683	195	1910		
150	1791	200	2042		
155	1910	205	2189		
160	2042	210	2352		
165	2189				
170	2352				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-2

Braidwood Unit 2 Cooldown Data at 12 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1028	60	1017	60	1011		
65	1073	65	1067				
70	1122						
75	1177						
80	1237						
85	1304						
90	1377						
95	1459						
100	1549						
105	1648						
110	1758						
115	1880						
120	2014						
125	2162						
130	2326						

* Note: The computer run for the '96 App. G using the highest <u>Axial Flaw</u> ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 16 EFPY: 1/4T, 83°F (N-588) & 58°F ('96 App. G)
3/4T, 70°F (N-588) & 47°F ('96 App. G)

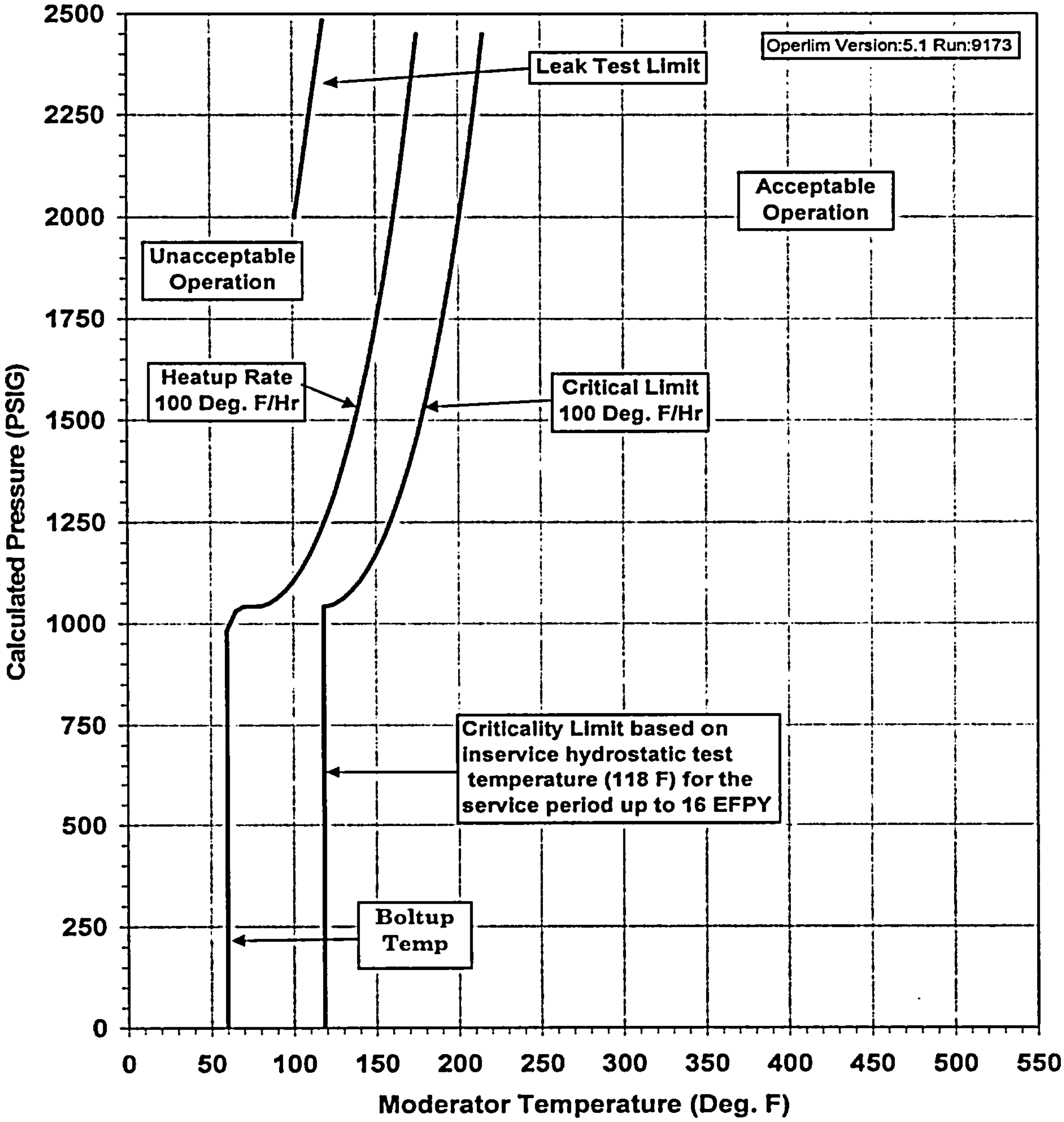

FIGURE 5-3 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 16 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 16 EFPY: 1/4T, 83°F (N-588) & 58°F ('96 App. G)
3/4T, 70°F (N-588) & 47°F ('96 App. G)

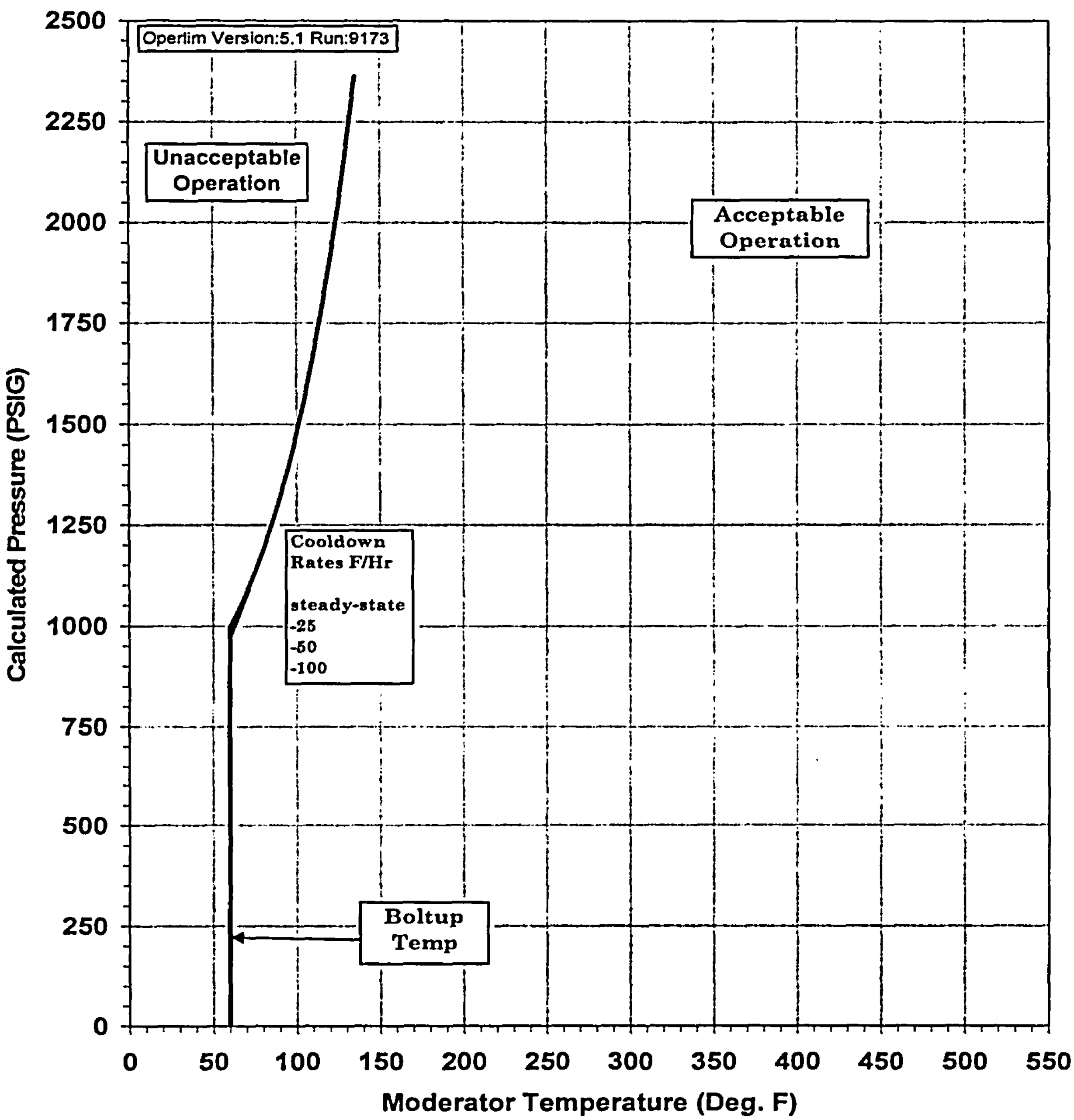

FIGURE 5-4 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 16 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

TABLE 5-3
Braidwood Unit 2 Heatup Data at 16 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	118	0	101	2000
60	983	118	1029	118	2485
65	1029	118	1042		
70	1042	120	1042		
75	1042	125	1048		
80	1042	130	1061		
85	1048	135	1079		
90	1061	140	1104		
95	1079	145	1134		
100	1104	150	1171		
105	1134	155	1214		
110	1171	160	1264		
115	1214	165	1322		
120	1264	170	1387		
125	1322	175	1460		
130	1387	180	1543		
135	1460	185	1635		
140	1543	190	1738		
145	1635	195	1852		
150	1738	200	1980		
155	1852	205	2121		
160	1980	210	2277		
165	2121	215	2450		
170	2277				
175	2450				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-4

Braidwood Unit 2 Cooldown Data at 16 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	995	60	980	60	970		
65	1037	65	1026	65	1022		
70	1082	70	1077				
75	1133						
80	1188						
85	1250						
90	1318						
95	1393						
100	1476						
105	1568						
110	1669						
115	1781						
120	1905						
125	2042						
130	2194						
135	2361						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY 1/4T, 88°F (N-588) & 62°F ('96 App. G)
3/4T, 74°F (N-588) & 50°F ('96 App. G)

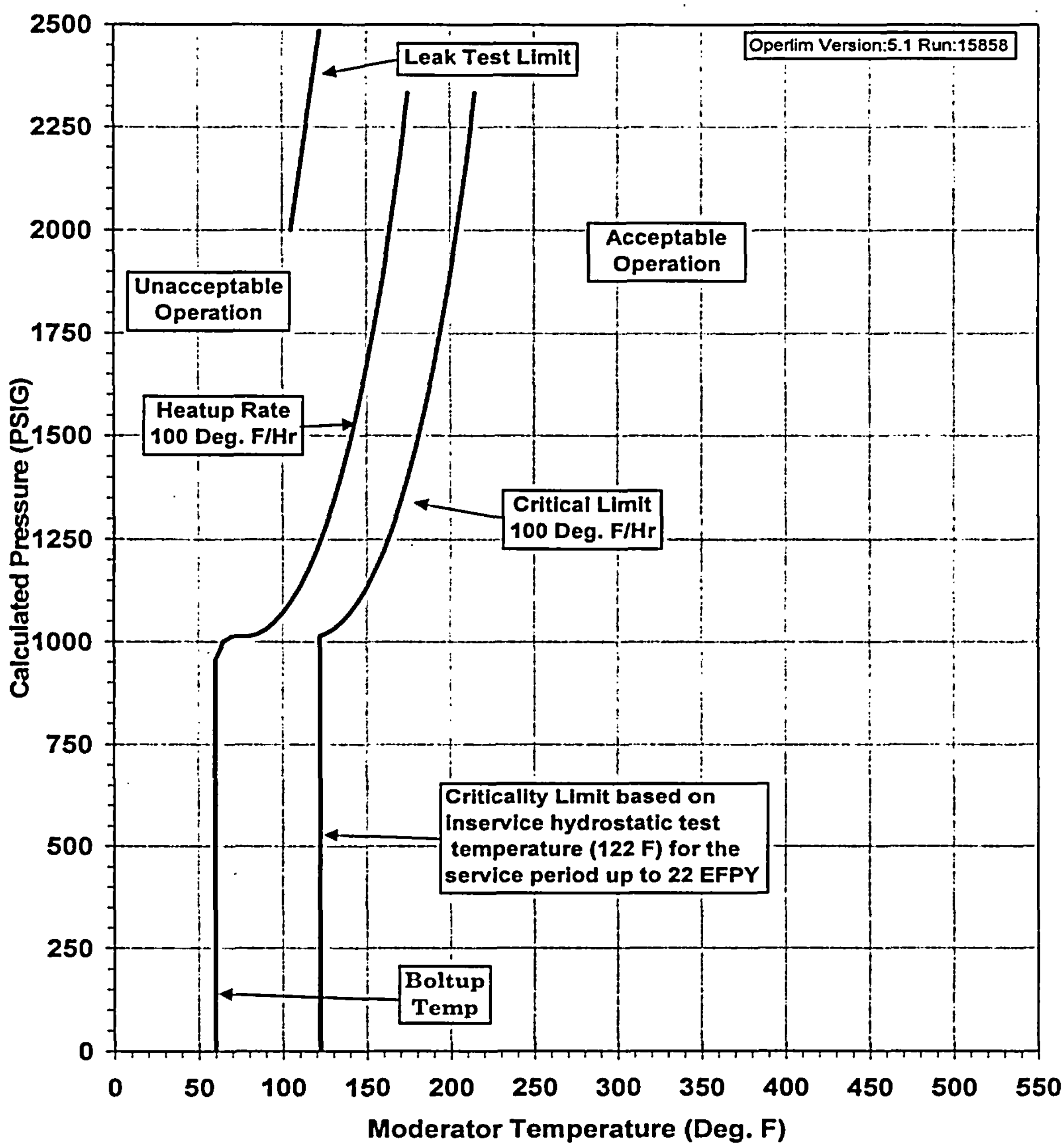

FIGURE 5-5 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 88°F (N-588) & 62°F ('96 App. G)
3/4T, 74°F (N-588) & 50°F ('96 App. G)

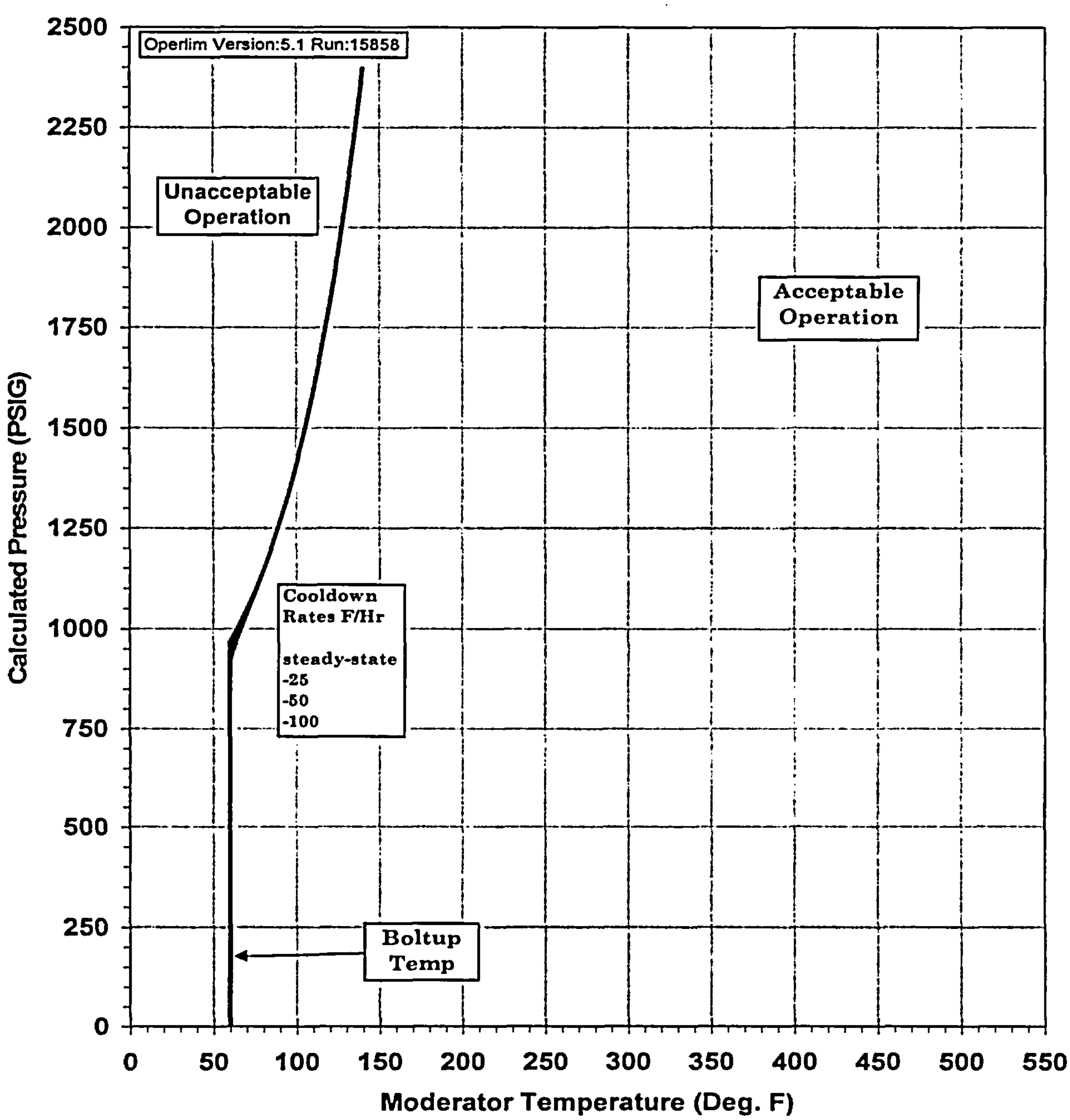

FIGURE 5-6 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

TABLE 5-5
Braidwood Unit 2 Heatup Data at 22 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	122	0	105	2000
60	956	122	1000	122	2485
65	1000	122	1013		
70	1013	125	1018		
75	1013	130	1030		
80	1013	135	1046		
85	1018	140	1069		
90	1030	145	1097		
95	1046	150	1131		
100	1069	155	1171		
105	1097	160	1218		
110	1131	165	1272		
115	1171	170	1333		
120	1218	175	1402		
125	1272	180	1479		
130	1333	185	1566		
135	1402	190	1663		
140	1479	195	1770		
145	1566	200	1890		
150	1663	205	2023		
155	1770	210	2170		
160	1890	215	2332		
165	2023				
170	2170				
175	2332				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-6

Braidwood Unit 2 Cooldown Data at 22 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	965	60	946	60	932	60	921
65	1003	65	989	65	980		
70	1045	70	1036	70	1033		
75	1092	75	1088				
80	1143						
85	1200						
90	1263						
95	1332						
100	1409						
105	1494						
110	1587						
115	1691						
120	1805						
125	1932						
130	2071						
135	2226						
140	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY 1/4T, 93°F (N-588) & 67°F ('96 App. G)
3/4T, 79°F (N-588) & 54°F ('96 App. G)

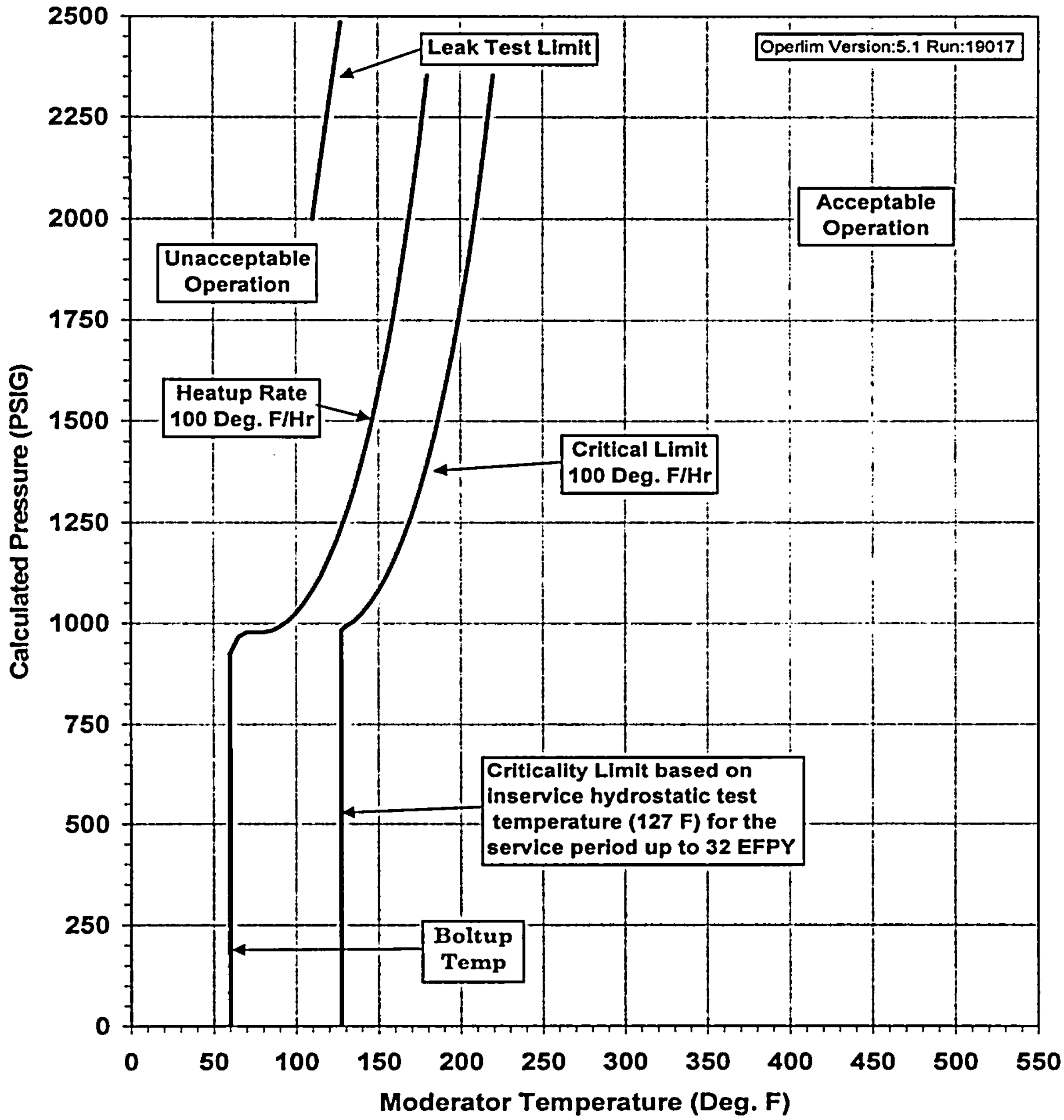

FIGURE 5-7 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 93°F (N-588) & 67°F ('96 App. G)
3/4T, 79°F (N-588) & 54°F ('96 App. G)

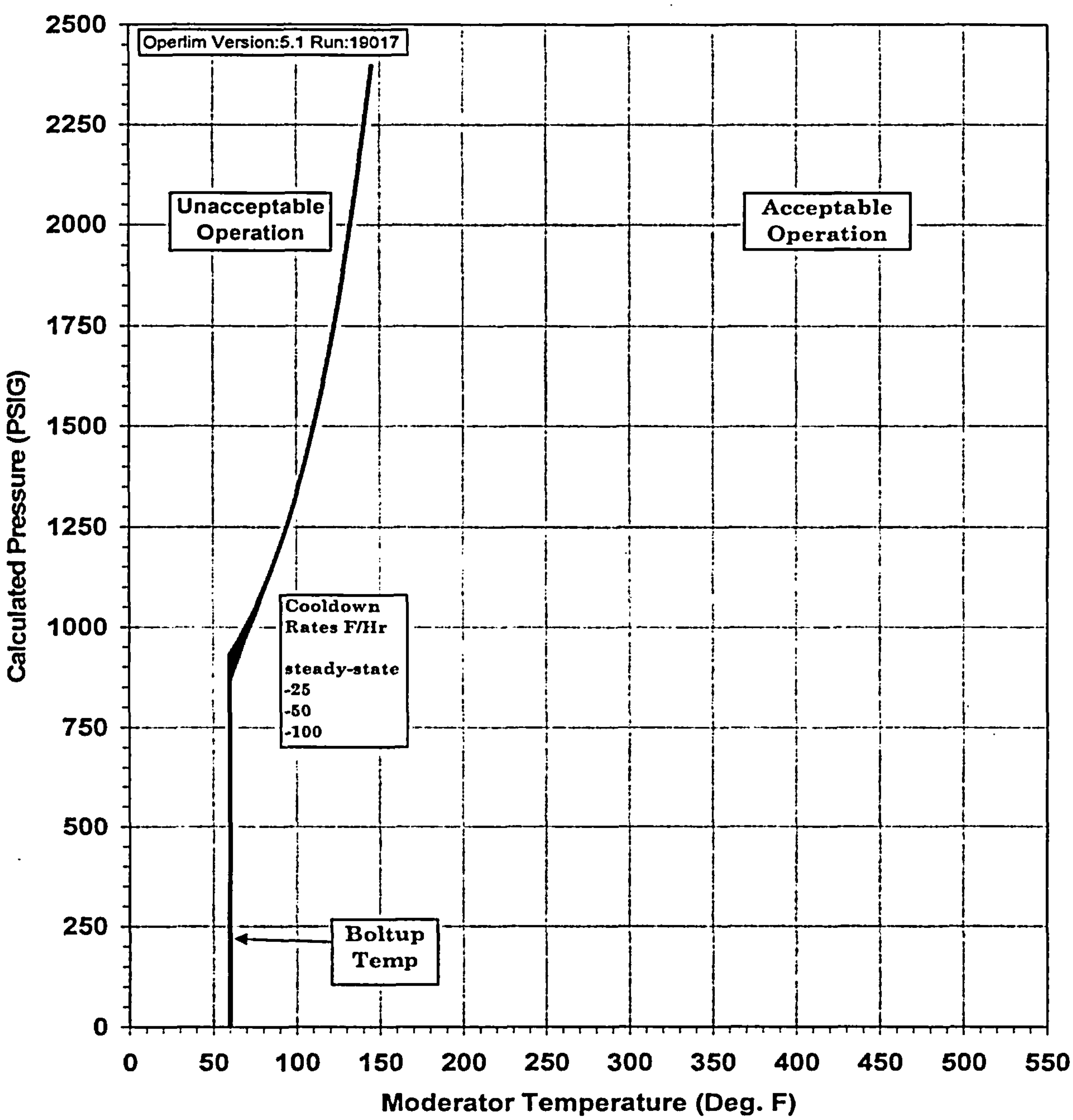

FIGURE 5-8 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins for Instrumentation Errors)

TABLE 5-7
Braidwood Unit 2 Heatup Data at 32 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	127	0	110	2000
60	924	127	965	127	2485
65	965	127	979		
70	977	127	977		
75	977	127	981		
80	977	130	990		
85	981	135	1005		
90	990	140	1025		
95	1005	145	1051		
100	1025	150	1081		
105	1051	155	1118		
110	1081	160	1161		
115	1118	165	1210		
120	1161	170	1266		
125	1210	175	1329		
130	1266	180	1400		
135	1329	185	1480		
140	1400	190	1569		
145	1480	195	1668		
150	1569	200	1778		
155	1668	205	1901		
160	1778	210	2036		
165	1901	215	2186		
170	2036	220	2353		
175	2186				
180	2353				

* Note: The computer run for the '96 App. G using the highest <u>Axial Flaw</u> ART value generated the most conservative curve overall.. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-8
Braidwood Unit 2 Cooldown Data at 32 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	931	60	908	60	889	60	866
65	965	65	946	65	932	65	921
70	1003	70	989	70	980		
75	1045	75	1036	75	1033		
80	1092	80	1088				
85	1143						
90	1200						
95	1263						
100	1332						
105	1409						
110	1494						
115	1587						
120	1691						
125	1805						
130	1932						
135	2071						
140	2226						
145	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

6 REFERENCES

1 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.

2 ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

3 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

4 WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2", K.R. Hsu, November 2003.

5 ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

6 ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", 1996.

7 WCAP-15368, "Commonwealth Edison Company Braidwood Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, March 2000.

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12 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration For Byron and Braidwood", T. J. Laubham, et al., November 1997. Ref. Errata letter CAE-97-233, CCE-97-316, "Transmittal of Updated Tables to WCAP-14824 Rev. 2"

13 Nuclear Design Information Transmittal NDIT No. BRW-DIT-97-321, Rev. 0, "Braidwood Station Historical Tcold Data for Units 1 & 2", Dated October 22, 1997.

14 Nuclear Design Information Transmittal NDIT No. BRW-DIT-2000-0010, Rev. 0, "Braidwood Station Historical Tcold Data for Units 1 & 2 Cycle 7", Dated January 20, 2000.

15 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

16 WCAP-11188, "Commonwealth Edison Company Braidwood Station Unit No. 2 Reactor Vessel Radiation Surveillance Program", L.R. Singer, December 1986.

17 WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program", E. Terek, et al., March 1991.

18 Analytical Request #15482, "Braidwood Nuclear Plant, Unit 2, Irradiated Low Alloy Steel Reactor Surveillance Dosimetry", L. Kardos, dated October 27, 1994.

APPENDIX A

Thermal Stress Intensity Factors (K_{It})

TABLE A1

K_{It} Values for 100°F/hr Heatup Curve (12 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
PT Curves are Limited by the ¼ T Location up to 65°F and ¾ T Limited for the Remainder of the Curve				
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892

Vessel Radius to the ¼T and ¾T Locations are as follows:

- 1/4T Radius = 88.750"
- 3/4T Radius = 93.000"

TABLE A2
K_{It} Values for 100°F/hr Cooldown Curve (12 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: The 100°F/hr Cooldown is limited by the lower rates at T = 60 to 70°F, and by the Steady State condition thereafter.

TABLE A3
K_{It} Values for 100°F/hr Heatup Curve (16 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
PT Curves are Limited by the ¼ T Location up to 65°F and ¾ T Limited for the Remainder of the Curve				
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892
175	152.82	-13.5158	129.06	10.2056

TABLE A4
K_{It} Values for 100°F/hr Cooldown Curve (16 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
135	159.95	15.4852
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: The 100°F/hr Cooldown is limited by the lower rates at T = 60 to 70°F, and by the Steady State condition thereafter.

TABLE A5
K_{It} Values for 100°F/hr Heatup Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
PT Curves are Limited by the ¼ T Location up to 65°F and ¾ T Limited for the Remainder of the Curve				
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892
175	152.82	-13.5158	129.06	10.2056

TABLE A6

K_{It} Values for 100°F/hr Cooldown Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
140	165.04	15.5523
135	159.95	15.4852
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: The 100°F/hr Cooldown is limited by the lower rates at T = 65 to 75°F, and by the Steady State condition thereafter.

TABLE A7
KIt Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
PT Curves are Limited by the ¼ T Location up to 65°F and ¾ T Limited for the Remainder of the Curve				
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892
175	152.82	-13.5158	129.06	10.2056
180	157.63	-13.6533	133.64	10.3136

ATTACHMENT 4-C

WCAP-15391, Revision 1

Byron Unit 1
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Non-Proprietary Class 3

WCAP-15391
Revision 1

November 2003

Byron Unit 1
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-15391, Revision 1

Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

November 2003

Approved: J. A. Gresham

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: J. H. Ledger *J. H. L*

Revision 1:

This revision was completed to incorporate an updated reference for the flange elimination from the PT limit curves. In the previous revision, WCAP-15315 was provided as justification for the flange notch elimination and it was changed to WCAP-16143-P in this revision. In addition to this change, the thermal stress intensity factors for the highest heatup and cooldown rates were added to this report in Appendix A.

EXECUTIVE SUMMARY

The purpose of this report is to generate new pressure-temperature limit curves for Byron Unit 1 for normal operation at 22 and 32 EFPY based of revised Uprated Fluences. The new pressure-temperature limit curves were generated using the methodology from WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-16143-P ("Flange Notch" Elimination Justification). Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART). The 1/4T and 3/4T ART values are summarized in Tables 4-18 and 4-19 and the limiting material is the intermediate shell forging 5P-5933. The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, Code Case N-588 was not used and only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-4.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Byron Station Unit No. 1 reactor pressure vessel (Specifically from forging 5P-5933). Also supplied was a weldment made with weld wire heat # 442002 Linde 80 flux lot number 8873, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company (now known as Exelon Nuclear) contracted Westinghouse to generate new pressure-temperature limit curves for 22 and 32 EFPY based on the revised fluences from the 5% Uprating. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10CFR Part 50[3] per WCAP-16143-P, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"[4],
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G,[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, PT Curves will be generated consisting of the 1996 Appendix G to ASME Section XI and the K_{Ic} methodologies (ASME Code Case N-640) for the limiting forging/base metal material. All curves will use the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-16143-P). The final PT curves to be presented herein will be the most limiting set of curves.

The purpose of this report is to present the calculations and the development of the Byron Unit 1 heatup and cooldown curves for 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Exelon, the surveillance weld data from the Byron Unit 1 and Unit 2 surveillance programs has been integrated. Note that Byron Unit 2 surveillance weld is identical to the surveillance weld (Heat No. 442002) at Byron Unit 1. In addition, the Braidwood Units 1 and 2 surveillance weld is identical to the nozzle to intermediate shell circumferential weld (Heat No. 442011). Per WCAP-15183[7], WCAP-15180[11] and WCAP-15368[17] all weld metal surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ic}	=	function of temperature relative to the RT_{NDT} of the material
C	=	2.0 for Level A and Level B service limits
C	=	1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \qquad (3)$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

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

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

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

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (4)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] addresses 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 preservice hydrostatic test pressure (3106 psi), which is 621 psig for Byron Unit 1 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, 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 pressure limitation of 20 percent of the hydro-test pressure were developed using the K_{IA} fracture toughness from the mid 1970's.

Improved knowledge of the 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 development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,"[4] concluded that the integrity of the closure head/vessel flange region is not a concern for Byron or Braidwood, therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta\ RT_{NDT} + Margin \qquad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta\ RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (9)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections for the 5% Uprating and the results are presented in SAE-REA-00-546[10]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 4-1, herein, contains the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Byron Unit 1 reactor vessel. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-1 and 4-2.

Ratio Procedure and Temperature Adjustment:

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT,\ measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\ measured} + 1.0*(T_{capsule} - T_{plant}) \quad (10)$$

The irradiation temperatures from Byron Units 1 & 2 are presented in WCAP-14824, Revision 2[12]. The average irradiation temperature from each of the four Units and operating cycles in question is 553°F. Therefore, no temperature adjustment is required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-5 through 4-8. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, the surveillance weld data for Byron Unit 1 is credible while the surveillance forging material is non-credible. In addition, Reference 7 also shows that the Table chemistry factor is conservative. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the adjusted reference temperature of the forging material with a full margin term. Per Reference 11, the surveillance weld data for Byron Unit 2 is credible. Position 2.1 chemistry factors are calculated in Table 4-9 and 4-10.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Byron Unit 1, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}\ .$$

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Byron Unit 1, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I.

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-1

Summary of the Peak Pressure Vessel Neutron Fluence Values at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	Surface[b] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 5P-5933	1.39×10^{19}	8.35×10^{18}	3.01×10^{18}
Lower Shell Forging 5P-5951	1.39×10^{19}	8.35×10^{18}	3.01×10^{18}
Nozzle Shell Forging 123J218	4.15×10^{18}	2.49×10^{18}	8.99×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat 442002)	1.34×10^{19}	8.05×10^{18}	2.90×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-501 (Heat 442011)	4.15×10^{18}	2.49×10^{18}	8.99×10^{17}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	Surface[b] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 5P-5933	2.02×10^{19}	1.21×10^{19}	4.37×10^{18}
Lower Shell Forging 5P-5951	2.02×10^{19}	1.21×10^{19}	4.37×10^{18}
Nozzle Shell Forging 123J218	6.04×10^{18}	3.63×10^{18}	1.31×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	1.94×10^{19}	1.16×10^{19}	4.20×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF501 (Heat 442011)	6.04×10^{18}	3.63×10^{18}	1.31×10^{18}

Notes:
(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.
(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-3
Calculated Integrated Neutron Exposure of the Byron Unit 1 Surveillance Capsules Tested to Date

Capsule	Fluence(a)
U	4.04×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.57×10^{19} n/cm^2, (E > 1.0 MeV)
W	2.43×10^{19} n/cm^2, (E > 1.0 MeV)

Notes:
(a) Documented in WCAP-15123[18].

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[16], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift(a)
Intermediate Shell Forging 5P-5933 (Tangential Orientation)	U	28.55°F
	X	9.82°F
	W	49.2°F
Intermediate Shell Forging 5P-5933 (Axial Orientation)	U	18.52°F
	X	53.03°F
	W	29.34°F
Surveillance Program Weld Metal	U	5.61°F
	X	40.11°F
	W	51.34°F
Heat Affected Zone	U	-60.2°F
	X	13.45°F
	W	15.23°F

Notes:
(a) Documented in WCAP-15123[18].

Table 4-5 contains the best estimate weight percent copper and nickel for the Byron Unit 1 base materials in the beltline region. Table 4-6 contains the best estimate weight percent copper and nickel for the Byron Unit 1 surveillance weld material, while Table 4-7 presents the overall best estimate average for that heat of weld. Table 4-8 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-9 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-11. Tables 4-9 and 4-10 provide the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-11.

TABLE 4-5
Calculation of the Best Estimate Cu and Ni Weight Percent for the Byron Unit 1 Forging Materials

Reference	Intermediate Shell Forging 5P-5933		Lower Shell Forging 5P-5951	
	Cu %	**Ni %**	**Cu %**	**Ni %**
Ref. 12	0.364[(a)]	0.747[(a)]	0.04	0.64
Charpy AL-31[(b)]	0.36	0.70	---	---
Charpy AL-36[(b)]	0.35	0.76	---	---
Best Estimate Average[(c)]	0.04	0.74	0.04	0.64

Notes:
(a) This is the average of 5 data points.
(b) Charpy Specimens from Capsule W of Byron Unit 1 (Ref. 18).
(c) The best estimate average was rounded per ASTM E29, using the "Rounding Method". This calculation is also presented in Reference 14.

TABLE 4-6
Calculation of the Average Cu and Ni Weight Percent for the Byron Unit 1 Surveillance Weld Material Only (Heat # 442002)

Reference	Weight % Copper	Weight % Nickel
Ref. 12	0.022[(a)]	0.69[(a)]
Charpy AW-36[(b)]	0.026	0.72
Charpy AW-40[(b)]	0.027	0.72
Surveillance Weld Average[(c)]	0.02	0.69

Notes:
(a) This is the average of 21 data points.
(b) Charpy Specimens from Capsule W of Byron Unit 1 (Ref. 18).
(c) The best estimate average was rounded per ASTM E29, using the "Rounding Method". This calculation is also presented in Reference 14.

TABLE 4-7

Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Units 1 & 2 Weld Material (Using Byron 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W WQ: BAW-2261	Ref. 12	0.024	0.70
B&W WQ: BAW-2261	Ref. 12	0.031	0.46
B&W WQ: BAW-2261	Ref. 12	0.03	0.72
B&W WQ: BAW-2261	Ref. 12	0.068	0.48
B&W WQ: BAW-2261	Ref. 12	0.053	0.62
B&W WQ: BAW-2261	Ref. 12	0.059	0.62
B&W WQ (from NDIT No. BYR97-346, Rev. 0)	Ref. 12	0.029	0.65
Round Robin Data Ave. on Weld WF-336 (from NDIT No. BRW-DIT-97-391, Rev. 0)	Ref. 12	0.038	0.658
Byron 1 Surveillance Data Ave. [a]	Table 4-6	0.022	0.69
Byron 2 Surveillance Ave. [b]	Ref. 12	0.023	0.712
BEST ESTIMATE AVERAGE		0.04[c, d]	0.63[c, d]

Notes:

(a) The weld material in the Byron Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld. (Weld seam WF-336, Wire Heat # 442002, Flux Type Linde 80, Flux Lot # 8873).

(b) The Byron Unit 2 surveillance weld is identical to that used in the reactor vessel core region girth seam (WF-447). The weld wire is type Linde MnMoNi (Low Cu-P), heat number 442002, with a Linde 80 type flux, lot number 8064.

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method". This calc as presented in Reference 14.

(d) This ave. Cu & Ni excluded two data points (Cu = 0.114, Ni = 0.54 and Cu = 0.148, Ni = 0.60), see Ref. 22.

TABLE 4-8[b]

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 124K358VA1	---	0.74	60
Vessel Flange 123J219VA1	---	0.73	10
Nozzle Shell Forging 123J218[b]	0.05	0.72	30
Intermediate Shell Forging 5P-5933	0.04	0.74	40
Lower Shell Forging 5P-5951	0.04	0.64	10
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	0.04	0.63	-30
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)	0.03	0.67	10
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 4-7 of Reference 14.

TABLE 4-9
Calculation of Chemistry Factors using Byron Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}^{(c)}$	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.404	0.748	28.55	21.36	0.560
	X	1.57	1.124	9.82	11.04	1.263
	W	2.43	1.239	49.20	60.96	1.535
Intermediate Shell Forging 5P-5933 (Axial)	U	0.404	0.748	18.52	13.85	0.560
	X	1.57	1.124	53.03	59.61	1.263
	W	2.43	1.239	29.34	36.35	1.535
				SUM:	203.17	6.716
	$CF_{5P\text{-}5933} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (203.17) \div (6.716) = 30.3°F$					
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	0.404	0.749	11.22 (5.61)[(d)]	8.40	0.561
	X	1.57	1.125	80.22 (40.11)[(d)]	90.25	1.266
	W	2.43	1.239	102.68 (51.34)[(d)]	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	0.405	0.749	16.88 (8.44)[(d)]	12.64	0.561
	W	1.27	1.067	57.76 (28.88)[(d)]	61.63	1.138
	X	2.30	1.225	108.02 (54.01)[(d)]	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv.\ Weld,\ 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) = 65.9°F$					

Notes:

(a) Calculated Capsule Fluence taken from References 18(or Table 4-3 herein) & Reference 19 for Byron Units 1 and 2, respectively.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (Byron Unit 1 from Ref. 18 or Table 4-4 herein, & Byron Unit 2 from Ref. 19).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustments are required.

TABLE 4-10

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}{}^{(c)}$	$FF*\Delta RT_{NDT}$	FF^2
Braidwood Unit 1	U	0.387	0.737	17.06(d)	12.57	0.543
Surveillance	X	1.24	1.060	30.15(d)	31.96	1.124
Weld Heat 442011, WF-501	W	2.09	1.201	49.68(d)	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0(d)	0	0.557
Surveillance	X	1.23	1.058	26.3(d)	27.83	1.119
Weld Heat 442011, WF-501	W	2.25	1.220	23.9(d)	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

Notes:

(a) Calculated Capsule Fluence values taken from References 20 and 21 for Braidwood Units 1 and 2, respectively.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values values (Braidwood Unit 1 from Ref. 20 & Braidwood Unit 2 from Ref. 21).

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

TABLE 4-11

Summary of the Byron Unit 1 Reactor Vessel Beltline Material Chemistry Factors Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	**Position 2.1**
Nozzle Shell Forging 123J218	31.0°F	---
Intermediate Shell Forging 5P-5933	26.0°F	30.3°F
Lower Shell Forging 5P-5951	26.0°F	---
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat 442002)	54.0°F	65.9°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat 442011)	41.0°F	25.7°F
Byron Unit 1 & 2 Surveillance Weld Metal	27.0°F	---
Braidwood Unit 1 & 2 Surveillance Weld Metal	41.0°F	

Contained in Tables 4-12 and 4-13 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Byron Unit 1 reactor vessel beltline materials for 22 and 32 EFPY.

TABLE 4-12
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 5P-5933	8.35×10^{18}	0.949	3.01×10^{18}	0.671
Lower Shell Forging 5P-5951	8.35×10^{18}	0.949	3.01×10^{18}	0.671
Nozzle Shell Forging 123J218	2.49×10^{18}	0.623	8.99×10^{17}	0.396
Inter. to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	8.05×10^{18}	0.939	2.90×10^{18}	0.662
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF501 (Heat 442011)	2.49×10^{18}	0.623	8.99×10^{17}	0.396

TABLE 4-13
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 5P-5933	1.21×10^{19}	1.053	4.37×10^{18}	0.770
Lower Shell Forging 5P-5951	1.21×10^{19}	1.053	4.37×10^{18}	0.770
Nozzle Shell Forging 123J218	3.63×10^{18}	0.720	1.31×10^{18}	0.473
Inter. to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	1.16×10^{19}	1.041	4.20×10^{18}	0.759
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF501 (Heat 442011)	3.63×10^{18}	0.720	1.31×10^{18}	0.473

Contained in Tables 4-14 through 4-17 are the calculations of the ART values used for the generation of the 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-14

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY (x 10^{19})	¼-t f (x 10^{19})	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	1.39	0.835	0.949	40	24.7	0	12.3	24.7	89
Intermediate Shell Forging → using S/C Data[e]				30.3	1.39	0.835	0.949	40	28.8	0	17.0	34.0	103
Lower shell Forging	5P-5951	0.04	0.64	26.0	1.39	0.835	0.949	10	24.7	0	12.3	24.7	59
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.34	0.805	0.939	-30	50.7	0	25.4	50.7	71
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.34	0.805	0.939	-30	61.9	0	14.0	28.0	60
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.415	0.249	0.623	30	19.3	0	9.7	19.3	69
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.415	0.249	0.623	10	25.5	0	12.8	25.5	61
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.415	0.249	0.623	10	16.0	0	8.0	16.0	42

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002). The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-15

Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY ($x 10^{19}$)	¾-t f ($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	1.39	0.301	0.671	40	17.4	0	8.7	17.4	75
Intermediate Shell Forging → using S/C Data[e]				30.3	1.39	0.301	0.671	40	20.3	0	17.0	34.0	94
Lower shell Forging	5P-5951	0.04	0.64	26.0	1.39	0.301	0.671	10	17.4	0	8.7	17.4	45
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.34	0.290	0.662	-30	35.7	0	17.9	35.7	41
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.34	0.290	0.662	-30	43.6	0	14.0	28.0	42
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.415	0.0899	0.396	30	12.3	0	6.1	12.3	55
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.415	0.0899	0.396	10	16.2	0	8.1	16.2	42
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.415	0.0899	0.396	10	10.2	0	5.1	10.2	30

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-16

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY ($x 10^{19}$)	¼-t f ($x 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	2.02	1.21	1.053	40	27.4	0	13.7	27.4	95
Intermediate Shell Forging → using S/C Data[e]				30.3	2.02	1.21	1.053	40	31.9	0	17.0	34.0	106
Lower shell Forging	5P-5951	0.04	0.64	26.0	2.02	1.21	1.053	10	27.4	0	13.7	27.4	65
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.94	1.16	1.041	-30	56.2	0	28.0	56.0	82
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.94	1.16	1.041	-30	68.6	0	14.0	28.0	67
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.604	0.363	0.720	30	22.3	0	11.2	22.3	75
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.604	0.363	0.720	10	29.5	0	14.8	29.5	69
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.604	0.363	0.720	10	18.5	0	9.3	18.5	47

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-17

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY ($x 10^{19}$)	¾-t f ($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	2.02	.437	.770	40	20.0	0	10.0	20.0	80
Intermediate Shell Forging → using S/C Data[e]				30.3	2.02	.437	.770	40	23.3	0	17.0	34.0	97
Lower shell Forging	5P-5951	0.04	0.64	26.0	2.02	.437	.770	10	20.0	0	10.0	20.0	50
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.94	.420	.759	-30	41.0	0	20.5	41.0	52
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.94	.420	.759	-30	50.0	0	14.0	28.0	48
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.604	0.131	0.473	30	14.7	0	7.3	14.7	59
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.604	0.131	0.473	10	19.4	0	9.7	19.4	49
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.604	0.131	0.473	10	12.2	0	6.1	12.2	34

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

The intermediate shell forging 5P-5933 is the limiting beltline material for all heatup and cooldown curves to be generated. The ART values associated with this material will be used in all sets of curves. Contained in Tables 4-18 and 4-19 is a summary of the limiting ARTs to be used in the generation of the Byron Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-18

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 22 EFPY

Material	**22 EFPY**	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging 5P-5933	89	75
- Using Surveillance Data[a]	**103[a]**	**94[a]**
Lower Shell Forging 5P-5951	59	45
Circumferential Weld WF-336	71	41
- Using Credible Surveillance Data	60	42
Circumferential Weld WF-501	61	42
- Using Credible Surveillance Data form Braidwood 1 and 2	42	30
Nozzle Shell Forging 123J218	69	55

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-1 and 5-2.

TABLE 4-19

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 32 EFPY

Material	32 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 5P-5933	95	80
- Using Surveillance Data	106[a]	97[a]
Lower Shell Forging 5P-5951	65	50
Circumferential Weld WF-336	82	52
- Using Credible Surveillance Data	67	48
Circumferential Weld WF-501	69	49
- Using Credible Surveillance Data form Braidwood 1 and 2	47	34
Nozzle Shell Forging 123J218	75	59

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-3 and 5-4.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-4 present the 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6]. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1 and 5-3 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Byron Unit 1 reactor vessel at 22 and 32 EFPY are 163°F, 166°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the Byron Unit 1 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-4 are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 103°F
3/4T, 94°F

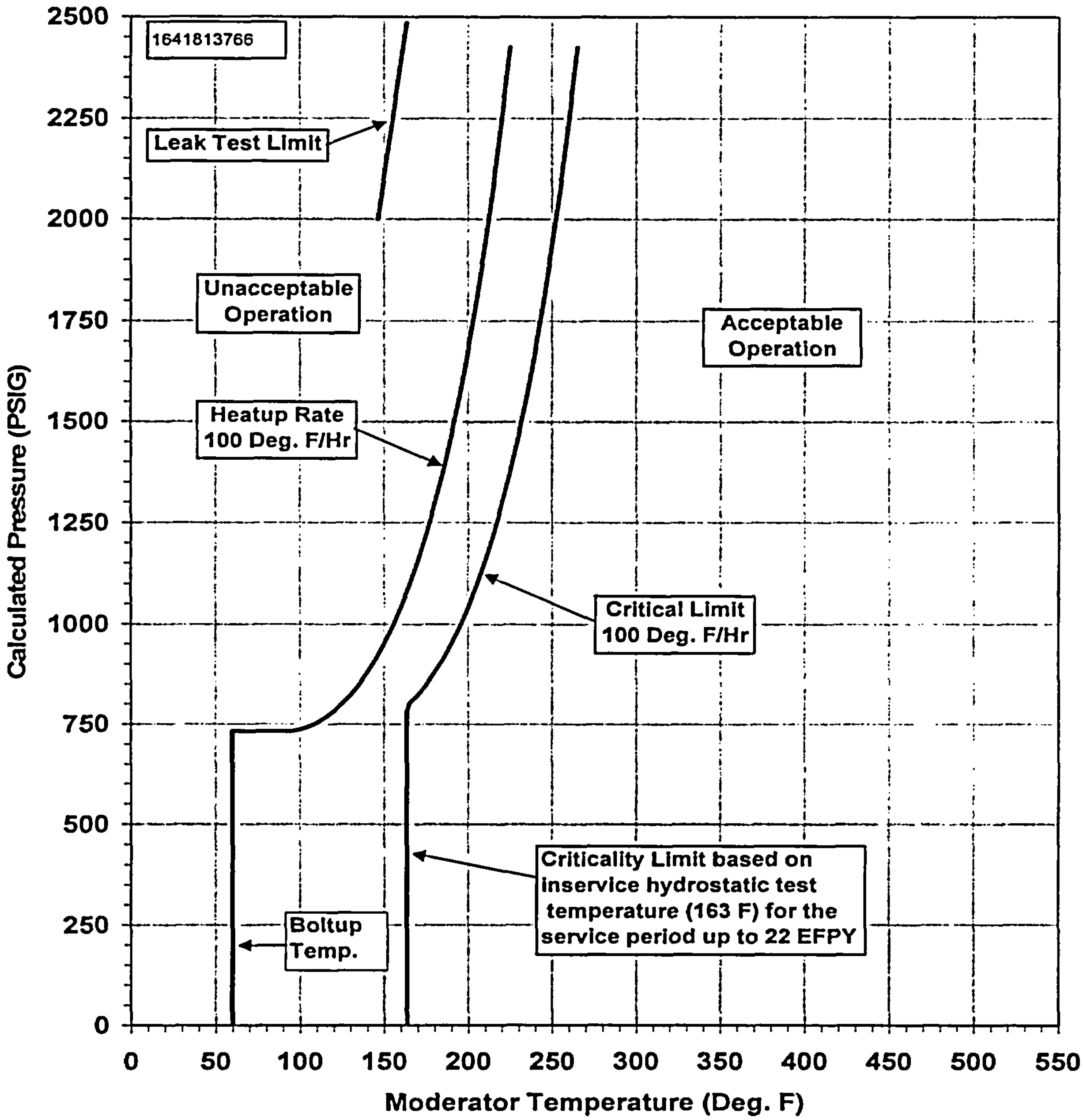

FIGURE 5-1 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY using 1996 Appendix G (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 103°F
3/4T, 94°F

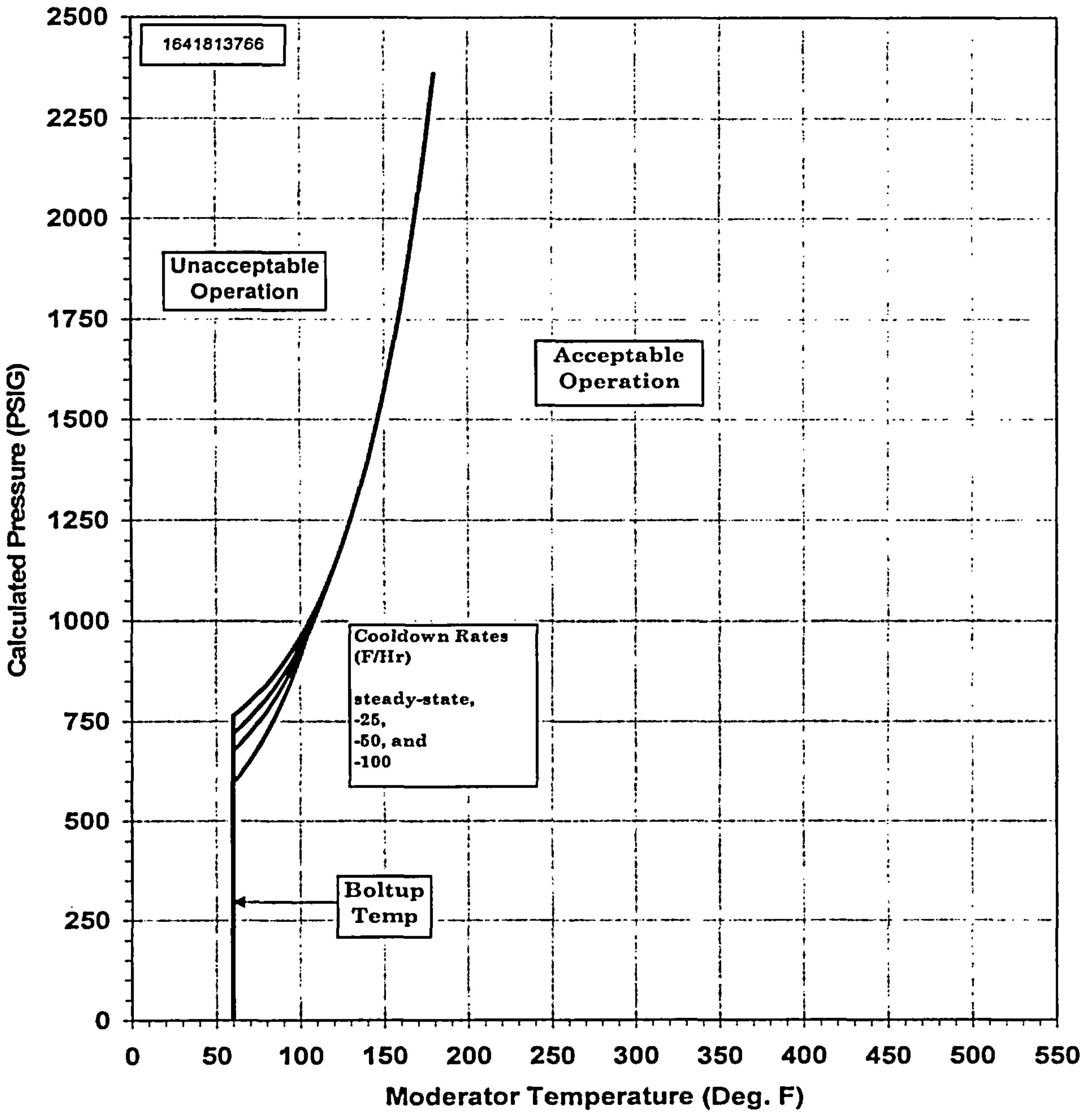

FIGURE 5-2 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY using.1996 Appendix G (Without Margins for Instrumentation Errors)

TABLE 5-1
Byron Unit 1 Heatup Data at 22 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	163	0	146	2000
60	732	163	732	163	2485
65	732	163	732		
70	732	163	732		
75	732	163	732		
80	732	163	732		
85	732	163	732		
90	732	163	733		
95	733	163	737		
100	737	163	743		
105	743	163	753		
110	753	163	766		
115	766	163	782		
120	782	165	801		
125	801	170	823		
130	823	175	849		
135	849	180	879		
140	879	185	913		
145	913	190	951		
150	951	195	994		
155	994	200	1042		
160	1042	205	1095		
165	1095	210	1155		
170	1155	215	1221		
175	1221	220	1294		
180	1294	225	1376		
185	1376	230	1466		
190	1466	235	1566		
195	1566	240	1676		
200	1676	245	1798		
205	1798	250	1932		
210	1932	255	2081		
215	2081	260	2245		
220	2245	265	2425		
225	2425				

TABLE 5-2
Byron Unit 1 Cooldown Data at 22 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	763	60	719	60	677	60	596
65	779	65	738	65	698	65	622
70	798	70	759	70	721	70	651
75	818	75	781	75	746	75	684
80	841	80	807	80	774	80	720
85	866	85	834	85	806	85	760
90	894	90	865	90	840	90	804
95	924	95	900	95	879	95	853
100	958	100	937	100	921	100	908
105	995	105	979	105	969		
110	1037	110	1026	110	1021		
115	1082	115	1077				
120	1133						
125	1188						
130	1250						
135	1318						
140	1393						
145	1476						
150	1568						
155	1669						
160	1781						
165	1905						
170	2042						
175	2194						
180	2361						

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F
3/4T, 97°F

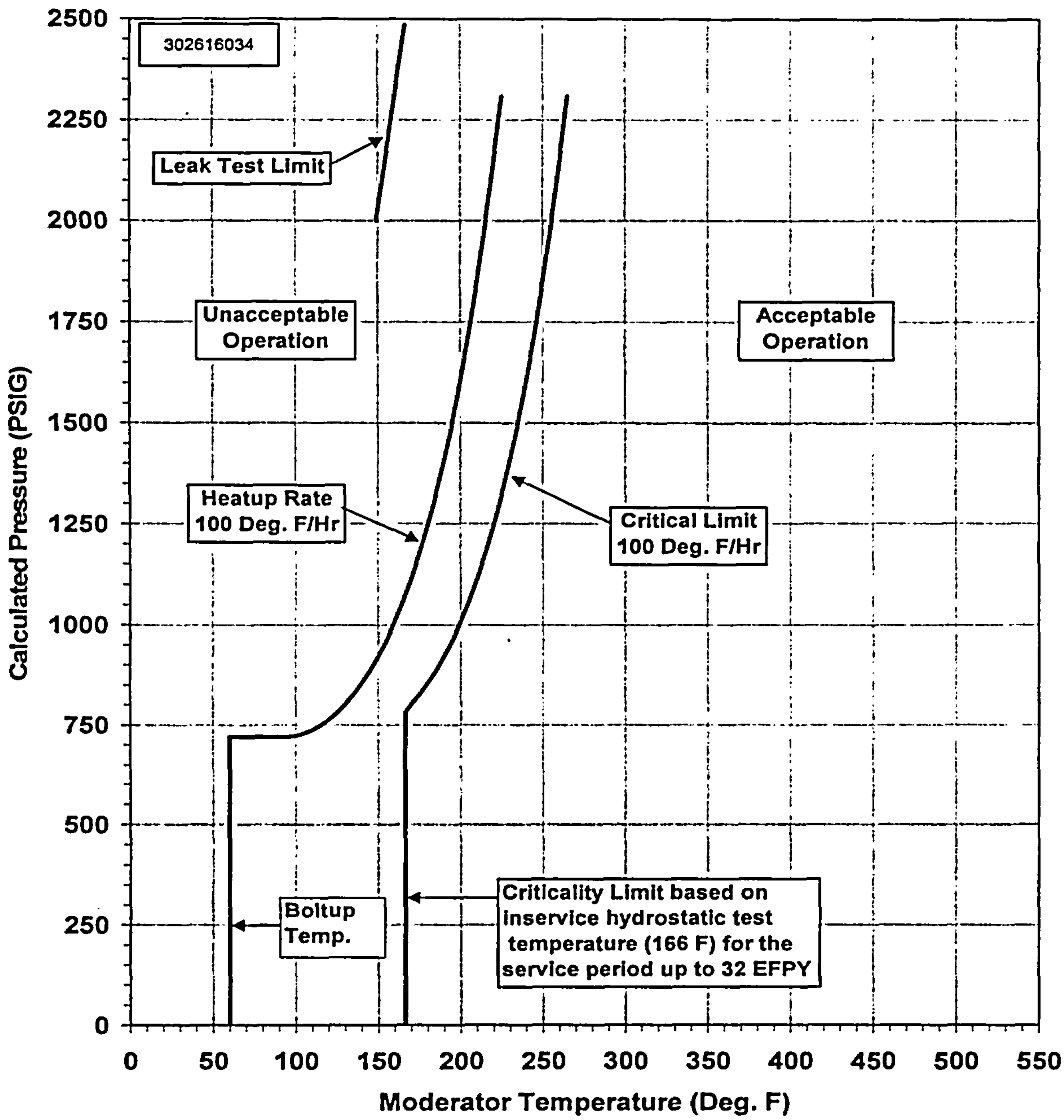

FIGURE 5-3 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F
3/4T, 97°F

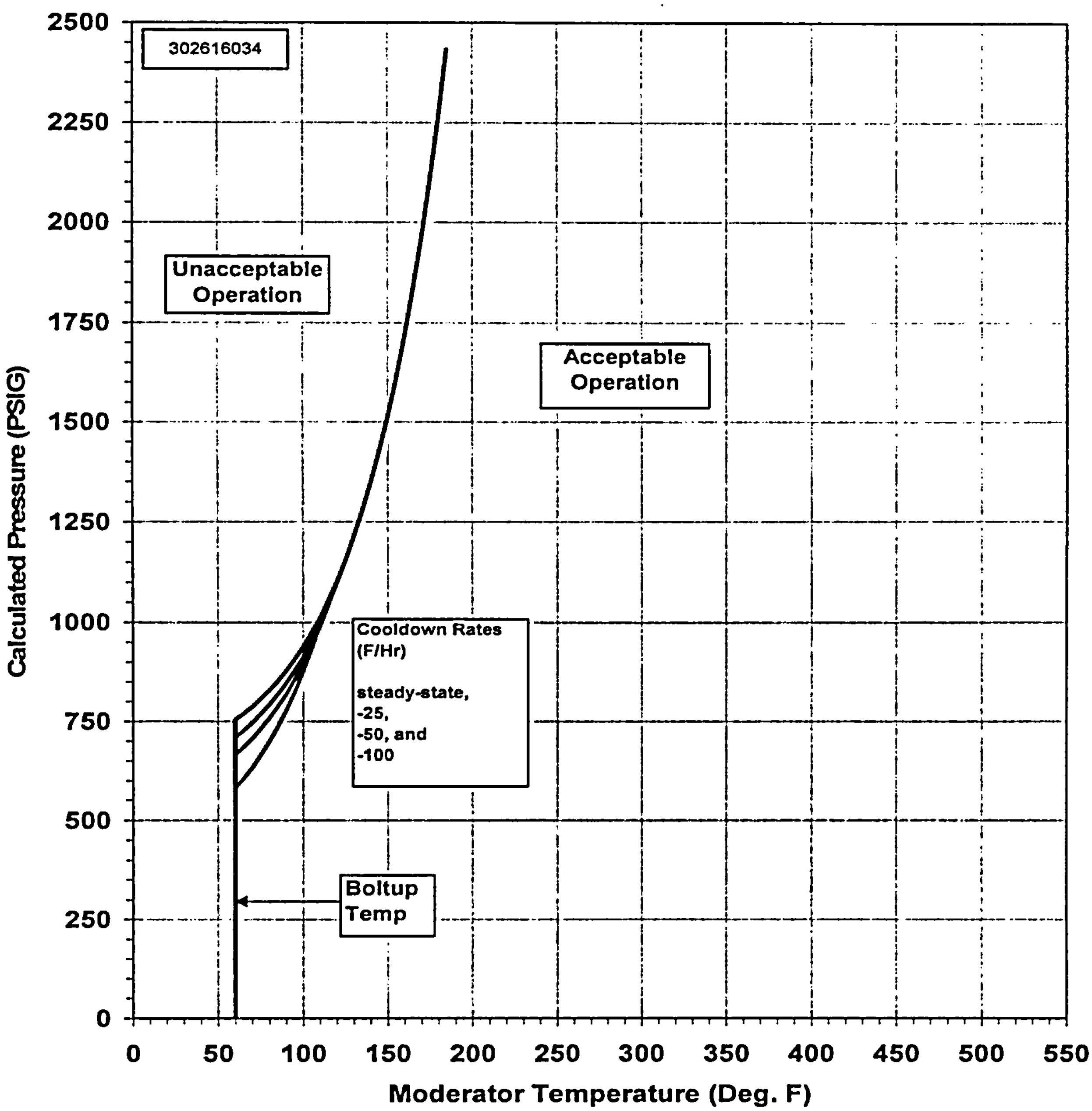

FIGURE 5-4 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G (Without Margins for Instrumentation Errors)

TABLE 5-3
Byron Unit 1 Heatup Data at 32 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	166	0	149	2000
60	720	166	720	166	2485
65	720	166	720		
70	720	166	720		
75	720	166	720		
80	720	166	720		
85	720	166	720		
90	720	166	720		
95	720	166	723		
100	723	166	729		
105	729	166	737		
110	737	166	749		
115	749	166	764		
120	764	166	781		
125	781	170	802		
130	802	175	826		
135	826	180	854		
140	854	185	886		
145	886	190	921		
150	921	195	962		
155	962	200	1007		
160	1007	205	1057		
165	1057	210	1113		
170	1113	215	1175		
175	1175	220	1244		
180	1244	225	1321		
185	1321	230	1406		
190	1406	235	1499		
195	1499	240	1603		
200	1603	245	1718		
205	1718	250	1844		
210	1844	255	1984		
215	1984	260	2138		
220	2138	265	2308		
225	2308				

TABLE 5-4
Byron Unit 1 Cooldown Data at 32 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	753	60	709	60	665	60	581
65	769	65	726	65	685	65	606
70	787	70	746	70	706	70	633
75	806	75	767	75	730	75	663
80	827	80	791	80	757	80	697
85	851	85	817	85	786	85	735
90	877	90	846	90	819	90	777
95	906	95	879	95	855	95	823
100	937	100	914	100	895	100	874
105	973	105	954	105	940	105	931
110	1011	110	997	110	989		
115	1054	115	1045	115	1043		
120	1102	120	1099				
125	1154						
130	1212						
135	1276						
140	1347						
145	1425						
150	1512						
155	1607						
160	1713						
165	1829						
170	1958						
175	2101						
180	2258						
185	2433						

6 REFERENCES

1 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.

2 ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

3 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

4 WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2", K.R. Hsu, November 2003.

5 ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

6 ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1995.

7 WCAP-15183, Revision 1, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, November 2003.

8 WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J. D. Andrachek, et al., January 1996.

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10 SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

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14 WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

15 WCAP-15373, Revision 1, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, October 2003.

16 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

17 WCAP-15368, "Commonwealth Edison Company Braidwood Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, March 2000.

18 WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., January 1999.

19 WCAP-15176, "Analysis of Capsule X from Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., March 1999.

20 WCAP-15316, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," Ed Terek, et. al., December 1999.

21 WCAP-15369, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., March 2000.

22 NDIT No. MSD-98-044, "Best Estimate Chemistry Values for Reactor Pressure Vessel Beltline Weld Heat Number 442002", Dated December 1998.

APPENDIX A

Thermal Stress Intensity Factors (K_{It})

TABLE A1

K_{It} Values for 100°F/hr Heatup Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
PT Curves are Limited by the ¾T Location for the Entire Curve				
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892
175	152.82	-13.5158	129.06	10.2056
180	157.63	-13.6533	133.64	10.3136
185	162.46	-13.7841	138.25	10.4146
190	167.29	-13.9046	142.88	10.5091
195	172.14	-14.0204	147.54	10.5982
200	176.99	-14.1279	152.22	10.6822
205	181.85	-14.2319	156.92	10.7619
210	186.72	-14.3292	161.63	10.8377
215	191.59	-14.4240	166.36	10.9102
220	196.47	-14.5135	171.09	10.9796
225	201.36	-14.6012	175.85	11.0465

Vessel Radius to the ¼T and ¾T Locations are as follows:

- 1/4T Radius = 88.750"
- 3/4T Radius = 93.000"

TABLE A2
K_{It} Values for 100°F/hr Cooldown Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
180	205.72	16.0887
175	200.64	16.0214
170	195.55	15.9546
165	190.47	15.8872
160	185.38	15.8204
155	180.30	15.7531
150	175.21	15.6862
145	170.13	15.6190
140	165.04	15.5523
135	159.95	15.4852
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: From T = 105°F to 180°F, the 100°F/hr Cooldown Rate is limited by the lower rates and/or the Steady State Condition.

TABLE A3
K_{It} Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
PT Curves are Limited by the ¾ T Location for the Entire Curve				
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892
175	152.82	-13.5158	129.06	10.2056
180	157.63	-13.6533	133.64	10.3136
185	162.46	-13.7841	138.25	10.4146
190	167.29	-13.9046	142.88	10.5091
195	172.14	-14.0204	147.54	10.5982
200	176.99	-14.1279	152.22	10.6822
205	181.85	-14.2319	156.92	10.7619
210	186.72	-14.3292	161.63	10.8377
215	191.59	-14.4240	166.36	10.9102
220	196.47	-14.5135	171.09	10.9796
225	201.36	-14.6012	175.85	11.0465

TABLE A4
K_{It} Values for 100°F/hr Cooldown Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
185	210.81	16.1555
180	205.72	16.0887
175	200.64	16.0214
170	195.55	15.9546
165	190.47	15.8872
160	185.38	15.8204
155	180.30	15.7531
150	175.21	15.6862
145	170.13	15.6190
140	165.04	15.5523
135	159.95	15.4852
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: From T = 110°F to 180°F, the 100°F/hr Cooldown Rate is limited by the lower rates and/or the Steady State Condition.

ATTACHMENT 4-D

WCAP-15392, Revision 2

Byron Unit 2
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Non-Proprietary Class 3

WCAP-15392
Revision 2

November 2003

Byron Unit 2
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-15392, Revision 2

Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

November 2003

Approved: J. A. Gresham, Manager
Reactor Component Design & Analysis

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Energy Systems
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Pittsburgh, PA 15230-0355

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: J. H. Ledger J. H. L.

Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15392 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15392 Rev. 0.

Note that only the heatup curves and associated data point tables in section 5 have changed. The cooldown curves and data points remain valid and were not changed.

Revision 2:

This revision was completed to incorporate an updated reference for the flange elimination from the PT limit curves. In the previous revision, WCAP-15315 was provided as justification for the flange notch elimination and it was changed to WCAP-16143-P in this revision. In addition to this change, the thermal stress intensity factors for the highest heatup and cooldown rates were added to this report in Appendix A.

EXECUTIVE SUMMARY

The purpose of this report is to generate new pressure-temperature limit curves for Byron Unit 2 for normal operation at 22 and 32 EFPY based of revised Uprated Fluences. The new pressure-temperature limit curves were generated using the methodology from WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-16143-P ("Flange Notch" Elimination Justification).. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values. The 1/4T and 3/4T values are summarized in Tables 4-18 and 19 and were calculated using the circumferential weld WF-447, Heat 442002 (The limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 4P-6107 (The limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-4.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Byron Station Unit No. 2 reactor pressure vessel (Specifically from forging 49D330-1/49C298-1). Also supplied was a weldment made with weld wire heat # 442002 Linde 80 flux, lot number 8064, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company (now known as Exelon Nuclear) contracted Westinghouse to generate new pressure-temperature limit curves for 22 and 32 EFPY based on the revised fluences from the 5% Uprating. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10CFR Part 50[3] per WCAP-16143-P, "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"[4],
- ASME Code Case N-588[5] (where applicable),
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G,[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, two sets of PT Curves will be generated. Set one will consist of the circumferential flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and the K_{Ic} methodology (ASME Code Case N-640) for the limiting circumferential weld material. Set two will consist of the 1996 Appendix G to ASME Section XI and the K_{Ic} methodology (ASME Code Case N-640) for the limiting forging/base metal material. Both sets of curves will use the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-16143-P). The final PT curves to be presented herein will be the most limiting set of curves. If the situation arises where portions of each set of curves are limiting, then composite curves will be generated that are based on the most limiting data (i.e. Circ. Flaw or Axial Flaw Case).

The purpose of this report is to present the calculations and the development of the Byron Unit 2 heatup and cooldown curves for 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Exelon, the surveillance weld data from the Byron Unit 1 and Unit 2 surveillance programs has been integrated. Note that Byron Unit 1 surveillance weld is identical to the surveillance weld (Heat No. 442002) at Byron Unit 2. In addition, the Braidwood Units 1 and 2 surveillance weld is identical to the nozzle to intermediate shell circumferential weld (Heat No. 442011). Per WCAP-15183[7], WCAP-15180[11] and WCAP-15368[17] all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress
K_{It} = stress intensity factor caused by the thermal gradients
K_{Ic} = function of temperature relative to the RT_{NDT} of the material
C = 2.0 for Level A and Level B service limits
C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \quad (3)$$

where, M_m for an inside surface flaw is given by:

$M_m \;=\; 1.85$ for $\sqrt{t} < 2$,
$M_m \;=\; 0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
$M_m \;=\; 3.21$ for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

$M_m \;=\; 1.77$ for $\sqrt{t} < 2$,
$M_m \;=\; 0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
$M_m \;=\; 3.09$ for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

$K_{Ib} = M_b$ * Maximum Stress, where M_b is two-thirds of M_m

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

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

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

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (4)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (5)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

<u>Code Case N-588; Circumferential Welds:</u>

In 1997, ASME Section XI, Appendix G was revised to add methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The earlier ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds.

In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of -2120 is

$$K_{Im} = M_m \times (pR_i / t)$$

where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Similarly, M_m for an outside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note again, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] addresses 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 preservice hydrostatic test pressure (3106 psi), which is 621 psig for Byron Unit 2 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, 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 pressure limitation of 20 percent of the hydrotest pressure were developed using the K_{IA} fracture toughness from the mid 1970's.

Improved knowledge of the 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 development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,"[4] concluded that the integrity of the closure head/vessel flange region is not a concern for Byron or Braidwood, therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta\ RT_{NDT} + Margin \qquad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta\ RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (9)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections for the 5% Uprating and the results are presented in SAE-REA-00-546[10]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-1 and 4-2, herein, contain the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Byron Unit 2 reactor vessel. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-3.

<u>Ratio Procedure and Temperature Adjustment:</u>

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT,\,measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\,measured} + 1.0*(T_{capsule} - T_{plant}) \qquad (10)$$

The irradiation temperatures from Byron Units 1 & 2 are presented in WCAP-14824, Revision 2[12]. The average irradiation temperature from each of the four Units and operating cycles in question is 553°F. Therefore, no temperature adjustment is required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-5 through 4-8. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 11, the surveillance weld data and the lower shell forging data for Byron Unit 2 is credible. In addition, per Reference 7, the surveillance weld data for Byron Unit 1 is credible. Position 2.1 chemistry factors are calculated in Tables 4-9 and 4-10. Reference17, the Braidwood 1 & 2 surveillance weld data is credible.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Byron Unit 2, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$.

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Byron Unit 2, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I.

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-1

Summary of the Peak Pressure Vessel Neutron Fluence Values at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	Surface[b] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Inter. Shell Forging 49D329-1/49C297-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Lower Shell Forging 49D330-1/49C298-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Nozzle Shell Forging 4P-6107	3.58×10^{18}	2.15×10^{18}	7.75×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	1.39×10^{19}	8.35×10^{18}	3.01×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	3.58×10^{18}	2.15×10^{18}	7.75×10^{17}

Notes:

(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.

(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material[a]	Surface[b] (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Inter. Shell Forging 49D329-1/49C297-1	2.06×10^{19}	1.24×10^{19}	4.46×10^{18}
Lower Shell Forging 49D330-1/49C298-1	2.06×10^{19}	1.24×10^{19}	4.46×10^{18}
Nozzle Shell Forging 4P-6107	5.22×10^{18}	3.13×10^{18}	1.13×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	2.03×10^{19}	1.22×10^{19}	4.40×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	5.22×10^{18}	3.13×10^{18}	1.13×10^{18}

Notes:

(a) All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV.

(b) Calculated Surface fluence documented in Reference 10.

TABLE 4-3
Calculated Integrated Neutron Exposure of the Byron Unit 2
Surveillance Capsules Tested to Date

Capsule	Fluence[a]
U	4.05×10^{18} n/cm^2, (E > 1.0 MeV)
W	1.27×10^{19} n/cm^2, (E > 1.0 meV)
X	2.30×10^{19} n/cm^2, (E > 1.0 MeV)

Notes:
(a) Documented in WCAP-15176[19].

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[16], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[a]
Intermediate Shell Forging 49D330-1/49C298-1 (Tangential Orientation)	U	-3.8°F
	W	3.65°F
	X	15.75°F
Intermediate Shell Forging 49D330-1/49C298-1 (Axial Orientation)	U	19.76°F
	W	31.88°F
	X	38.91°F
Surveillance Program Weld Metal (Heat # 442002)	U	8.44°F
	W	28.88°F
	X	54.01°F
Heat Affected Zone	U	6.74°F
	W	30.44°F
	X	34.22°F

Notes:

(a) Documented in WCAP-15176[19].

Table 4-5 contains the best estimate weight percent copper and nickel for the Byron Unit 2 base materials in the beltline region. Table 4-6 contains the calculation of the best estimate weight percent copper and nickel for the Byron Unit 2 surveillance weld material, while Table 4-7 presents the overall best estimate average for that heat of weld. Table 4-8 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-8 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-11. Tables 4-9 and 4-10 provide the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-11.

TABLE 4-5

Best Estimate Cu and Ni Weight Percent for the Byron Unit 2 Forging Materials

Reference	Intermediate Shell Forging 49D329-1/49C297-1		Lower Shell Forging 49D330-1/49C298-1	
	Cu %	Ni %	Cu %	Ni %
Ref. 22	0.01	0.70	0.05[a]	0.71[a]
Charpy YL-48[b]	---	---	0.075	0.78
Charpy YT-52[b]	---	---	0.073	0.78
Best Estimate Average[c]	0.01	0.70	0.06	0.73

Notes:

(a) This is the average of 4 data points.

(b) Charpy Specimens from Capsule X of Byron Unit 2 (Ref. 19).

(c) The best estimate average was rounded per ASTM E29, using the "Rounding Method". This calculation is also presented in Reference 13.

TABLE 4-6

Average Cu and Ni Weight Percent for the Byron Unit 2 Surveillance Weld Material Only (Heat # 442002)

Reference	Weight % Copper	Weight % Nickel
Ref. 12	0.023[a]	0.712[a]
Charpy YW-51[b]	0.031	0.70
Charpy YW-52[b]	0.033	0.70
Surveillance Weld Average[c]	0.02	0.71

Notes:

(a) This is the average of 31 data points.

(b) Charpy Specimens from Capsule X of Byron Unit 2 (Ref. 19).

(c) The best estimate average was rounded per ASTM E29, using the "Rounding Method". This calculation is also presented in Reference 13.

TABLE 4-7

Best Estimate Cu and Ni Weight Percent Values for the Byron Units 1 & 2 Weld Material (Using Byron 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W WQ: BAW-2261	Ref. 12	0.024	0.70
B&W WQ: BAW-2261	Ref. 12	0.031	0.46
B&W WQ: BAW-2261	Ref. 12	0.03	0.72
B&W WQ: BAW-2261	Ref. 12	0.068	0.48
B&W WQ: BAW-2261	Ref. 12	0.053	0.62
B&W WQ: BAW-2261	Ref. 12	0.059	0.62
B&W WQ (from NDIT No. BYR97-346, Rev. 0)	Ref. 12	0.029	0.65
Round Robin Data Ave. on Weld WF-336 (from NDIT No. BRW-DIT-97-391, Rev. 0)	Ref. 12	0.038	0.658
Byron 1 Surveillance Data Ave. [(a)]	Ref. 12	0.022	0.69
Byron 2 Surveillance Ave. [(b)]	Table 4-6	0.023	0.712
BEST ESTIMATE AVERAGE		0.04[(c, d)]	0.63[(c, d)]

Notes:

(a) The weld material in the Byron Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld. (Weld seam WF-336, Wire Heat # 442002, Flux Type Linde 80, Flux Lot # 8873).

(b) The Byron Unit 2 surveillance weld is identical to that used in the reactor vessel core region girth seam (WF-447). The weld wire is type Linde MnMoNi (Low Cu-P), heat number 442002, with a Linde 80 type flux, lot number 8064.

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method". This calc as presented in Reference 13.

(d) This ave. Cu & Ni excluded two data points (Cu = 0.114, Ni = 0.54 and Cu = 0.148, Ni = 0.60), see Ref. 23.

TABLE 4-8[b]

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 5P7382 / 3P6407	---	0.71	0
Vessel Flange 124L556VA1	---	0.70	30
Nozzle Shell Forging 4P-6107[b]	0.05	0.74	10
Intermediate Shell Forging 49D329-1/49C297-1	0.01	0.70	-20
Lower Shell Forging 49D330-1/49C298-1	0.06	0.73	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat # 442002)	0.04	0.63	10
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat # 442011)	0.03	0.67	40
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 4-7 of Reference 13.

TABLE 4-9

Calculation of Chemistry Factors using Byron Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	ΔRT_{NDT} $^{(c)}$	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging	U	0.405	0.749	0.0[e]	0	0.561
49D330-1/49C298-1	W	1.27	1.067	3.65	3.89	1.138
(Tangential)	X	2.30	1.225	15.75	19.29	1.500
Lower Shell Forging	U	0.405	0.749	19.76	14.80	0.561
49D330-1/49C298-1	W	1.27	1.067	31.88	34.02	1.138
(Axial)	X	2.30	1.225	38.91	47.66	1.500
				SUM:	119.66	6.398
	$CF_{Forging} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (119.66) \div (6.398) = 18.7°F$					
Byron Unit 1 Surv. Weld Material	U	0.404	0.749	11.22 (5.61)[d]	8.40	0.561
(Heat # 442002)	X	1.57	1.125	80.22 (40.11)[d]	90.25	1.266
	W	2.43	1.239	102.68 (51.34)[d]	127.22	1.535
Byron Unit 2 Surv. Weld Material	U	0.405	0.749	16.88 (8.44)[d]	12.64	0.561
(Heat # 442002)	W	1.27	1.067	57.76 (28.88)[d]	61.63	1.138
	X	2.30	1.225	108.02 (54.01)[d]	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv. Weld, 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) = 65.9°F$					

Notes:

(a) Calculated Capsule Fluence taken from References 18 and 19 (or Table 4-3 herein) for Byron Units 1 and 2, respectively.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (Byron Unit 1 from Ref. 18 & Byron Unit 2 from Ref. 19 or Table 4-4 herein).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustment are required.

(e) Actual shift was a -3.81. Since this physically should not occur, a conservative value of 0.0 will be used.

TABLE 4-10

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}{}^{(c)}$	$FF^*\Delta RT_{NDT}$	FF^2
Braidwood Unit 1	U	0.387	0.737	17.06[(d)]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[(d)]	31.96	1.124
Weld Heat 442011, WF-501	W	2.09	1.201	49.68[(d)]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[(d, e)]	0	0.557
Surveillance	X	1.23	1.058	26.3[(d)]	27.83	1.119
Weld Heat 442011, WF-501	W	2.25	1.220	23.9[(d)]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

Notes:

(a) Calculated Capsule Fluence values taken from References 20 and 21 for Braidwood Units 1 and 2, respectively.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (Braidwood Unit 1 from Ref. 20 & Braidwood Unit 2 from Ref. 21).

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

(e) Actual shift was a -0.58. Since this physically should not occur, a conservative value of 0.0 will be used.

TABLE 4-11
Summary of the Byron Unit 2 Reactor Vessel Beltline Material Chemistry Factors
Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Inter. Shell Forging 49D329-1/49C297-1	20.0°F	---
Lower Shell Forging 49D330-1/49C298-1	37.0°F	18.7°F
Nozzle Shell Forging 4P-6107	31.0°F	---
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	54.0°F	65.9°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Byron Unit 1 Surveillance Program Weld Metal	27.0°F	---
Braidwood Unit 1 & 2 Surveillance Weld Metal	41.0°F	

Contained in Tables 4-12 and 4-13 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Byron Unit 2 reactor vessel beltline materials for 22 and 32 EFPY.

TABLE 4-12
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate. Shell Forging 49D329-1/49C297-1	8.47 x 10^{18}	0.953	3.05 x 10^{18}	0.675
Lower Shell Forging 49D330-1/49C298-1	8.47 x 10^{18}	0.953	3.05 x 10^{18}	0.675
Nozzle Shell Forging 4P-6107	2.15 x 10^{18}	0.587	7.75 x 10^{17}	0.368
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	8.35 x 10^{18}	0.949	3.01 x 10^{18}	0.671
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	2.15 x 10^{18}	0.587	7.75 x 10^{17}	0.368

TABLE 4-13
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate. Shell Forging 49D329-1/49C297-1	1.24 x 10^{19}	1.06	4.46 x 10^{18}	0.775
Lower Shell Forging 49D330-1/49C298-1	1.24 x 10^{19}	1.06	4.46 x 10^{18}	0.775
Nozzle Shell Forging 4P-6107	3.13 x 10^{18}	0.681	1.13 x 10^{18}	0.442
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	1.22 x 10^{19}	1.05	4.40 x 10^{18}	0.772
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	3.13 x 10^{18}	0.681	1.13 x 10^{18}	0.442

Contained in Tables 4-14 through 4-17 are the calculations of the ART values used for the generation of the 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-14

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY ($x 10^{19}$)	¼-t f($x 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	1.41	0.847	0.953	-20	19.1	0	9.5	19.1	18
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	1.41	0.847	0.953	-20	35.3	0	17.0	34.0	49
Lower shell Forging → using S/C Data				18.7	1.41	0.847	0.953	-20	17.8	0	8.5	17.0	15
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	1.39	0.835	0.949	10	51.2	0	25.6	51.2	112
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.39	0.835	0.949	10	62.5	0	14.0	28.0	101
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.358	0.215	0.587	10	18.2	0	9.1	18.2	46
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.358	0.215	0.587	40	24.1	0	12.0	24.1	88
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.358	0.215	0.587	40	15.1	0	7.5	15.1	70

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm², E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-15

Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 22[(a)] EFPY ($x 10^{19}$)	¾-t f($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	1.41	0.305	0.675	-20	13.5	0	6.7	13.5	7
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	1.41	0.305	0.675	-20	25.0	0	12.5	25.0	30
Lower shell Forging → using S/C Data				18.7	1.41	0.305	0.675	-20	12.6	0	6.3	12.6	5
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	1.39	0.301	0.671	10	36.2	0	18.1	36.2	82
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.39	0.301	0.671	10	44.2	0	14.0	28.0	82
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.358	0.0775	0.368	10	11.4	0	5.7	11.4	33
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.358	0.0775	0.368	40	15.1	0	7.5	15.1	70
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.358	0.0775	0.368	40	9.5	0	4.7	9.5	59

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-16

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY (x 10^{19})	¼-t f(x 10^{19})	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	2.06	1.24	1.060	-20	21.2	0	10.6	21.2	22
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	2.06	1.24	1.060	-20	39.2	0	17.0	34.0	53
Lower shell Forging → using S/C Data				18.7	2.06	1.24	1.060	-20	19.8	0	8.5	17.0	17
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	2.03	1.22	1.05	10	56.7	0	28.0	56.0	123
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	2.03	1.22	1.05	10	69.2	0	14.0	28.0	107
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.522	0.313	0.681	10	21.1	0	10.6	21.1	52
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.522	0.313	0.681	40	27.9	0	14.0	27.9	96
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.522	0.313	0.681	40	17.5	0	8.8	17.5	75

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-17

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 32[(a)] EFPY ($x\ 10^{19}$)	¾-t f($x\ 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	2.06	.446	.775	-20	15.5	0	7.8	15.5	11
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	2.06	.446	.775	-20	28.7	0	14.3	28.7	37
Lower shell Forging → using S/C Data				18.7	2.06	.446	.775	-20	14.5	0	7.2	14.5	9
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	2.03	.440	.772	10	41.7	0	20.8	41.7	93
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	2.03	.440	.772	10	50.9	0	14.0	28.0	89
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.522	0.113	0.442	10	13.7	0	6.9	13.7	37
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.522	0.113	0.442	40	18.1	0	9.1	18.1	76
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.522	0.113	0.442	40	11.4	0	5.7	11.4	63

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

The girth weld WF-447 and the nozzle shell forging 4P-6107 are the limiting beltline materials for all heatup and cooldown curves to be generated. The ART value associated with these materials will be used in all three sets of curves. The girth weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw), otherwise, the nozzle shell forging ART will be used. The ART associated with the limiting axial material is considered to determine if this case would be more conservative or overlap the circ. flaw curves. Contained in Tables 4-18 and 4-19 is a summary of the limiting ARTs to be used in the generation of the Byron Unit 2 reactor vessel heatup and cooldown curves.

TABLE 4-18

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 22 EFPY

Material	22 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging [49D329/49C297]-1-1	18	7
Lower Shell Forging [49D330/49C298]-1-1	49	30
- Using Surveillance Data[(a)]	15	5
Circumferential Weld WF-447	112	82
- Using Surveillance Data	101	82
Circumferential Weld WF-562	88	70
- Using Surveillance Data from Braidwood 1 and 2	70	59
Nozzle Shell Forging 4P-6107	46[(a)]	33[(a)]

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-1 and 5-2. They were generated using the '96 App. G Methodology, and were more conservative than the curves generated using the limiting circumferential flaw ART values and Code Case N-588 Methodology.

TABLE 4-19
Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 32 EFPY

Material	32 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging [49D329/49C297]-1-1	22	11
Lower Shell Forging [49D330/49C298]-1-1	53	37
- Using Surveillance Data[a]	17	9
Circumferential Weld WF-447	123	93
- Using Surveillance Data	107	89
Circumferential Weld WF-562	96	76
- Using Surveillance Data from Braidwood 1 and 2	75	63
Nozzle Shell Forging 4P-6107	52[a]	37[a]

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-3 and 5-4. They were generated using the '96 App. G Methodology, and were more conservative than the curves generated using the limiting circumferential flaw ART values and Code Case N-588 Methodology.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-4 present the 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6] and Code Case N-588[5], respectively. The heatup and cooldown curves that are presented herein are actually are curves generated using the 1996 App. G methodology with the *lower axial flaw ART value* with exception to the last two temperatures of the 22 EFPY 100°F/hr. heatup curve and the last four temperatures of the 32 EFPY 100°F/hr heatup curve. The reason is, these curves are more conservative (with exception noted) than the curves generated using Code Case N-588 methodology with the *higher circ. flaw ART value.* This is true throughout the entire temperature range, including criticality.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1and 5-3 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\ e^{[0.02\,(T - RT_{NDT})]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Byron Unit 2 reactor vessel at 22 and 32 EFPY are 106°F and 112°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of non-ductile failure for the Byron Unit 2 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-4 are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 22 EFPY: 1/4T, 101°F (N-588) & 46°F ('96 App. G)
3/4T, 82°F (N-588) & 33°F ('96 App. G)

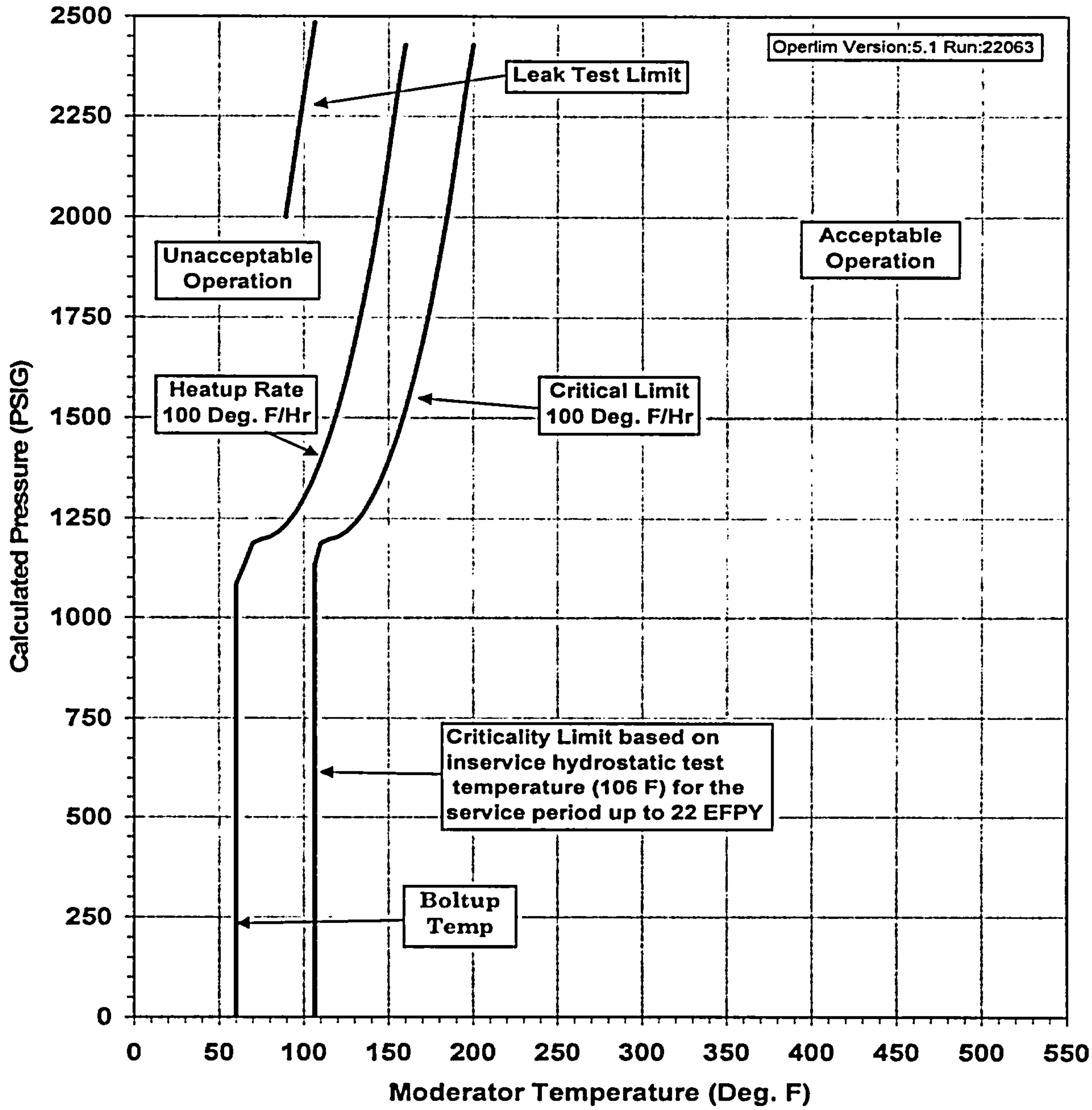

FIGURE 5-1 Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 101°F (N-588) & 46°F ('96 App. G)
3/4T, 82°F (N-588) & 33°F ('96 App. G)

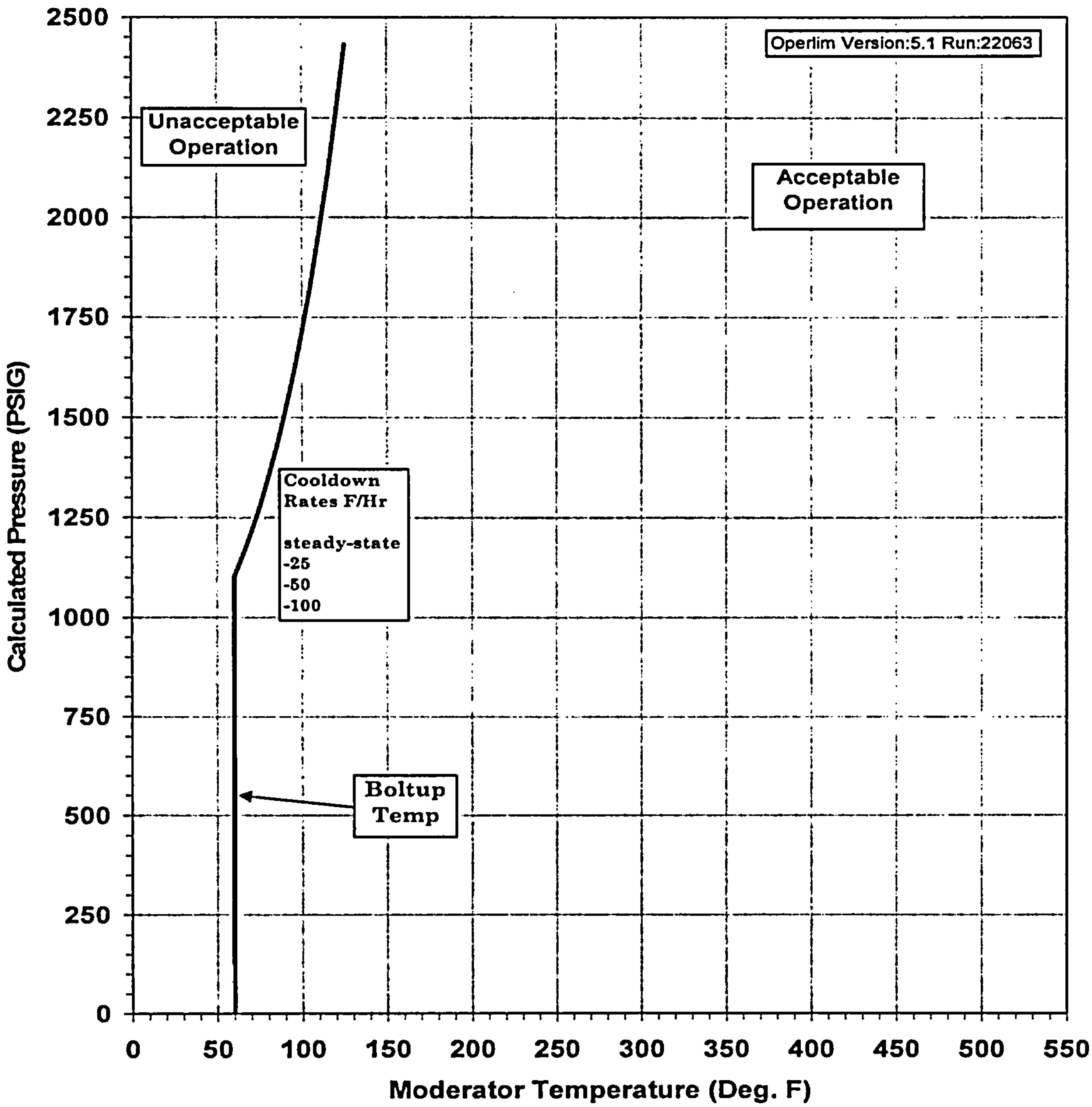

FIGURE 5-2 Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins for Instrumentation Errors)

TABLE 5-1

Byron Unit 2 Heatup Data at 22 EFPY Using Code Case N-640 & 1996 Appendix G Methodology*

(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	106	0	89	2000
60	1082	106	1133	106	2485
65	1133	110	1186		
70	1186	115	1197		
75	1197	120	1202		
80	1202	125	1215		
85	1215	130	1235		
90	1235	135	1263		
95	1263	140	1299		
100	1299	145	1342		
105	1342	150	1393		
110	1393	155	1452		
115	1452	160	1520		
120	1520	165	1598		
125	1598	170	1686		
130	1686	175	1784		
135	1784	180	1895		
140	1895	185	2018		
145	2018	190	2155		
150	2155	195	2304		
155	2304	200	2428		
160	2428				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall with exception to the last two temperatures, which are limited by the Circ-Flaw ART with Code Case N-588.

TABLE 5-2

Byron Unit 2 Cooldown Data at 22 EFPY Using Code Case N-640 & 1996 App. G Methodology*

(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1102						
65	1154						
70	1212						
75	1276						
80	1347						
85	1425						
90	1512						
95	1607						
100	1713						
105	1829						
110	1958						
115	2101						
120	2258						
125	2433						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

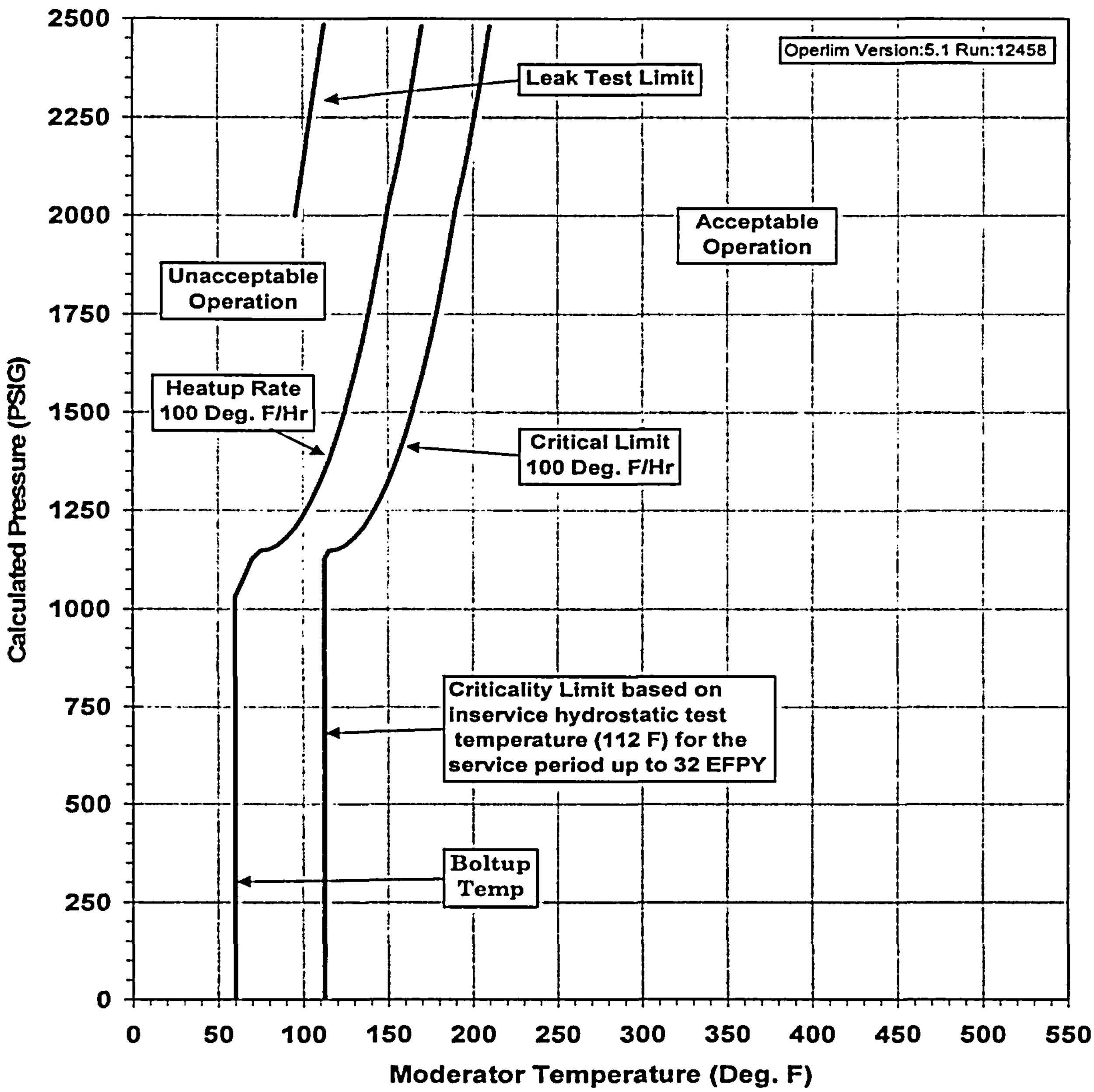

FIGURE 5-3 Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

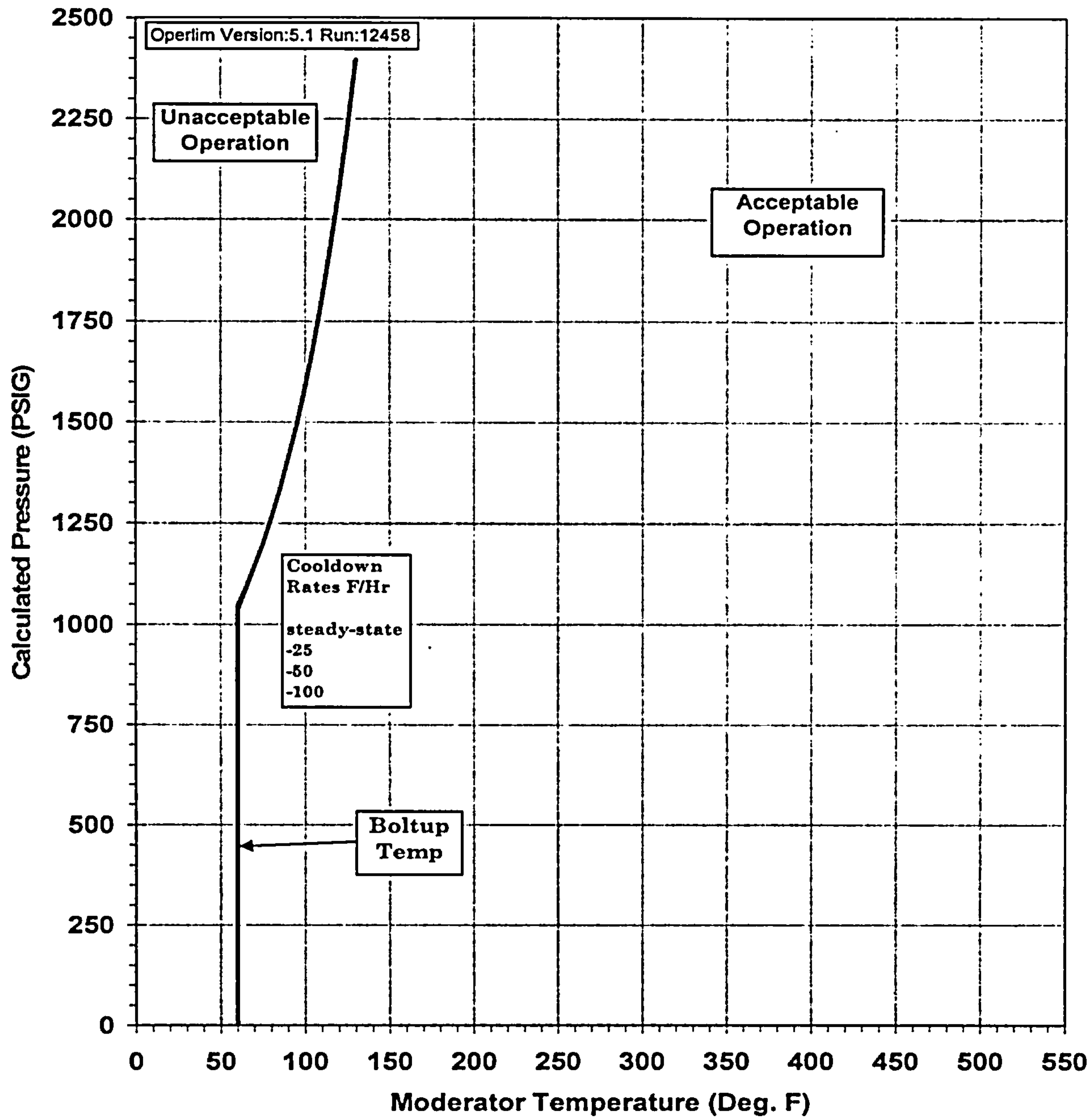

FIGURE 5-4 Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins for Instrumentation Errors)

TABLE 5-3
Byron Unit 2 Heatup Data at 32 EFPY Using Code Case N-640 & 1996 App. G Methodology*
(Without Margins for Instrumentation Errors)

100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	1030	112	0	95	2000
65	1078	112	1078	112	2485
70	1128	112	1128		
75	1148	115	1148		
80	1152	120	1152		
85	1162	125	1162		
90	1180	130	1180		
95	1205	135	1205		
100	1237	140	1237		
105	1276	145	1276		
110	1323	150	1323		
115	1377	155	1377		
120	1440	160	1440		
125	1511	165	1511		
130	1592	170	1592		
135	1682	175	1682		
140	1784	180	1784		
145	1897	185	1897		
150	2023	190	2023		
155	2120	195	2120		
160	2227	200	2227		
165	2347	205	2347		
170	2480	210	2480		

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall with exception to the last four temperatures, which are limited by the Circ-Flaw ART with Code Case N-588.

TABLE 5-4
Byron Unit 2 Cooldown Data at 32 EFPY Using Code Case N-640 & 1996 App. G Methodology*
(Without Margins for Instrumentation Errors)

Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1045	60	1036	60	1033		
65	1092	65	1088				
70	1143						
75	1200						
80	1263						
85	1332						
90	1409						
95	1494						
100	1587						
105	1691						
110	1805						
115	1932						
120	2071						
125	2226						
130	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.

6 REFERENCES

1 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.

2 ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

3 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

4 WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2", K.R. Hsu, November 2003.

5 ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

6 ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1995.

7 WCAP-15183, Revision 1, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, November 2003.

8 WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J. D. Andrachek, et al., January 1996.

9 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

10 SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

11 WCAP-15180, Revision 1, "Commonwealth Edison Company Byron Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, November 2003.

12 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration For Byron and Braidwood", T. J. Laubham, et al., November 1997. Ref. Errata letter CAE-97-233, CCE-97-316, "Transmittal of Updated Tables to WCAP-14824 Rev. 2"

13 WCAP-15178, "Byron Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, June 1999.

14 WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

15 WCAP-15373, Revision 2, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 2003.

16 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

17 WCAP-15368, "Commonwealth Edison Company Braidwood Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, March 2000.

18 WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., January 1999.

19 WCAP-15176, "Analysis of Capsule X from Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., March 1999.

20 WCAP-15316, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," Ed Terek, et. al., December 1999.

21 WCAP-15369, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., March 2000.

22 WCAP-14940, "Byron Unit 2 Heatup and Cooldown Limit Curves For Normal Operation", T. J. Laubham, October 1997. Ref. Errata letters CAE-97-210 & 232, CCE-97-289 & 316, "Transmittal of Updated Tables to WCAP-14940 and WCAP-14970"

23 NDIT No. MSD-98-044, "Best Estimate Chemistry Values for Reactor Pressure Vessel Beltline Weld Heat Number 442002", Dated December 1998.

APPENDIX A

Thermal Stress Intensity Factors (K_{It})

TABLE A1

K_{It} Values for 100°F/hr Heatup Curve (22 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
PT Curves are Limited by the ¾T Location for the Entire Curve				
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277

Vessel Radius to the ¼T and ¾T Locations are as follows:

- 1/4T Radius = 88.750"
- 3/4T Radius = 93.000"

TABLE A2

K_{It} Values for 100°F/hr Cooldown Curve (22 EFPY)

Water Temp. (°F)	**Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown** (°F)	**100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor** (KSI SQ. RT. IN.)
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: From T = 60°F to 125°F, the 100°F/hr Cooldown Rate is limited by the Steady State Condition.

TABLE A3
K_{It} Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
PT Curves are Limited by the ¾ T Location for the Entire Curve				
60	56.01	-0.9945	55.05	0.4767
65	58.62	-2.4413	55.31	1.4422
70	61.70	-3.6836	56.01	2.4209
75	65.01	-4.8600	57.20	3.3365
80	68.58	-5.8718	58.80	4.1525
85	72.27	-6.7937	60.80	4.8806
90	76.14	-7.5912	63.14	5.5230
95	80.11	-8.3181	65.77	6.0965
100	84.21	-8.9517	68.68	6.6042
105	88.40	-9.5267	71.81	7.0564
110	92.68	-10.0310	75.15	7.4589
115	97.04	-10.4906	78.67	7.8204
120	101.46	-10.8960	82.35	8.1440
125	105.95	-11.2667	86.16	8.4352
130	110.48	-11.5953	90.10	8.6969
135	115.06	-11.8972	94.14	8.9336
140	119.68	-12.1661	98.28	9.1474
145	124.34	-12.4147	102.49	9.3417
150	129.03	-12.6374	106.78	9.5183
155	133.75	-12.8446	111.14	9.6799
160	138.48	-13.0315	115.55	9.8277
165	143.25	-13.2066	120.01	9.9638
170	148.02	-13.3657	124.52	10.0892

TABLE A4
K_{It} Values for 100°F/hr Cooldown Curve (32 EFPY)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
130	154.87	15.4187
125	149.78	15.3518
120	144.70	15.2854
115	139.61	15.2187
110	134.53	15.1525
105	129.45	15.0860
100	124.36	15.0200
95	119.28	14.9538
90	114.19	14.8881
85	109.11	14.8221
80	104.02	14.7566
75	98.94	14.6909
70	93.86	14.6257
65	88.77	14.5603
60	83.69	14.4946

NOTE: From T = 60°F to 130°F, the 100°F/hr Cooldown Rate is limited by the lower rates and/or the Steady State Condition.

ATTACHMENT 5

Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points

ATTACHMENT 5

Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points

The WCAP reports [1] provide the necessary information to validate the proper application of previously approved Cases of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Nuclear Code Cases N-640 [2] and N-588 [3]. However, as discussed in Attachment 1, these reports also apply the exemption from the flange temperature requirements of 10 CFR 50 Appendix G. This regulation states that the metal temperature at the closure flange regions must exceed the material unirradiated initial 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. For the Braidwood Station and Byron Station units, this 20 percent pressure value is 621 psig.

To facilitate review, a comparison of the current Braidwood and Byron Station heat up, cooldown, steady state, and criticality limit data points to those of the 22 effective full power year (EFPY) curves with the flange temperature requirement of Appendix G restored and Code Cases N-640 and N-588 applied is provided. The purpose of this comparison is to demonstrate that the current Braidwood and Byron Station curves are more restrictive than the 22 EFPY limits using the approved methodology of ASME Code Cases N-640 and N-588. Consequently, it is justifiable to extend the applicability time of the current Braidwood and Byron Station P-T curves an additional two EFPY for each unit as provided in Attachment 6.

To clarify, EGC does not propose to adopt P-T curves based on Code Cases N-640 or N-588 or modify the data points in any way, but rather EGC intends to extend the current P-T curves by an additional two EFPY. The current P-T curves based on previously approved methodology, i.e., 1996 ASME Section XI, Appendix G methodology and ASME Code Case N-514, are more conservative and operationally more restrictive. The low temperature overpressure protection (LTOP) system setpoints, the LTOP enable temperature, the reactor vessel boltup temperature, and the Reactor Vessel Minimum Pressurization Temperature are unchanged from the current Braidwood and Byron Stations PTLRs. Since the Braidwood Station and Byron Station P-T limit curves use K_{Ia} to determine allowable pressure in the vessel, the LTOP system will continue to limit the maximum vessel pressure to 110% of the allowable pressure.

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[1] WCAP-15365, Revision 2, Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, WCAP-15373, Revision 2, Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation, WCAP-15391, Revision 1, Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, and WCAP-15392, Revision 1, Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation.

[2] Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"

[3] Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1"

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Braidwood Unit 1

100 Degree Heat Up			Criticality Limit				Leak Test Limit			
Temperature	22 EFPY Pressure	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressure	Current PTLR Temp	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressures	Current PTLR Temps	Current PTLR Pressures
60	0	0	103	0	210	0	86	2000	188	2000
60	565.09	565.09	103	1110	210	611.83	103	2485	210	2485
65	565.09	565.09	105	1163	210	597.56				
70	565.09	565.09	110	1218	210	585.60				
75	565.09	565.09	115	1222	210	576.77				
80	565.09	565.09	120	1229	210	570.35				
85	565.09	565.09	125	1243	210	566.61				
90	565.09	565.09	130	1264	210	565.09				
95	565.09	565.09	135	1294	210	565.87				
100	565.87	565.87	140	1331	210	568.69				
105	568.69	568.69	145	1376	210	573.56				
110	573.56	573.56	150	1430	210	580.30				
115	1492	580.30	155	1492	210	588.84				
120	1563	588.84	160	1563	210	599.36				
125	1644	599.36	165	1644	210	611.78				
130	1736	611.78	170	1736	210	626.07				
135	1838	626.07	175	1838	210	642.16				
140	1954	642.16	180	1954	210	660.36				
145	2082	660.36	185	2082	210	680.59				
150	2225	680.59	190	2225	210	702.80				
155	2384	702.80	195	2384	210	727.33				
160		727.33			210	754.07				
165		754.07			210	783.17				
170		783.17			215	814.98				
175		814.98			220	849.37				
180		849.37			225	886.54				
185		886.54			230	926.73				
190		926.73			235	970.11				
195		970.11			240	1016.91				
200		1016.91			245	1067.33				
205		1067.33			250	1121.63				
210		1121.63			255	1180.01				
215		1180.01			260	1242.62				
220		1242.62			265	1309.84				
225		1309.84			270	1382.03				
230		1382.03			275	1459.45				
235		1459.45			280	1542.27				
240		1542.27			285	1630.97				
245		1630.97			290	1726.05				
250		1726.05			295	1827.80				
255		1827.80			300	1936.51				
260		1936.51			305	2052.39				
265		2052.39			310	2176.33				
270		2176.33			315	2308.42				
275		2308.42			320	2449.09				
280		2449.09								

ATTACHMENT 5
Comparison of 22 EFPY Data Points and
Current Byron and Braidwood Station PTLR Data Points
Braidwood Unit 1

Steady State			25 Degree F Cooldown			50 Degree F Cooldown			100 Degree F Cooldown		
Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure
60	0	0.00	60	0	0	60	0	0	60	0	0
60	620	620.27	60	577	577.45	60	534	534.28	60	447	446.98
65	621	621.00	65	591	590.68	65	549	548.52	65	464	463.79
70	621	621.00	70	605	605.03	70	564	563.98	70	482	481.93
75	621	621.00	75	621	620.51	75	581	580.67	75	501	501.49
80	621	621.00	80	621	621.00	80	599	598.51	80	523	522.68
85	621	621.00	85	621	621.00	85	618	617.90	85	546	545.50
90	621	621.00	90	621	621.00	90	621	621.00	90	570	570.23
95	621	621.00	95	621	621.00	95	621	621.00	95	597	596.83
100	621	621.00	100	621	621.00	100	621	621.00	100	621	621.00
105	621	621.00	105	621	621.00	105	621	621.00	105	621	621.00
110	621	621.00	110	621	621.00	110	621	621.00	110	621	621.00
110	2042	795.92	110	2042	766.92	110	2042	739.27	110	2042	690.04
115	2194	821.55	115	2194	794.59	115	2194	769.53	115	2194	726.24
120	2361	849.00	120	2361	824.45	120	2361	801.97	120	2361	765.12
125		878.42	125		856.54	125		836.87	125		807.07
130		910.25	130		890.97	130		874.41	130		852.23
135		944.34	135		928.00	135		915.03	135		900.91
140		980.89	140		967.79	140		958.57	140		953.33
145		1020.15	145		1010.84	145		1005.42	145		1009.81
150		1062.35	150		1056.88	150		1055.76	150		
155		1107.92	155		1106.38	155			155		
160		1156.42	160			160			160		
165		1208.78	165			165			165		
170		1265.05	170			170			170		
175		1325.37	175			175			175		
180		1390.04	180			180			180		
185		1459.41	185			185			185		
190		1533.55	190			190			190		
195		1613.49	195			195			195		
200		1699.01	200			200			200		
205		1790.55	205			205			205		
210		1888.61	210			210			210		
215		1993.61	215			215			215		
220		2105.69	220			220			220		
225		2225.77	225			225			225		
230		2353.75	230			230			230		

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Braidwood Unit 2

100 Degree Heat Up			Criticality Limit				Leak Test Limit			
Temperature	22 EFPY Pressure	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressure	Current PTLR Temp	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressures	Current PTLR Temps	Current PTLR Pressures
60	0	0	122	0	207	0	105	2000	186	2000
60	617	617	122	1000	207	621	122	2485	207	2485
65	617	617	122	1013	207	696				
70	617	617	125	1018	207	715				
75	617	617	130	1030	207	736				
80	617	617	135	1046	207	760				
85	617	617	140	1069	207	786				
90	617	617	145	1097	207	815				
95	617	617	150	1131	210	846				
100	617	617	155	1171	215	880				
105	619	619	160	1218	220	917				
110	621	621	165	1272	225	957				
115	621	621	170	1333	230	1000				
120	621	621	175	1402	235	1047				
125	621	621	180	1479	240	1097				
130	621	621	185	1566	245	1152				
135	621	621	190	1663	250	1210				
140	621	621	195	1770	255	1273				
140	1479	696	200	1890	260	1341				
145	1566	715	205	2023	265	1415				
150	1663	736	210	2170	270	1493				
155	1770	760	215	2332	275	1578				
160	1890	786			280	1669				
165	2023	815			285	1766				
170	2170	846			290	1871				
175	2332	880			295	1984				
180		917			300	2105				
185		957			305	2235				
190		1000			310	2374				
195		1047								
200		1097								
205		1152								
210		1210								
215		1273								
220		1341								
225		1415								
230		1493								
235		1578								
240		1669								
245		1766								
250		1871								
255		1984								
260		2105								
265		2235								
270		2374								

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Braidwood Unit 2

Steady State			25 Degree F Cooldown			50 Degree F Cooldown			100 Degree F Cooldown		
Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure
60	0	0	60	0	0	60	0	0	60	0	0
60	621	621	60	602	602	60	601	554	60	455	455
65	621	621	65	616	616	65	619	568	65	471	471
70	621	621	70	621	621	70	621	583	70	489	489
75	621	621	75	621	621	75	621	599	75	508	508
80	621	621	80	621	621	80	621	617	80	529	529
85	621	621	85	621	621	85	621	621	85	552	552
90	621	621	90	621	621	90	621	621	90	576	576
95	621	621	95	621	621	95	621	621	95	603	603
100	621	621	100	621	621	100	621	621	100	621	621
105	621	621	105	621	621	105	621	621	105	621	621
110	621	621	110	621	621	110	621	621	110	621	621
115	621	621	115	621	621	115	621	621	115	621	621
120	621	621	120	621	621	120	621	621	120	621	621
125	621	621	125	621	621	125	621	621	125	621	621
130	621	621	130	621	621	130	621	621	130	621	621
135	621	621	135	621	621	135	621	621	135	621	621
140	621	621	140	621	621	140	621	621	140	621	621
140	2396	1010	140	2396	991	140	2396	975	140	2396	957
145		1050	145		1034	145		1022	145		1013
150		1092	150		1080	150		1072	150		1074
155		1137	155		1129	155		1126	155		1137
160		1186	160		1183	160		1185	160		1186
165		1239	165		1239	165		1239	165		1239
170		1295	170		1295	170		1295	170		1295
175		1356	175		1356	175		1356	175		1356
180		1422	180		1422	180		1422	180		1422
185		1492	185		1492	185		1492	185		1492
190		1567	190		1567	190		1567	190		1567
195		1649	195		1649	195		1649	195		1649
200		1736	200		1736	200		1736	200		1736
205		1830	205		1830	205		1830	205		1830
210		1931	210		1931	210		1931	210		1931
215		2039	215		2039	215		2039	215		2039
220		2156	220		2156	220		2156	220		2156
225		2281	225		2281	225		2281	225		2281
230		2416	230		2416	230		2416	230		2416

ATTACHMENT 5
Comparison of 22 EFPY Data Points and
Current Byron and Braidwood Station PTLR Data Points
Byron Unit 1

100 Degree Heat Up			Criticality Limit				Leak Test Limit			
Temperature	22 EFPY Pressure	Current PTLR Pressure	22 EFPYTemps	22 EFPY Pressure	Current PTLR Temps	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressures	Current PTLR Temps	Current PTLR Pressures
60	0	0	163	0	225	0	146	2000	204	2000
60	587	587	163	732	225	587	163	2485	225	2485
65	587	587	163	732	225	588				
70	587	587	163	732	225	591				
75	587	587	163	732	225	596				
80	587	587	163	732	225	602				
85	587	587	163	732	225	611				
90	587	587	163	732	225	622				
95	587	587	163	733	225	634				
100	587	587	163	737	225	648				
105	587	587	163	743	225	665				
110	587	587	163	753	225	683				
115	587	587	163	766	225	703				
120	588	588	163	782	225	725				
125	591	591	165	801	225	750				
130	596	596	170	823	225	777				
135	602	602	175	849	230	806				
140	611	611	180	879	235	838				
145	621	621	185	913	240	872				
150	621	621	190	951	245	910				
155	621	621	195	994	250	950				
160	621	621	200	1042	255	994				
165	621	621	205	1095	260	1041				
170	621	621	210	1155	265	1092				
175	621	621	215	1221	270	1147				
180	621	621	220	1294	275	1206				
180	1294	750	225	1376	280	1269				
185	1376	777	230	1466	285	1338				
190	1466	806	235	1566	290	1411				
195	1566	838	240	1676	295	1490				
200	1676	872	245	1798	300	1575				
205	1798	910	250	1932	305	1666				
210	1932	950	255	2081	310	1764				
215	2081	994	260	2245	315	1869				
220	2245	1041	265	2425	320	1982				
225	2425	1092			325	2104				
230		1147			330	2234				
235		1206			335	2374				
240		1269								
245		1338								
250		1411								
255		1490								
260		1575								
265		1666								
270		1764								
275		1869								
280		1982								
285		2104								
290		2234								
295		2374								

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Byron Unit 1

Steady State			25 Degree F Cooldown			50 Degree F Cooldown			100 Degree F Cooldown		
Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure
60	0	0	60	0	0	60	0	0	60	0	0
60	613	613	60	561	561	60	509	509	60	402	402
65	621	621	65	572	572	65	520	520	65	414	414
70	621	621	70	582	582	70	531	531	70	427	427
75	621	621	75	594	594	75	544	544	75	442	442
80	621	621	80	607	607	80	557	557	80	458	458
85	621	621	85	620	620	85	572	572	85	475	475
90	621	621	90	621	621	90	588	588	90	494	494
95	621	621	95	621	621	95	605	605	95	514	514
100	621	621	100	621	621	100	621	621	100	535	535
105	621	621	105	621	621	105	621	621	105	559	559
110	621	621	110	621	621	110	621	621	110	584	584
115	621	621	115	621	621	115	621	621	115	611	611
120	587	621	120	621	621	120	621	621	120	621	621
125	588	621	125	621	621	125	621	621	125	621	621
130	621	621	130	621	621	130	621	621	130	621	621
135	621	621	135	621	621	135	621	621	135	621	621
140	621	621	140	621	621	140	621	621	140	621	621
145	621	621	145	621	621	145	621	621	145	621	621
150	621	621	150	621	621	150	621	621	150	621	621
155	621	621	155	621	621	155	621	621	155	621	621
160	621	621	160	621	621	160	621	621	160	621	621
165	621	621	165	621	621	165	621	621	165	621	621
170	621	621	170	621	621	170	621	621	170	621	621
175	621	621	175	621	621	175	621	621	175	621	621
180	621	621	180	621	621	180	621	621	180	621	621
180	2361	1207	180	2361	1205	180	2361	1205	180	2361	1205
185		1261	185			185			185		
190		1319	190			190			190		
195		1382	195			195			195		
200		1449	200			200			200		
205		1521	205			205			205		
210		1599	210			210			210		
215		1683	215			215			215		
220		1773	220			220			220		
225		1869	225			225			225		
230		1973	230			230			230		
235		2085									
240		2205									
245		2334									
250		2473									

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Byron Unit 2

100 Degree Heat Up			Criticality Limit				Leak Test Limit			
Temperature	22 EFPY Pressure	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressures	Current PTLR Temps	Current PTLR Pressure	22 EFPY Temps	22 EFPY Pressures	Current PTLR Temps	Current PTLR Pressures
60	0	0	106	0	219	0	89	2000	198	2000
60	621	621	106	1133	219	635	106	2485	219	2485
65	621	621	110	1186	219	674				
70	621	621	115	1197	219	660				
75	621	621	120	1202	219	650				
80	621	621	125	1215	219	643				
85	621	621	130	1235	219	638				
90	621	621	135	1263	219	637				
95	621	621	140	1299	219	637				
100	621	621	145	1342	219	641				
105	621	621	150	1393	219	646				
110	621	621	155	1452	219	654				
115	621	621	160	1520	219	664				
120	621	621	165	1598	219	676				
125	621	621	170	1686	219	690				
130	621	621	175	1784	219	707				
135	621	621	180	1895	219	725				
140	621	621	185	2018	219	746				
145	621	621	190	2155	219	770				
150	621	621	195	2304	219	796				
150	2155	707	200	2428	219	824				
155	2304	725			220	855				
160	2428	746			225	889				
165		770			230	926				
170		796			235	966				
175		824			240	1010				
180		855			245	1057				
180		889			250	1107				
185		926			255	1162				
190		966			260	1221				
195		1010			265	1285				
200		1057			270	1353				
205		1107			275	1427				
210		1162			280	1506				
215		1221			285	1591				
220		1285			290	1683				
225		1353			295	1781				
230		1427			300	1887				
235		1506			305	2001				
240		1591			310	2123				
245		1683			315	2254				
250		1781			320	2395				
255		1887								
260		2001								
265		2123								
270		2254								
275		2395								

ATTACHMENT 5
Comparison of 22 EFPY Data Points and Current Byron and Braidwood Station PTLR Data Points
Byron Unit 2

Steady State			25 Degree F Cooldown			50 Degree F Cooldown			100 Degree F Cooldown		
Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure	Temperature	22 EFPY Pressure	Current PTLR Pressure
60	0	0	60	0	0	60	0	0	60	0	0
60	621	621	60	574	574	60	523	523	60	418	418
65	621	621	65	585	585	65	534	534	65	431	431
70	621	621	70	597	597	70	547	547	70	446	446
75	621	621	75	610	610	75	561	561	75	462	462
80	621	621	80	621	621	80	576	576	80	480	480
85	621	621	85	621	621	85	592	592	85	498	498
90	621	621	90	621	621	90	609	609	90	519	519
95	621	621	95	621	621	95	621	621	95	541	541
100	621	621	100	621	621	100	621	621	100	564	564
105	621	621	105	621	621	105	621	621	105	590	590
110	621	621	110	621	621	110	621	621	110	618	618
115	621	621	115	621	621	115	621	621	115	621	621
120	621	621	120	621	621	120	621	621	120	621	621
125	621	621	125	621	621	125	621	621	125	621	621
130	621	621	130	621	621	130	621	621	130	621	621
135	621	621	135	621	621	135	621	621	135	621	621
140	621	621	140	621	621	140	621	621	140	621	621
145	621	621	145	621	621	145	621	621	145	621	621
150	621	621	150	621	621	150	621	621	150	621	621
150	2433	995	150	2433	974	150	2433	957	150	2433	935
155		1033	155		1016	155		1002	155		989
160		1075	160		1061	160		1051	160		1048
165		1119	165		1109	165		1104	165		1112
170		1166	170		1161	170		1161	170		
175		1217	175		1217	175			175		
180		1272	180			180			180		
185		1331	185			185			185		
190		1395	190			190			190		
195		1463	195			195			195		
200		1536	200			200			200		
205		1615	205			205			205		
210		1700	210			210			210		
215		1791	215			215			215		
220		1889	220			220			220		
225		1995	225			225			225		
230		2108	230			230			230		
235		2230	235			235			235		
240		2361	240			240			240		

ATTACHMENT 6-A

Revised P-T Limit Curves for Braidwood Station, Units 1 and 2

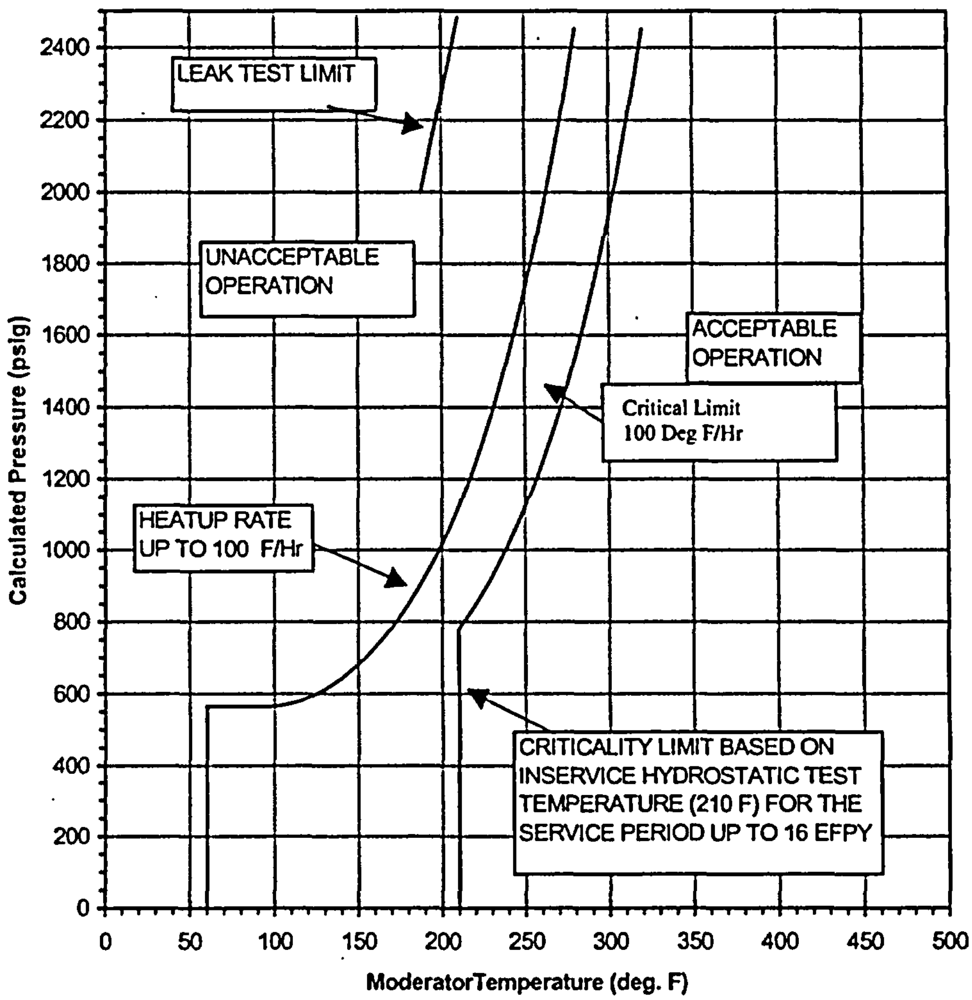

Figure 2.1
Braidwood Unit 1 Reactor Coolant System Heatup Limitations (heatup rate up to 100°F/hr)
Applicable for the First 16 EFPY
(Without Margins for Instrumentation Errors)

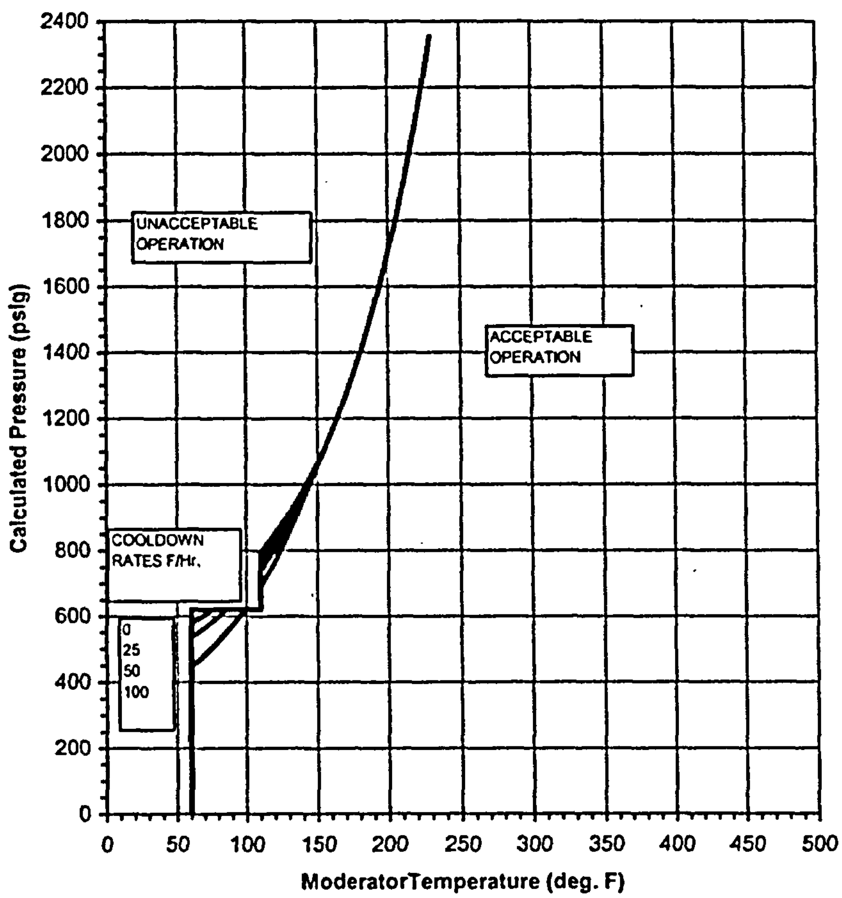

Figure 2.2

Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 0, 25, 50 and 100 °F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
(Page 1 of 2)
Braidwood Unit 1 Heatup* Data Points at 16 EFPY
(Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	210	0	188	2000
60	565.09	210	611.83	210	2485
65	565.09	210	597.56		
70	565.09	210	585.60		
75	565.09	210	576.77		
80	565.09	210	570.35		
85	565.09	210	566.61		
90	565.09	210	565.09		
95	565.09	210	565.87		
100	565.87	210	568.69		
105	568.69	210	573.56		
110	573.56	210	580.30		
115	580.30	210	588.84		
120	588.84	210	599.36		
125	599.36	210	611.78		
130	611.78	210	626.07		
135	626.07	210	642.16		
140	642.16	210	660.36		
145	660.36	210	680.59		
150	680.59	210	702.80		
155	702.80	210	727.33		
160	727.33	210	754.07		
165	754.07	210	783.17		
170	783.17	215	814.98		
175	814.98	220	849.37		
180	849.37	225	886.54		
185	886.54	230	926.73		
190	926.73	235	970.11		
195	970.11	240	1016.91		
200	1016.91	245	1067.33		
205	1067.33	250	1121.63		
210	1121.63	255	1180.01		
215	1180.01	260	1242.62		
220	1242.62	265	1309.84		
225	1309.84	270	1382.03		
230	1382.03	275	1459.45		
235	1459.45	280	1542.27		
240	1542.27	285	1630.97		
245	1630.97	290	1726.05		
250	1726.05	295	1827.80		

Table 2.1a Page 2 of 2					
Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
255	1827.80	300	1936.51		
260	1936.51	305	2052.39		
265	2052.39	310	2176.33		
270	2176.33	315	2308.42		
275	2308.42	320	2449.09		
280	2449.09				

* Heatup and Cooldown data includes vessel flange requirements of 110°F and 621 psig per 10CFR50, Appendix G.

Table 2.1b
Page 1 of 1
Braidwood Unit 1 Cooldown* Data Points at 16 EFPY**
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	620.27	60	577.45	60	534.28	60	446.98
65	621.00	65	590.68	65	548.52	65	463.79
70	621.00	70	605.03	70	563.98	70	481.93
75	621.00	75	620.51	75	580.67	75	501.49
80	621.00	80	621.00	80	598.51	80	522.68
85	621.00	85	621.00	85	617.90	85	545.50
90	621.00	90	621.00	90	621.00	90	570.23
95	621.00	95	621.00	95	621.00	95	596.83
100	621.00	100	621.00	100	621.00	100	621.00
105	621.00	105	621.00	105	621.00	105	621.00
110	621.00	110	621.00	110	621.00	110	621.00
110	795.92	110	766.92	110	739.27	110	690.04
115	821.55	115	794.59	115	769.53	115	726.24
120	849.00	120	824.45	120	801.97	120	765.12
125	878.42	125	856.54	125	836.87	125	807.07
130	910.25	130	890.97	130	874.41	130	852.23
135	944.34	135	928.00	135	915.03	135	900.91
140	980.89	140	967.79	140	958.57	140	953.33
145	1020.15	145	1010.84	145	1005.42	145	1009.81
150	1062.35	150	1056.88	150	1055.76		
155	1107.92	155	1106.38				
160	1156.42						
165	1208.78						
170	1265.05						
175	1325.37						
180	1390.04						
185	1459.41						
190	1533.55						
195	1613.49						
200	1699.01						
205	1790.55						
210	1888.61						
215	1993.61						
220	2105.69						
225	2225.77						
230	2353.75						

* Heatup and Cooldown data includes vessel flange requirements of 110°F and 621 psig per 10CFR50, Appendix G.

** For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.

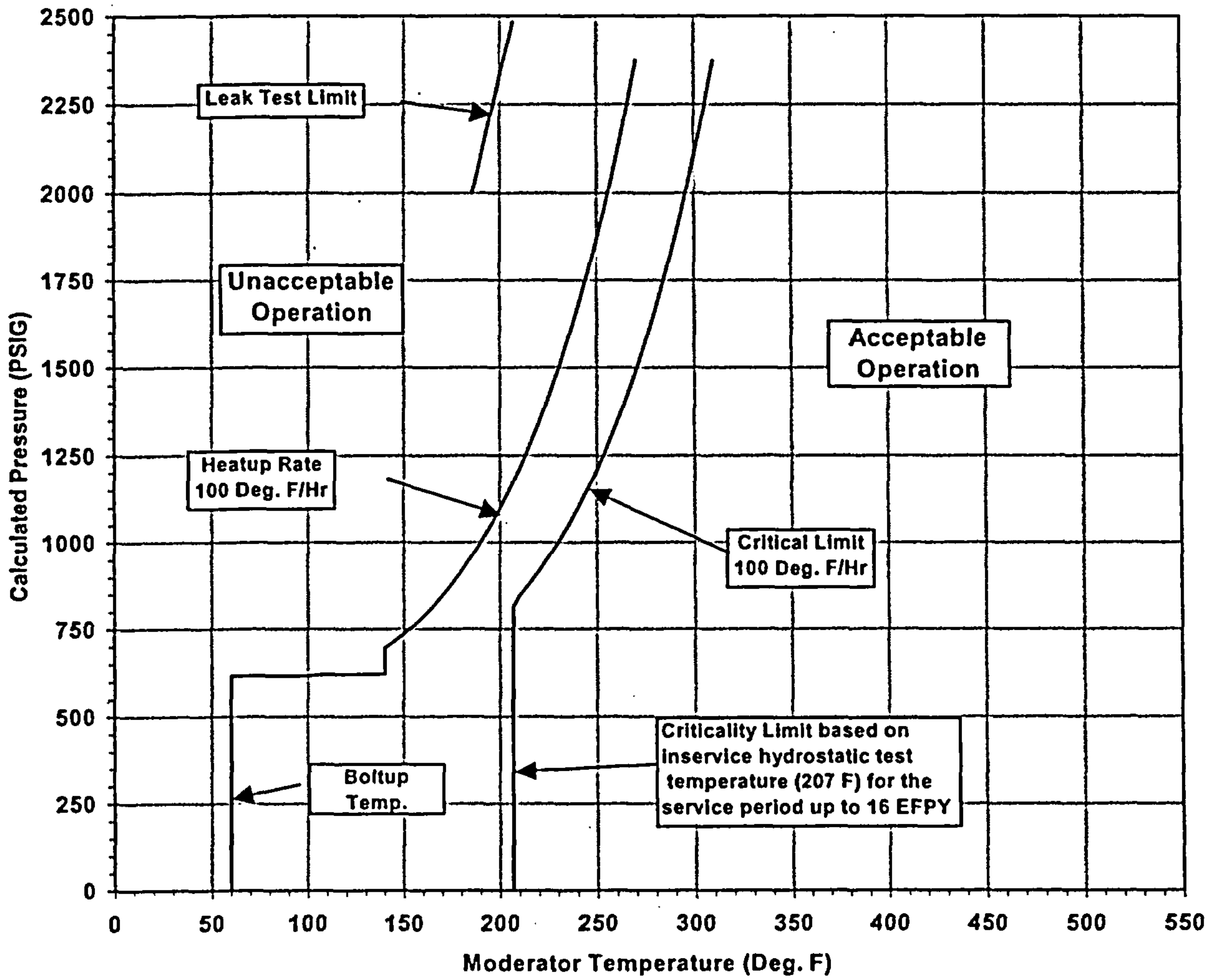

Figure 2.1
Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 16 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)

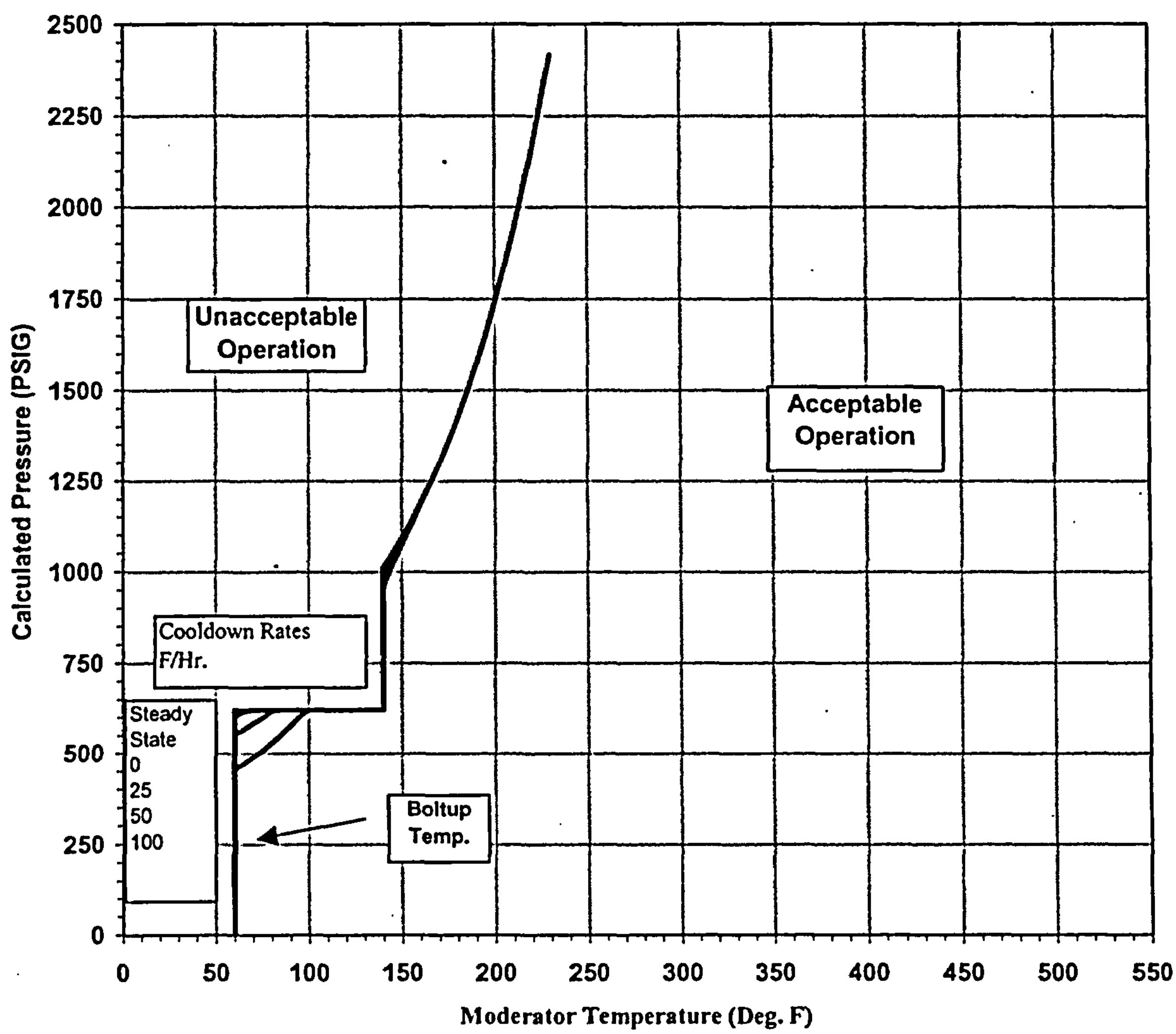

Figure 2.2
Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to the First 16 EFPY using 1996 Appendix G Methodology (Without Margins of Instrumentation Errors)

Table 2.1a
(Page 1 of 2)

Braidwood Unit 2 Heatup* Data Points at 16 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	207	0	186	2000
60	617	207	621	207	2485
65	617	207	621		
70	617	207	621		
75	617	207	621		
80	617	207	621		
85	617	207	621		
90	617	207	621		
95	617	207	621		
100	617	207	621		
105	619	207	621		
110	621	207	621		
115	621	207	621		
120	621	207	621		
125	621	207	621		
130	621	207	621		
135	621	207	621		
140	621	207	696		
140	621	207	715		
140	696	207	736		
145	715	207	760		
150	736	207	786		
155	760	207	815		
160	786	210	846		
165	815	215	880		
170	846	220	917		
175	880	225	957		
180	917	230	1000		
185	957	235	1047		
190	1000	240	1097		
195	1047	245	1152		
200	1097	250	1210		
205	1152	255	1273		

Table 2.1a
Page 2 of 2

Heatup Curve

100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
210	1210	260	1341		
215	1273	265	1415		
220	1341	270	1493		
225	1415	275	1578		
230	1493	280	1669		
235	1578	285	1766		
240	1669	290	1871		
245	1766	295	1984		
250	1871	300	2105		
255	1984	305	2235		
260	2105	310	2374		
265	2235				
270	2374				

* Heatup and Cooldown data includes the vessel flange requirements of 140 °F and 621 psig per 10CFR50, Appendix G..

Table 2.1b

(Page 1 of 1)

Braidwood Unit 2 Cooldown* Data at 16 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	621	60	602	60	554	60	455
65	621	65	616	65	568	65	471
70	621	70	621	70	583	70	489
75	621	75	621	75	599	75	508
80	621	80	621	80	617	80	529
85	621	85	621	85	621	85	552
90	621	90	621	90	621	90	576
95	621	95	621	95	621	95	603
100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621
140	621	140	621	140	621	140	621
140	1010	140	991	140	975	140	957
145	1050	145	1034	145	1022	145	1013
150	1092	150	1080	150	1072	150	1074
155	1137	155	1129	155	1126	155	1137
160	1186	160	1183	160	1185	160	1186
165	1239	165	1239	165	1239	165	1239
170	1295	170	1295	170	1295	170	1295
175	1356	175	1356	175	1356	175	1356
180	1422	180	1422	180	1422	180	1422
185	1492	185	1492	185	1492	185	1492
190	1567	190	1567	190	1567	190	1567
195	1649	195	1649	195	1649	195	1649
200	1736	200	1736	200	1736	200	1736
205	1830	205	1830	205	1830	205	1830
210	1931	210	1931	210	1931	210	1931
215	2039	215	2039	215	2039	215	2039
220	2156	220	2156	220	2156	220	2156
225	2281	225	2281	225	2281	225	2281
230	2416	230	2416	230	2416	230	2416

* Heatup and Cooldown data includes the vessel flange requirements of 140 °F and 621 psig per 10CFR50, Appendix G.

ATTACHMENT 6-B

Revised PT Limit Curves for Byron Station, Units 1 and 2

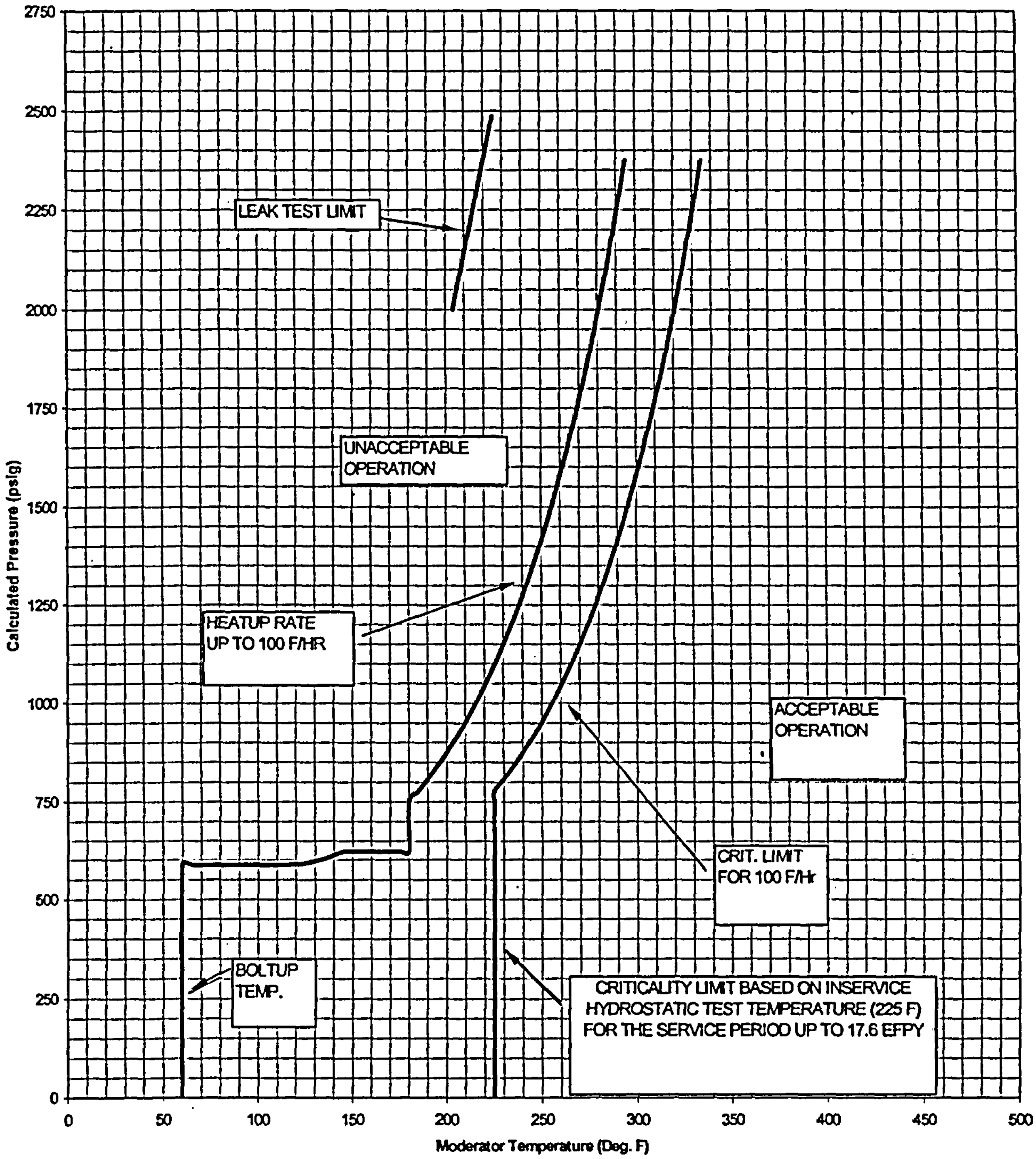

Figure 2.1:

Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 100°F/hr) Applicable for 17.6 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

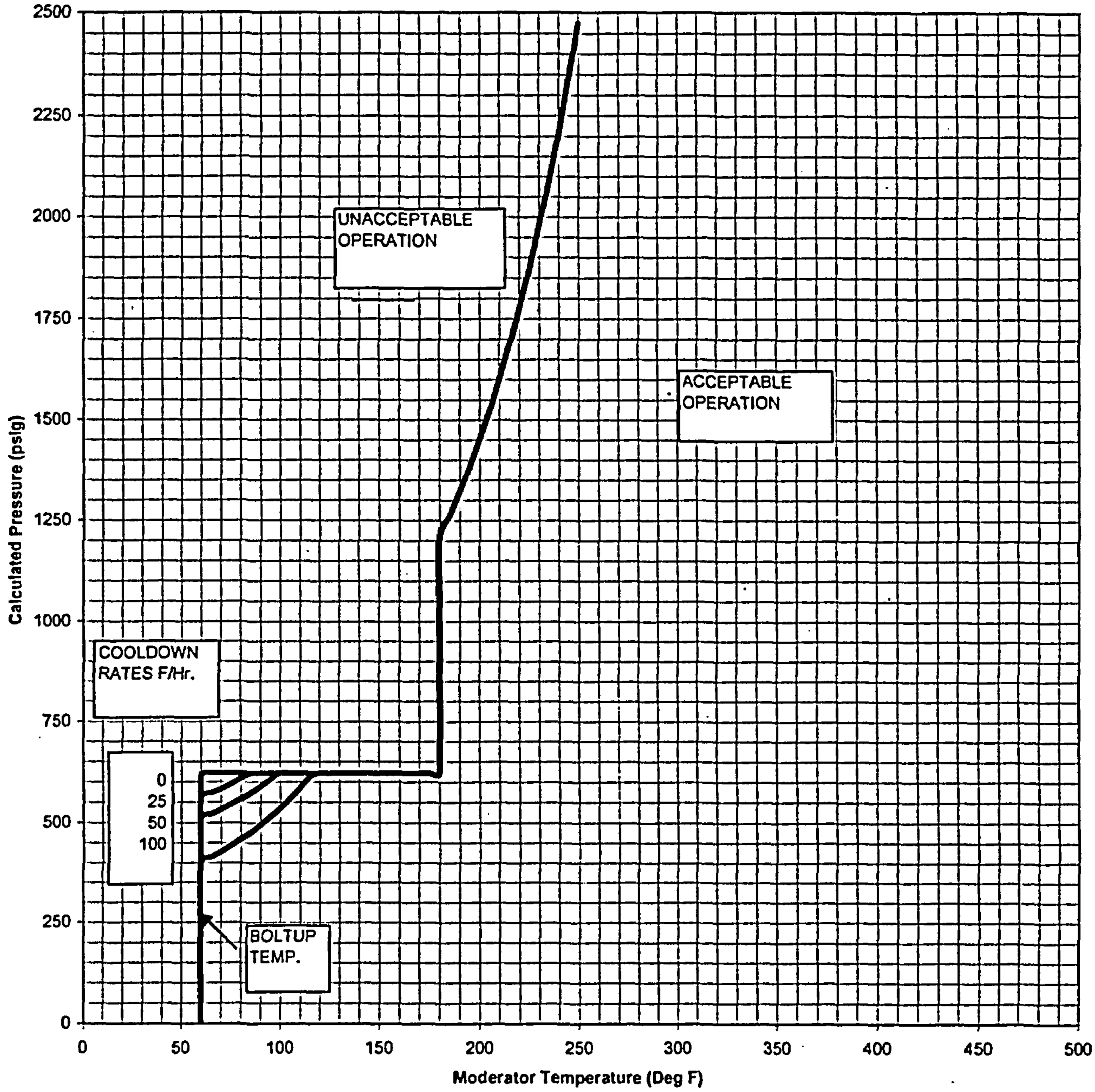

Figure 2.2:
Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for 17.6 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

Table 2.1
Byron Unit 1 Heatup and Cooldown Data Points at 17.6 EFPY (Without Margins for Instrumentation Errors and Using the 1996 Appendix G Methodology)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
60	0	225	0	204	2000	60	0	60	0	60	0	60	0
60	587	225	587	225	2485	60	613	60	561	60	509	60	402
65	587	225	587			65	621	65	572	65	520	65	414
70	587	225	587			70	621	70	582	70	531	70	427
75	587	225	587			75	621	75	594	75	544	75	442
80	587	225	587			80	621	80	607	80	557	80	458
85	587	225	587			85	621	85	620	85	572	85	475
90	587	225	587			90	621	90	621	90	588	90	494
95	587	225	587			95	621	95	621	95	605	95	514
100	587	225	587			100	621	100	621	100	621	100	535
105	587	225	587			105	621	105	621	105	621	105	559
110	587	225	587			110	621	110	621	110	621	110	584
115	587	225	587			115	621	115	621	115	621	115	611
120	588	225	587			120	621	120	621	120	621	120	621
125	591	225	587			125	621	125	621	125	621	125	621
130	596	225	587			130	621	130	621	130	621	130	621
135	602	225	587			135	621	135	621	135	621	135	621
140	611	225	587			140	621	140	621	140	621	140	621
145	621	225	588			145	621	145	621	145	621	145	621
150	621	225	591			150	621	150	621	150	621	150	621
155	621	225	596			155	621	155	621	155	621	155	621
160	621	225	602			160	621	160	621	160	621	160	621
165	621	225	611			165	621	165	621	165	621	165	621
170	621	225	622			170	621	170	621	170	621	170	621
175	621	225	634			175	621	175	621	175	621		
180	621	225	648			180	621	180	621				
180	750	225	665			180	1207	180	1205				
185	777	225	683			185	1261						
190	806	225	703			190	1319						
195	838	225	725			195	1382						
200	872	225	750			200	1449						
205	910	225	777			205	1521						
210	950	230	806			210	1599						
215	994	235	838			215	1683						
220	1041	240	872			220	1773						
225	1092	245	910			225	1869						
230	1147	250	950			230	1973						
235	1206	255	994			235	2085						

Table 2.1 (Continued)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
240	1269	260	1041			240	2205						
245	1338	265	1092			245	2334						
250	1411	270	1147			250	2473						
255	1490	275	1206										
260	1575	280	1269										
265	1666	285	1338										
270	1764	290	1411										
275	1869	295	1490										
280	1982	300	1575										
285	2104	305	1666										
290	2234	310	1764										
295	2374	315	1869										
		320	1982										
		325	2104										
		330	2234										
		335	2374										

Note 1: Heatup and Cooldown data includes the vessel flange requirements of 180 °F and 621 psig per 10CFR50, Appendix G.
Note 2: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.
Note 3: Temperatures and pressures are given in ° F and psig, respectively.

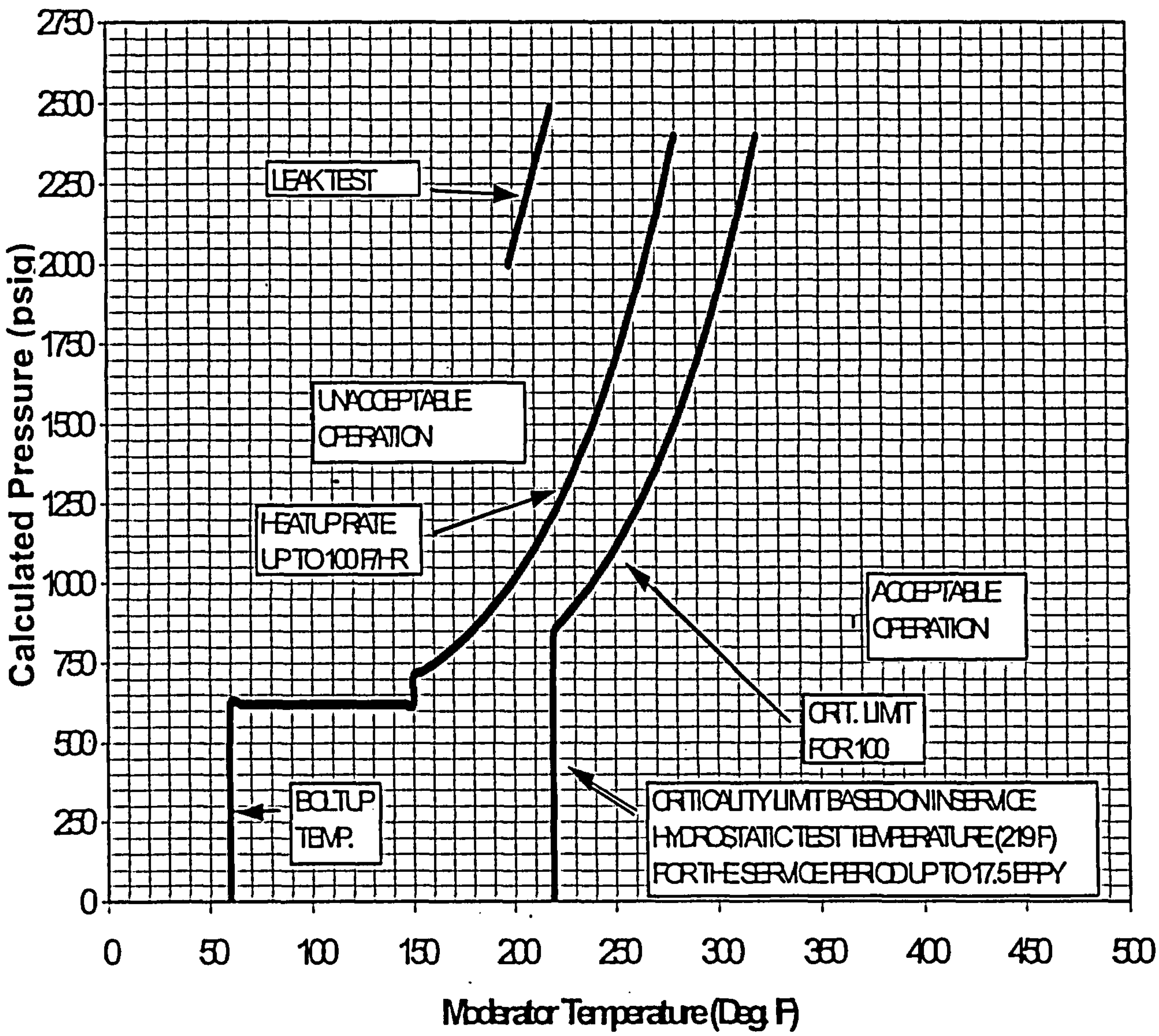

Figure 2.1:
Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates up to 100 °F/hr) Applicable for 17.5 EFPY (Without margins for instrumentation errors and using 1996 Appendix G Methodology)

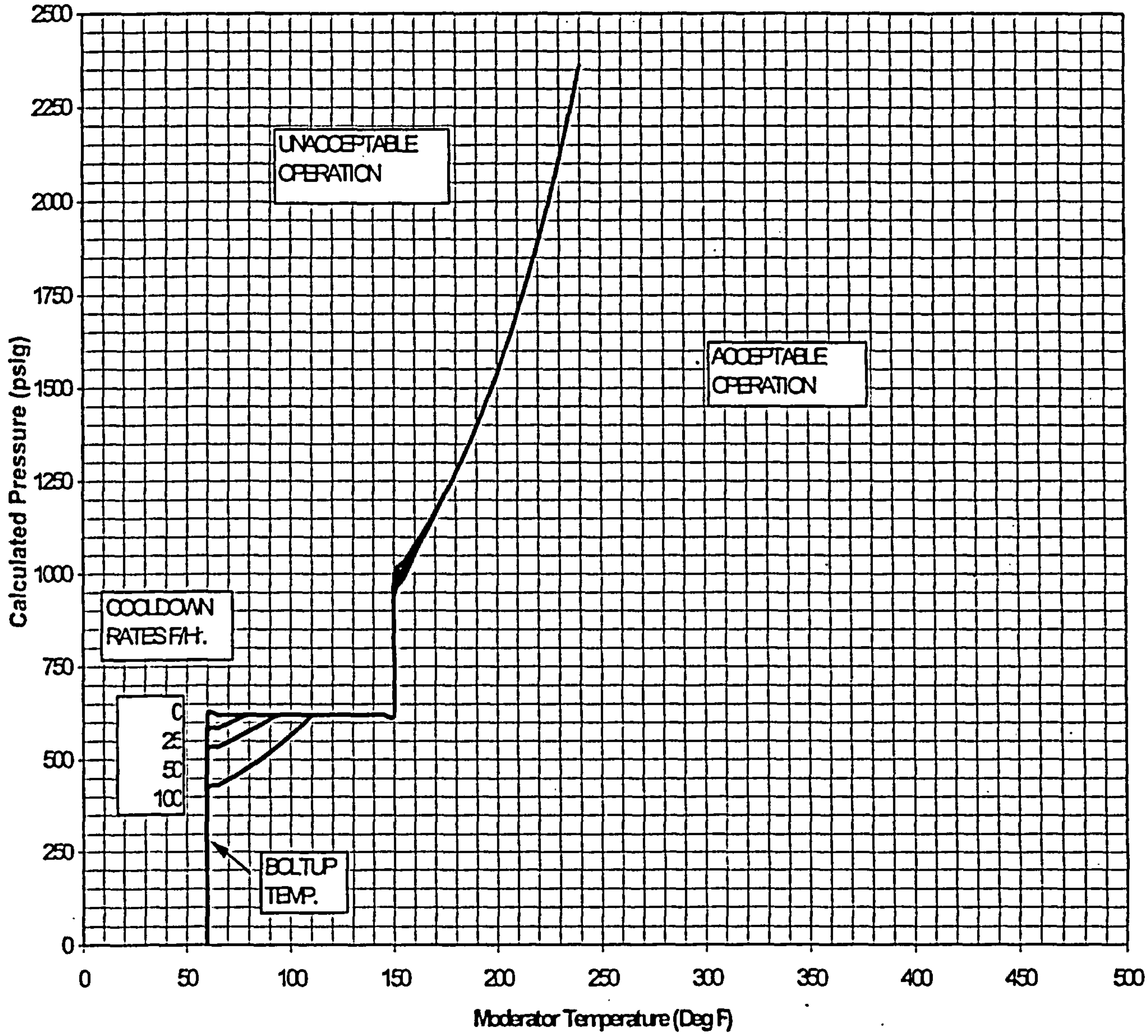

Figure 2.2:
Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 °F/hr) Applicable for 17.5 EFPY (Without Margins for Instrumentation Errors and using 1996 Appendix G Methodology)

PRESSURE AND TEMPERATURE LIMITS REPORT

Table 2.1: Byron Unit 2 Heatup and Cooldown Data Points at 17.5 EFPY (Without Margins for Instrumentation Errors and Using the 1996 Appendix G Methodology)

Heatup Curve						Cooldown Curves							
100 F Heatup		Criticality Limit		Leak Test Limit		Steady State		25 F Cooldown		50 F Cooldown		100 F Cooldown	
T	P	T	P	T	P	T	P	T	P	T	P	T	P
60	0	219	0	198	2000	60	0	60	0	60	0	60	0
60	621	219	635	219	2485	60	621	60	574	60	523	60	418
65	621	219	674			65	621	65	585	65	534	65	431
85	621	219	660			70	621	70	597	70	547	70	446
90	621	219	650			75	621	75	610	75	561	75	462
95	621	219	643			80	621	80	621	80	576	80	480
100	621	219	638			85	621	85	621	85	592	85	498
105	621	219	637			90	621	90	621	90	609	90	519
110	621	219	637			95	621	95	621	95	621	95	541
115	621	219	641			100	621	100	621	100	621	100	564
120	621	219	646			105	621	105	621	105	621	105	590
125	621	219	654			110	621	110	621	110	621	110	618
130	621	219	664			115	621	115	621	115	621	115	621
135	621	219	676			120	621	120	621	120	621	120	621
140	621	219	690			125	621	125	621	125	621	125	621
145	621	219	707			130	621	130	621	130	621	130	621
150	621	219	725			135	621	135	621	135	621	135	621
150	707	219	746			140	621	140	621	140	621	140	621
155	725	219	770			145	621	145	621	145	621	145	621
160	746	219	796			150	621	150	621	150	621	150	621
165	770	219	824			150	995	150	974	150	957	150	935
170	796	220	855			155	1033	155	1016	155	1002	155	989
175	824	225	889			160	1075	160	1061	160	1051	160	1048
180	855	230	926			165	1119	165	1109	165	1104	165	1112
185	889	235	966			170	1166	170	1161	170	1161		
190	926	240	1010			175	1217	175	1217				
195	966	245	1057			180	1272						
200	1010	250	1107			185	1331						
205	1057	255	1162			190	1395						
210	1107	260	1221			195	1463						
215	1162	265	1285			200	1536						
220	1221	270	1353			205	1615						
225	1285	275	1427			210	1700						
230	1353	280	1506			215	1791						
235	1427	285	1591			220	1889						
240	1506	290	1683			225	1995						
245	1591	295	1781			230	2108						
250	1683	300	1887			235	2230						
255	1781	305	2001			240	2361						
260	1887	310	2123										
265	2001	315	2254										
270	2123	320	2395										
275	2254												
280	2395												

Note 1: Heatup and Cooldown data includes the vessel flange requirements of 150 °F and 621 psig per 10CFR50, Appendix G.

Note 2: For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided.

Note 3: Temperatures and pressures are given in ° F and psig, respectively.