Exelon Generation.

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RS-19-087

10 CFR 50.90
10 CFR 50.12
10 CFR 50.60

September 30, 2019

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

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

Subject: License Amendment Request and 10 CFR 50.12 Exemption Request for Alternate Material Properties Bases

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2 (Braidwood), and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (Byron). The amendment request proposes to revise Braidwood and Byron Technical Specifications (TS) 5.6.6 to allow the use of a new methodology; AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials." The revision would add BAW-2308, Revisions 1-A and 2-A as an approved methodology to TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," for determining RCS pressure-temperature (P-T) limits.

In addition, in accordance with 10 CFR 50.12 and 10 CFR 50.60(b), EGC requests an exemption to portions of the following regulations:

1. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," which sets forth fracture toughness requirements for protection against pressurized thermal shock (PTS); and

2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements," which sets forth 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.

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The attached request is subdivided as follows:

- Attachment 1 provides a description and evaluation of the proposed changes.
- Attachment 2 provides the markup of the affected TS pages for Braidwood.
- Attachment 3 provides the markup of the affected TS pages for Byron.
- Attachment 4 provides the Exemption Request.

These proposed changes have been reviewed and approved by the Braidwood and Byron Plant Operations Review Committees in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed license amendments within one year of this submittal date; i.e., by September 30, 2020. Once approved, the amendments shall be implemented within 60 days.

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

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Lisa M. Zurawski at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of September 2019.

Respectfully,

Dwi Murray
Senior Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1) Evaluation of Proposed Changes
2) Markup of Technical Specifications Pages – Braidwood Station, Units 1 and 2
3) Markup of Technical Specifications Pages – Byron Station, Units 1 and 2
4) Exemption Request

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

ATTACHMENT 1
Evaluation of Proposed Changes

Subject: **License Amendment Request to revise Braidwood and Byron Technical Specifications (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)"**

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2 (Braidwood), and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (Byron).

The proposed change revises the Braidwood and Byron Technical Specifications (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to allow the use of a new methodology for determining RCS pressure and temperature (P-T) limits, as described by the following report:

AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials."

In addition, the proposed amendments would require exemption, in accordance with 10 CFR 50.12 and 10 CFR 50.60(b), to portions of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements." Attachment 4 provides the Exemption Request to portions of 10 CFR 50.61 and 10 CFR 50, Appendix G.

2.0 DETAILED DESCRIPTION

Currently, TS 5.6.6 reference the following NRC Safety Evaluations (SEs) and topical report, which describe the NRC approved PTLR methodologies for Braidwood and Byron:

- 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,"
- NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"
- Westinghouse WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2"

The proposed amendments would allow use of a new method, the Master Curve Methodology, as described in BAW-2308, Revisions 1-A and 2-A, for determining the adjusted reference nil-ductility temperature (RT_{NDT}). The Adjusted Reference Temperatures (ARTs), which support the P-T limits, were determined using Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." However, for the Linde 80 welds in the extended beltline region of the Reactor Pressure Vessel (RPV), alternate initial RT_{NDT} values were used in accordance with the NRC-approved Topical Report BAW-2308.

AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, were approved for referencing in plant-specific license amendments in SEs dated August 4, 2005 (Reference 4) and March 24, 2008 (Reference 5), respectively. The proposed amendments would revise TS 5.6.6 to add the

NRC SE approving the use of the above methodology for determining RCS P-T limits. Approval of this methodology will allow future changes to the PTLR to be performed by the licensee in accordance with 10 CFR 50.59, provided the methodology approved by the NRC is used to develop these PTLR changes.

2.1 Proposed Changes

TS 5.6.6.b

Replace:

4. The PTLR will contain the complete identification for each of the TS referenced Topical Report used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements; and

With:

4. NRC letter dated [NRC Exemption Request Approval Date], "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G for Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2," and NRC letter date [NRC SE Date], "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 – Issuance of Amendments Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications," and

5. The PTLR will contain the complete identification for each of the TS referenced Topical Report used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements; and

Basis for the change: The proposed change adds the NRC SE for this License Amendment Request as an approved analytical method to determine RCS P-T limits.

3.0 TECHNICAL EVALUATION

The proposed analytical method to be used to determine the revised RCS P-T limits for Braidwood and Byron are discussed in the NRC-approved AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A. The use of this topical report is needed to specifically address the Linde 80 welds which attach various nozzle forgings to the shell plates in the reactor pressure vessel's (RPV) extended beltline region.

This topical report was developed to provide an alternate method for determining the initial RT_{NDT} for Linde 80 welds in RPVs. The alternative methodology was based on brittle-to-ductile transition range fracture toughness test data of the weld materials, in accordance with American Standard for Testing and Materials (ASTM) Standard Test Method E1921 and using ASME Code Case N-629. This methodology would be used in lieu of the nil-ductility reference temperature parameter specified in ASME Section III, Paragraph NB-2331. The use of ASME Section III, Paragraph NB-2331, is specified in 10 CFR 50, Appendix G(II)(D)(i), and its use is also described in RG 1.99, Revision 2. Revision 1-A of Topical Report BAW-2308 was approved by the NRC via a SE dated August 4, 2005. Revision 2-A, which supplements Revision 1-A of Topical Report BAW-2308, was approved by the NRC via a SE dated March 24, 2008.

The SE for Revision 1-A of Topical Report BAW-2308 contained six Conditions and Limitations. Licensees had to satisfy four of these Conditions and Limitations in order to use the alternate methodology. The first three Conditions and Limitations Items describe performance requirements for use of the methodology. These Conditions and Limitations have been satisfied, as described in Attachment 4. The fourth Condition and Limitation Item requires an exemption to portions of 10 CFR 50.61 and 10 CFR 50, Appendix G. Attachment 4 provides the Exemption Request to portions of 10 CFR 50.61 and 10 CFR 50, Appendix G in accordance with 10 CFR 50.12 and 10 CFR 50.60(b).

Topical Report BAW-2308 was revised to incorporate additional information from the Pressurized Water Reactor Owner's Group (PWROG) to satisfy the remaining two Conditions and Limitations Items contained in the SE for Revision 1-A. The SE for Revision 2-A required licensees to address Conditions and Limitations Items one through four contained in the SE for Revision 1-A.

The revised Braidwood and Byron P-T limit curves have been developed using the NRC-approved Westinghouse Owner's Group Topical Report WCAP-14040-A, Revision 4 and the NRC-approved AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A. The Low Temperature Overpressure Protection reanalysis will also be developed using the aforementioned methodologies. Since the methodologies are approved for use by the NRC, the methodologies provide an acceptable means of satisfying 10 CFR 50, Appendix G, which governs the development of P-T limit curves and Low Temperature Overpressure Protection limits.

TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," requires revision to reflect the use of the above mentioned alternative methodologies. In accordance with TS 5.6.6 and after NRC approval of this License Amendment Request and Exemption Request, revised PTLRs will be submitted to the NRC as appropriate.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

10 CFR 50, Appendix G, "Fracture Toughness Requirements," requires P-T limits and minimum temperature requirements be established for the reactor pressure vessel. These limits are defined by the various operating conditions. The operating conditions include, but are not limited to, hydrostatic pressure and leak testing, and normal operation including anticipated operational occurrences. The regulation also requires the P-T limits be at least as conservative as the limits obtained by following the methods of analysis and margins contained in the ASME Code, Section XI, Appendix G.

10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," established screening criteria on Pressurized Water Reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. The RT_{PTS} screening values were set by the NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of

license. The procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values is consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The revised Braidwood and Byron P-T limit curves have been developed using ASME Code, Section XI, Appendix G. The ARTs were determined using Regulatory Guide 1.99, Revision 2. For the Linde 80 welds, alternate initial RT_{NDT} values were used in accordance with Topical Report BAW-2308, Revisions 1-A and 2-A. In order to utilize these alternate initial RT_{NDT} values, an exemption request in accordance with 10 CFR 50.12 and 10 CFR 50.60(b) has been provided in Attachment 4.

4.2 Precedent

The proposed amendments would allow use of a new method, as described in AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," for determining the adjusted reference nil-ductility temperature (RT_{NDT}). The NRC approved the use of Topical Report BAW-2308, Revisions 1-A and 2-A methodology for Point Beach Nuclear Plant, Units 1 and 2 by an SE dated June 30, 2014 (Reference 8), Oconee Nuclear Power Station, Units 1, 2 and 3 by an SE dated February 27, 2014 (Reference 9), and Three Mile Island Nuclear Station Unit 1 by an SE dated December 13, 2013 (Reference 10).

4.3 No Significant Hazards Consideration Determination

The proposed amendments would allow use of a new method, the Master Curve Methodology, as described in AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," for determining the adjusted reference nil-ductility temperature (RT_{NDT}).

The proposed amendments would revise Technical Specifications (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to add the Master Curve Methodology as an approved methodology.

Exelon Generation Company, LLC (EGC) has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

1.0 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems or components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits.

There will be no adverse change to normal plant operating parameters, engineered safety feature actuation setpoints, accident mitigation capabilities, or accident analysis assumptions or inputs. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

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

2.0 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change does not impose any new or different requirements or eliminate any existing requirements. The proposed change is consistent with the current safety analysis assumptions and current plant operating practice. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. Equipment important to safety will continue to operate as designed. The change does not result in any event previously deemed incredible being made credible. The change does not result in adverse conditions or result in any increase in the challenges to safety systems.

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

3.0 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change does not alter safety limits, limiting safety system settings, or limiting conditions for operation. The setpoints at which protective actions are initiated are not altered by the proposed change. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendments will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components important to safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendments (i) involves no significant hazards consideration; (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (iii) there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, in accordance with 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

6.0 REFERENCES

1. Letter from Robert A. Capra (U.S. NRC) to Oliver D. Kingsley (Exelon), "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report – (TAC NOS. M98799, M98800, M98801 and M98802)," dated January 21, 1998. (ACN 9802040389)

2. Letter from Mahesh Chawla (U.S. NRC) to Oliver D. Kingsley (Exelon), "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 – (TAC NOS. MB0751, MB0752, MB0753 and MB0754)," dated August 8, 2001. (ML011720218)

3. Westinghouse WCAP-16143, "Reactor Vessel Closure Head /Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2." (ML15232A441)

4. Letter from Herbert N. Berkow (U.S. NRC) to Jerald S. Holm (AREVA NP), " Final Safety Evaluation for Topical Report BAW-2308, Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC No. MB6636)," dated August 4, 2005. (ML052070408)

5. Letter from Ho K. Nieh (U.S. NRC) to Gordon Bischoff (Owners Group Program Management Office), "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2308, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC No. MD4241)," dated March 24, 2008. (ML080770349)

6. Westinghouse Electric Company Report, WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" dated May 2004. (ML050120209)

7. U.S. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

8. NRC letter to NextEra Point Beach, LLC, dated June 30, 2014, "Point Beach Nuclear Plant, Units 1 and 2 – Issuance of Amendment Regarding Change to Technical Specification 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" (TAC Nos. MF0532 and MF0533)." (ML14126A378)

9. NRC letter to Duke Energy Carolinas, LLC, dated February 27, 2014, "Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Revised Pressure – Temperature Limits (TAC Nos. MF0763, MF0764, and MF0765)." (ML14041A093)

10. NRC letter to Exelon Generation Company, LLC, dated December 13, 2013, "Three Mile Island Nuclear Station Unit 1 – Issuance of Amendment Re: Revision to the Pressure and Temperature Limit Curves and the Low Temperature Overpressure Protection Limits (TAC No. MF0424)." (ML13325A023)

ATTACHMENT 2

MARKUP OF TECHNICAL SPECIFICATIONS PAGE

BRAIDWOOD STATION
UNITS 1 and 2

Renewed Facility Operting License Nos. NPF-72 and NPF-77

Docket Nos. STN-50-456 and STN-50-457

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 the following documents:

   1. 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,"

   2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"

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

   4. NRC letter dated [NRC Exemption Request Approval Date], "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G for Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2," and NRC letter date [NRC SE Date], "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 - Issuance of Amendments Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications," and

   ~~4.~~ 5. The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements); and

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

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or G 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

MARKUP OF TECHNICAL SPECIFICATIONS PAGE

BYRON STATION
UNITS 1 and 2

Renewed Facility Operting License Nos. NPF-37 and NPF-66

Docket Nos. STN-50-454 and STN-50-455

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 the following documents:

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

2. NRC letter dated August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2,"

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

4. NRC letter dated [NRC Exemption Request Approval Date], "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G for Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2," and NRC letter date [NRC SE Date], "Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 - Issuance of Amendments Regarding Reactor Coolant System Pressure and Temperature Limits Report Technical Specifications," and

~~4.~~ 5 The PTLR will contain the complete identification for each of the TS referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements); and

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

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or G 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

EXEMPTION REQUEST

BRAIDWOOD STATION
UNITS 1 and 2

Renewed Facility Operting License Nos. NPF-72 and NPF-77

Docket Nos. STN-50-456 and STN-50-457

BYRON STATION
UNITS 1 and 2

Renewed Facility Operting License Nos. NPF-37 and NPF-66

Docket Nos. STN-50-454 and STN-50-455

ATTACHMENT 4
Exemption Request

I. SPECIFIC EXEMPTION REQUEST

In accordance with 10 CFR 50.12, "Specific exemptions," paragraph (a)(1) and (a)(2)(v), Exelon Generation Company, LLC (EGC) is requesting NRC approval of an exemption from certain requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The requested exemption would allow use of an alternate method, the Master Curve Methodology, as described in AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," for determining the initial, unirradiated material reference temperatures of the Linde 80 weld materials present in the reactor pressure vessel (RPV) extended beltline region for Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Units 1 and 2 (Byron).

II. BASIS FOR EXEMPTION REQUEST

BACKGROUND

10 CFR 50.61(a)(5) and 10 CFR 50, Appendix G(II)(D)(i), require that the pre-service or unirradiated condition reference temperature, nil-ductility transition, (RT_{NDT}) be evaluated according to the procedures in the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331, which requires Charpy V-notch impact tests and drop weight tests. In addition, 10 CFR 50.61 permits other methods for use in determining the initial material properties provided such methods are approved by the Director, Office of Nuclear Reactor Regulation.

AREVA NP Topical Report BAW-2308, Revisions 1-A and 2-A, provides NRC-approved alternate initial RT_{NDT} and associated uncertainty term (σ_I) values of the Linde 80 weld materials present in the extended beltline region of the RPV at Braidwood and Byron. BAW-2308, Revision 2-A, is a supplement to Revision 1-A, and incorporated additional test data and a re-evaluation of the reference temperature T_0, determination, as requested by the NRC in the Safety Evaluation (SE) for Revision 1-A of BAW-2308. As stated in the NRC SE for Revision 2-A of the topical report dated March 24, 2008, the Conditions and Limitation, Items (1) through (4), contained in the SE for Revision 1-A of the topical report must be satisfied in order for licensees to reference the topical report in specific licensing applications.

Conditions and Limitations Item (4) contained in the SE for Revision 1-A of BAW-2308 states:

> *Any licensee who wants to utilize the methodology of TR BAW-2308, Revision 1 as outlined in items (1) through (3) above, must request an exemption, per 10 CFR 50.12, from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61 to do so. As part of a licensee's exemption request, the NRC staff expects that the licensee will also submit information which demonstrates what values the licensee proposes to use for ΔRT_{NDT} and the margin term for each Linde 80 weld in its RPV through the end of its facility's current operating license.*

In the above quotation, Conditions and Limitations Item (1) pertains to NRC-accepted values of initial (unirradiated) reference temperature, IRT_{TO}, and the corresponding uncertainty term, σ_I, for Linde 80 weld materials based on the Master Curve methodology using direct testing of fracture toughness in accordance with American Society for Testing and Materials (ASTM) Standard Test Method E1921 (1997 and 2002 Editions).

Conditions and Limitations Item (2) requires that a minimum chemistry factor of 167.0°F be applied when the methodology of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," and 10 CFR 50.61 is used to assess the shift in nil-ductility transition temperature due to irradiation.

Conditions and Limitations Item (3) requires that a value of σ_{Δ} = 28.0°F be used to determine the margin term, as defined in BAW-2308, Revision 1-A, and RG 1.99, Revision 2.

Enclosures 1 and 2 contain the Westinghouse WCAP documents which support the above Conditions and Limitations, whose compliance will be discussed hereafter. Enclosure 1 is applicable to Braidwood Station, Units 1 and 2, Enclosure 2 is applicable to Byron Station, Units 1 and 2. Each enclosure is structured similarly such that text, table and figures are readily comparable.

PROPOSED EXEMPTION

The exemption requested by EGC addresses portions of the following regulations:

(1) 10 CFR 50, Appendix G, "Fracture Toughness Requirements," which sets forth 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 pressurized boundary may be subjected over its service lifetime; and

(2) 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," which sets forth fracture toughness requirements for protection against pressurized thermal shock (PTS).

The exemption from Appendix G to 10 CFR 50 is to replace the required use of the existing Charpy V-notch and drop-weight-based methodology with the use of an alternate methodology that incorporates the use of fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in Braidwood and Byron RPV extended beltline regions. The alternate methodology employs direct fracture toughness testing per the Master Curve methodology based on use of ASTM Standard Test Method E1921 (1997 and 2002 editions), and ASME Code Case N-629. The exemption is required since 10 CFR 50, Appendix G, requires that for the preservice or unirradiated condition, RT_{NDT} be evaluated by Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Section III, Paragraph NB-2331.

The exemption from 10 CFR 50.61 is to use an alternate methodology to allow the use of direct fracture toughness test data for evaluating the integrity of the Linde 80 weld materials present in the Braidwood and Byron RPV extended beltline regions, based on the use of the ASTM Standard Test Method E1921 (1997 and 2002 editions), and ASME Code Case N-629. The exemption is required because the methodology for evaluating RPV material fracture toughness in 10 CFR 50.61 requires that the pre-service or unirradiated condition be evaluated using by Charpy V-notch impact tests and drop weight tests according to the procedures in the ASME Code, Section III, Paragraph NB-2331.

Additionally, the NRC SE for Topical Report BAW-2308, Revision 1-A, concludes that an exemption is required to address issues related to 10 CFR 50.61 as the methodology presented in Topical Report BAW-2308, Revision 1-A, as modified and approved by the NRC staff, represents a significant change to the methodology specified in 10 CFR 50.61 for determining the Pressurized Thermal Shock (PTS) reference temperature (RT_{PTS}) value for Linde 80 weld material. The changes in the methodology described in BAW-2308, Revisions 1-A and 2-A, with respect to the methodology per 10 CFR 50.61, include the requirements for use of a minimum chemistry factor of 167°F and a value of $\sigma_\Delta = 28.0$°F for Linde 80 weld materials.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law; 2) the exemption will not result in an undue risk to public health and safety; 3) the exemption is consistent with the common defense and security; and 4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemption to allow the use of Topical Report BAW-2308, Revisions 1-A and 2-A (Revision 2-A is a supplement to Revision 1-A), as the basis for the Linde 80 weld material initial properties at Braidwood and Byron satisfy these requirements, as described below.

1. The requested exemption is authorized by law.

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

   In addition, 10 CFR 50.61 permits other methods for use in determining the initial material properties provided such methods are approved by the Director, Office of Nuclear Reactor Regulation.

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

   The proposed material initial properties basis described in Topical Report BAW-2308, Revisions 1-A and 2-A, represent an NRC-approved methodology for establishing weld wire specific and generic IRT_{TO} values for Linde 80 welds. Topical Report BAW-2308, Revisions 1-A and 2-A, includes appropriate conservatisms to ensure that use of the proposed initial material properties basis does not increase the probability of occurrence or the consequences of an accident at Braidwood or Byron and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety.

   The use of this proposed approach ensures that the intent of the requirements specified in 10 CFR 50, Appendix G, and 10 CFR 50.61, are satisfied.

   The requested exemption is consistent with the NRC requirements specified in the SE for the approved Topical Report BAW-2308, Revisions 1-A and 2-A; consequently, the exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security.

   The requested exemption is specifically concerned with RPV material properties and is consistent with NRC requirements specified in the SE (References 1 and 2) for the approved Topical Report BAW-2308, Revisions 1-A and 2-A. Consequently, the requested exemption will not endanger the common defense and security.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50, Appendix G, and 10 CFR 50.61.

   In accordance with 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii), since application of the methodology in Topical Report BAW-2308, Revisions 1-A and 2-A, in this particular circumstance serves the underlying purpose of the regulations.

   The underlying purpose of 10 CFR 50, Appendix G, and 10 CFR 50.61, is to protect the integrity of the reactor coolant pressure boundary by ensuring that each reactor vessel material has adequate fracture toughness. Application of ASME Code, Section III, Paragraph NB-2331, in the determination of initial material properties was conservatively developed based on the level of knowledge existing in the early 1970's concerning RPV materials and the estimated effects of operation.

   Since the early 1970's, the level of knowledge concerning these topics has greatly expanded. This increased knowledge level permits relaxation of the ASME Code, Section III, Paragraph NB-2331 requirements, via application of Topical Report BAW-2308, Revision 2-A, while maintaining the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety is maintained.

Enclosures 1 and 2 present the reactor vessel integrity assessments for Braidwood and Byron, respectively, utilizing the methodology of Topical Report BAW-2308, Revisions 1-A and 2-A for Linde 80 weld materials, specifically in the extended beltline region for the welds between the nozzle forgings and the RPV shell. The assessments document the integrity of the RPV's for Braidwood and Byron relative to the requirements and underlying purpose of 10 CFR 50.61 and 10 CFR 50, Appendix G. Appendix G of both enclosures demonstrates the applicability of the topical report to the Braidwood and Byron nozzle-to-shell welds, i.e. the specific location of the Linde 80 welds.

COMPLIANCE WITH "CONDITIONS AND LIMITATIONS" OF TOPICAL REPORT BAW-2308

The use of the Master Curve method, which is part of the basis of Topical Report BAW-2308, is introduced in a subsection of Section 3 of Enclosures 1 and 2. Compliance with the Conditions and Limitations of Topical Report BAW-2308, Revisions 1-A and 2-A is provided below.

Conditions and Limitations Item (1): Section 3 of Enclosures 1 and 2 describes the calculation of the fracture toughness properties. Referring to Tables 3-1 and 3-2 of those enclosures, a footnote to these tables highlights where Topical Report BAW-2308 was utilized; the Conditions and Limitations Item (1) was satisfied.

Conditions and Limitations Item (2): Section 5 of Enclosures 1 and 2 describes the calculation of the Chemistry Factors used in the analyses. Referring to Tables 5-4 and 5-5 of those

enclosures, a footnote to these tables highlights where Topical Report BAW-2308 was utilized (i.e. where the minimum chemistry factor of 167°F was employed); the Conditions and Limitations Item (2) was satisfied.

Conditions and Limitations Item (3): Section 7 of Enclosures 1 and 2 describes the calculation of the adjusted reference temperatures. Referring to Tables 7-3, 7-4, and 7-5 (for Unit 1 at Braidwood and Byron) of those enclosures, a footnote to those tables highlights how the overall margin term is calculated; this calculation is performed separately at the 1/4T location, 3/4T location, and surface location, respectively. Similarly, Tables 7-6, 7-7 and 7-8 document the calculation results for Unit 2 at Braidwood and Byron. These table illustrate that Conditions and Limitations Item (3) was satisfied.

Conditions and Limitations Item (4) is met by this exemption request submittal which contains the information (refer to those specific sections and tables of Enclosures 1 and 2 highlighted above) which demonstrates the values used for ΔRT_{NDT} and the margin term for each Linde 80 weld in the RPV.

Therefore, the intent of 10 CFR 50.61 and 10 CFR 50, Appendix G will continue to be satisfied for the proposed change in reactor vessel material initial properties basis, thus justifying the exemption request. Issuance of an exemption for the criteria of these regulations to permit the use of Topical Report BAW-2308, Revisions 1-A and 2-A for Braidwood and Byron will not compromise the safe operation of the reactors and will ensure that RPV integrity is maintained.

PRECEDENT

As further support for this requested exemption, EGC notes that relevant precedent exists for granting an exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The NRC has approved similar exemption requests for Oconee Nuclear Station, Three Mile Island Nuclear Station and Point Beach Nuclear Plant.

The exemption requests and NRC approvals are listed below.

1. NRC letter to Duke Energy Carolinas, LLC, dated April 26, 2012, "Oconee Nuclear Station, Units 1, 2, and 3, Exemption from the Requirements of 10 CFR Part 50.61 and 10 CFR Part 50, Appendix G (TAC NOS. ME7000, ME7001, ME7002, ME7003, ME7004, AND ME7005)," (ML120580196)

2. NRC letter to Exelon Generation Company, LLC, dated December 13, 2013, "Three Mile Island Nuclear Station, Unit 1 – Exemption from Certain Requirements of 10 CFR Part 50, Appendix G and 10 CFR 50.61, for Initial RTNDT Values for Linde 80 Welds (TAC NO. MF0425)." (ML13324A086)

3. NRC letter to NextEra Energy Point Beach, LLC, dated June 30, 2014, "Point Beach Nuclear Plant (Point Beach), Units 1 and 2 – Exemption from the Requirements of 10 CFR Section 50.61 and Appendix G to 10 CFR Par 50 (TAC NOS. MF0534 and MF0535)." (ML14126A612)

III. ENVIRONMENTAL ASSESSMENT

EGC has determined that the requested exemption meets the categorical exclusion provision in 10 CFR 51.22(c)(9), as an amendment to Braidwood and Byron's license under 10 CFR 50 requires the issuance of an exemption request, with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Therefore, in accordance with 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed exemption.

IV. CONCLUSION

This exemption request is associated with certain requirements of 10 CFR 50, Appendix G, and 10 CFR 50.61. This required exemption would allow use of an alternate method, the Master Curve Methodology, as described in AREVA NP Tropical Report BAW-2308, Revisions 1-A and 2-A, for determining the initial, unirradiated material reference temperatures of the Linde 80 weld materials present in the reactor pressure vessel extended beltline region of Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2. The Conditions and Limitations on the use of the topical report were satisfied.

As demonstrated above, we consider that this exemption request is in accordance with the criteria of 10 CFR 50.12. Specifically, the requested exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present as previously described.

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

V. REFERENCES

1) NRC letter to AREVA NP, dated August 4, 2005, Final Safety Evaluation for Topical Report BAW-2308, Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials," (TAC No. MB6636). (ML052070408)

2) NRC letter to Westinghouse Electric Company, dated March 24, 2008, Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2308, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials," (TAC No. MD4241). (ML080770349)

ENCLOSURE 1

WCAP-18370-NP, Revision 0

Braidwood Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

WCAP-18370-NP
Revision 0

June 2019

Braidwood Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-18370-NP
Revision 0

Braidwood Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

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RECORD OF REVISION

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the Braidwood Units 1 and 2 reactor vessels. The P-T limit curves were generated using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G. This P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Note that the 4th ISI interval ASME Section XI Code of Record for Braidwood is the 2013 Edition. The requirements of Appendix G of the 2013 Edition of the Code are equivalent to those from the 1998 Edition through 2000 Addenda edition used herein. The heatup and cooldown P-T limit curves utilize the Adjusted Reference Temperature (ART) values for Braidwood Units 1 and 2 calculated using Regulatory Guide 1.99, Revision 2. The limiting ART values in material with a postulated axial flaw were those of the Braidwood Unit 2 Nozzle Shell Forging 5P-7056 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. The limiting ART values in material with a postulated circumferential flaw were those of the Braidwood Unit 1 Intermediate to Lower Shell Weld at both 1/4T and 3/4T locations. The axial oriented flaw cases are limiting; therefore, only the axially oriented flaw curves are presented in this report.

The P-T limit curves were generated for 57 effective full-power years (EFPY) using a heatup rate of 100°F/hr, and cooldown rates of 0° (steady-state), 25°, 50°, and 100°F/hr and are applicable to both Braidwood Unit 1 and Unit 2. The curves were developed without the flange requirements of 10 CFR 50, Appendix G, as justified by WCAP-16143-P and approved in [26]. The NRC originally approved the exemption from the Appendix G flange requirements in [26] based on Revision 0 of WCAP-16143-P. Reference [44] revised the exemption to account for Revision 1 of WCAP-16143-P, which considers a 53 stud configuration. The curves were also developed without margins for instrumentation errors. The curves can be found in Figures 8-1 and 8-2.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 57 EFPY.

Appendix B contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles to adhere to the requirements of Regulatory Issue Summary (RIS) 2014-11 [14]. As discussed in Appendix B, the P-T limit curves generated based on cylindrical beltline material (Nozzle Shell Forging 5P-7056) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Braidwood Units 1 and 2 at 57 EFPY.

Appendix C contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix C, all of the other ferritic RCPB components meet the applicable requirements of Section III of the ASME Code.

Appendix D contains the credibility evaluation of the Braidwood Units 1 and 2 reactor vessels surveillance data per the requirements of Regulatory Guide 1.99, Revision 2. Braidwood Units 1 and 2 fluence values and ex-vessel neutron dosimetry (EVND), described in Section 2.0, were used to complete the evaluation.

Appendix E contains a pressurized thermal shock (PTS) evaluation for all of the Braidwood Units 1 and 2 reactor vessel beltline and extended beltline materials. Per Appendix E, all beltline and extended beltline materials have projected RT_{PTS} values below the screening criteria set forth in 10 CFR 50.61 at 57 EFPY.

Appendix F provides the validation of the radiation transport calculation models based on neutron dosimetry measurement.

Appendix G demonstrates the applicability of BAW-2308 results to the Braidwood Units 1 and 2 nozzle-to-shell welds as utilized herein.

1 INTRODUCTION

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) of the beltline region material of the reactor vessel. The adjusted RT_{NDT} 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 (T_{NDT}) 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} ($RT_{NDT(U)}$). 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 U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + 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 heatup and cooldown P-T limit curves documented in this report were generated using the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [2]. Specifically, the K_{Ic} methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, Appendix G [3] was used. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors.

The purpose of this report is to present the calculations and the development of the Braidwood Units 1 and 2 heatup and cooldown P-T limit curves for 57 EFPY. This report documents the calculated ART values and the development of the P-T limit curves for normal operation. The calculated ART values for 57 EFPY are documented in Section 7 of this report. The fluence projections used in calculation of the ART values are provided in Section 2 of this report.

The P-T limit curves herein were generated without instrumentation errors and are applicable to both Braidwood Unit 1 and Unit 2. The reactor vessel flange requirements of 10 CFR 50, Appendix G [4] have not been incorporated in the P-T limit curves. As part of the P-T limit curve development, the initial RT_{NDT} for Units 1 and 2 inlet and outlet nozzles to upper shell forging welds were redefined to take advantage of the "Master Curve" method. The use of the Master Curve is a departure from the ASME Code, Section III Subsection NB-2300 method required by 10 CFR 50, Appendix G; therefore, it requires the submittal and NRC approval of a 10 CFR 50.12 exemption for use. Additional details about this method are defined in Section 3, and a justification for its use is contained in Appendix G. Per [5] and [26], the flange requirements have been eliminated from Braidwood Units 1 and 2. As discussed in Appendix B, the P-T limit curves generated in Section 8 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Braidwood Units 1 and 2 at 57 EFPY. Discussion of the other reactor coolant pressure boundary (RCPB) ferritic components relative to P-T limits is contained in Appendix C.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

Two discrete ordinates (S_N) transport analyses were performed for the Braidwood Units 1 and 2 reactors, respectively, to determine the neutron radiation environment within the reactor pressure vessels. In these analyses, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. The dosimetry analysis documented in WCAP-18245-NP [7] and WCAP-18345-NP [8] showed that the $\pm 20\%$ (1σ) acceptance criteria specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [9], is met, based on comparison with the four in-vessel surveillance capsules and twelve Ex-Vessel Neutron Dosimetry (EVND) capsules tested to-date for each unit from Braidwood Unit 1 and Unit 2. More details on compliance with Regulatory Guide 1.190 are contained in Appendix F. These validated calculations form the basis for providing projections of the neutron exposure of the reactor pressure vessel through the end of license extension (EOLE).

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2].

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the Braidwood Units 1 and 2 reactor vessels, a series of fuel-cycle-specific forward transport calculations were performed using the following two-dimensional/one-dimensional synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) \times \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron fluence rate distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at Braidwood Units 1 and 2, respectively.

All of the transport calculations were carried out using a two-dimensional discrete ordinates code, DORT [10], and the BUGLE-96 cross-section library [11]. The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions were treated on a fuel-cycle-specific basis.

The r, θ models used for this analysis are depicted in Figure 2-1, Figure 2-2, and Figure 2-3. In each of these figures, a single octant is depicted showing the arrangement of neutron pads and surveillance capsules, as applicable. In regard to these three geometries, the maximum exposure of the pressure vessel occurs in octants with the 12.5 degree neutron pad span where no surveillance capsules are present. The surveillance capsules are modeled in the octants where the neutron pads span 20.0 degrees and 22.5 degrees, whereas EVND is located in an octant with the 12.5 degrees neutron pad span.

In addition to the core, reactor internals, pressure vessel, and primary biological shield, the transport models developed for these octant geometries included explicit representations of the surveillance capsules, the pressure vessel cladding, the insulation located external to the pressure vessel. The reactor vessel insulation is modeled as a mixture of stainless steel and air, with majority of the volume being air. The reactor vessel insulation can be seen in Figure 2-1 and Figure 2-2 as the brown donut shape band outside of the reactor vessel wall, which is the blue color donut shape band.

From a neutronic standpoint, the inclusion of the surveillance capsules and associated support structure in the analytical model is significant. Because the presence of the capsules and structure has a marked effect on the magnitude of the neutron flux, as well as on the relative neutron and gamma-ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are properly accounted for in the analysis. The ex-core detector wells located in the concrete biological shielding of Braidwood Unit 1 and Unit 2 were also incorporated into the r-θ models, because past experience has shown that these voids can significantly affect EVND dosimetry results. The detector wells do not affect the in-vessel surveillance capsule results.

In contrast to the relatively massive stainless steel and carbon steel structures associated with the in-vessel surveillance capsules, the small aluminum capsules used in the EVND program were designed to minimize perturbations in the neutron flux, and, thus, to provide essentially free-field data at the measurement locations. Therefore, specific modeling of these small capsules was not required.

The r, θ analytical models of the reactor geometry shown in Figure 2-1, Figure 2-2, and Figure 2-3 employed nominal design dimensions for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The r, θ geometric mesh description of the reactor model shown in Figure 2-1 consisted of 257 radial by 131 azimuthal intervals, whereas the reactor models shown in Figure 2-2 and Figure 2-3 consisted of 255 radial by 143 azimuthal intervals. Differences in the number of radial and azimuthal intervals between neutron pad configurations can be attributed to extra mesh refinement in the region of the surveillance capsules. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r, θ calculations was set at a value of 0.001.

A section view of the r, z model of the Braidwood Units 1 and 2 reactors is shown in Figure 2-4. The model extended radially from the centerline of the reactor core out to a location interior to the primary

biological shield and over an axial span from an elevation approximately six feet below the active fuel to approximately five feet above the active fuel. As in the case of the r, θ model, nominal design dimensions and full-power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The volume fractions utilized for the fuel region, the bypass region, the downcomer region, and the reactor pressure vessel (RPV) insulation region were consistent with the equivalent regions in the r, θ model.

The r, z geometric mesh description of the reactor model shown in Figure 2-4 consisted of 153 radial by 216 axial intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration convergence criterion utilized in the r, z calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 153 radial mesh intervals included in the r, z model. Thus, radial synthesis factors could easily be determined on a mesh-wise basis throughout the entire geometry.

The core power distributions used in the plant-specific transport analysis for the Braidwood Unit 1 and Unit 2 reactors were taken from nuclear design documentation. The data extracted included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial power distributions were also obtained.

For each fuel cycle of operation, the fuel assembly-specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope-dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the spatial mesh arrays used in the discrete ordinates calculations.

In Table 2-1, axial and azimuthal locations of the Braidwood Unit 1 and Unit 2 pressure vessel materials are provided. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the midplane of the active fuel stack.

Cycle-specific calculations were performed for Cycles 1 through 19 for both Braidwood Unit 1 and Unit 2, with core thermal powers given in Table 2-2 and Table 2-3, respectively. Please note Cycle 20 fluence data were not included for both units because they were completed under a separate project and were not finished until after the generation of the P-T limits curves within this report.

Neutron fluence rate and fluence are given in Table 2-4, Table 2-5, and Table 2-8 for Braidwood Unit 1 and in Table 2-10, Table 2-11, and Table 2-14 for Braidwood Unit 2. Similarly, iron atom displacement rate and iron atom displacements are provided in Table 2-6, Table 2-7, and Table 2-9 for Braidwood Unit1, and Table 2-12, Table 2-13, and Table 2-15 for Braidwood Unit 2. The data presented represent the maximum neutron exposure experienced by RPV materials that will constitute inputs to the reactor vessel integrity analysis. The reported data consider both the inner and outer radius of the RPV base metal, and

account for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for each operating cycle of the Braidwood Unit 1 and Unit 2 reactors. Note that, for any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

In Table 2-4 through Table 2-15, calculated exposure values are projected to 32, 48, 54, 57, and 60 EFPY. Projections were based on the average of Cycles 17, 18 and 19 spatial power distributions and reactor operating conditions with a rated core power of 3658 MWt, accounting for calorimetric uncertainty. The projected results will remain valid as long as future plant operation is consistent with these assumptions.

Results of the discrete ordinates transport analyses pertinent to the Braidwood Unit 1 surveillance capsule evaluations are provided in Table 2-16 through Table 2-18. In Table 2-16, the calculated fast neutron fluence rate and fluence (E > 1.0 MeV) are provided at the geometric center of capsules and at core midplane, as a function of irradiation time for the Braidwood Unit 1 reactor. Similar data presented in terms of iron atom displacement rate (dpa/s) and integrated iron atom displacements (dpa) are given in Table 2-17.

In Table 2-18, lead factors associated with surveillance capsules are provided as a function of operating time for the Braidwood Unit 1 reactor. The lead factor is defined as the ratio of the neutron fluence (E > 1.0 MeV) at the geometric center of the surveillance capsule to the maximum neutron fluence (E > 1.0 MeV) at the pressure vessel clad/base metal interface.

All surveillance capsules have been removed from Braidwood Unit 1, so fluence data at the surveillance positions beyond Cycle 14 is unnecessary (because there are no capsules receiving any fluence). However, if any capsules were to be re-inserted, it would be necessary to know the fast fluence rate at the surveillance capsule holder positions. To allow determination of potential fast fluence accumulation, projected fast fluence rate (E > 1.0 MeV) at each surveillance capsule location is provided in Table 2-19. Projections of future operation are based on an average of Cycles 17, 18 and 19. The additional fast fluence accumulated for any re-inserted capsule can be determined by multiplying the fast fluence rate value in Table 2-19 for the appropriate capsule position times the irradiation duration in effective full-power seconds (EFPS).

Results of the discrete ordinates transport analyses pertinent to the Braidwood Unit 2 surveillance capsule evaluations are provided in Table 2-20 through Table 2-22. In Table 2-20, the calculated fast neutron fluence rate and fluence (E > 1.0 MeV) are provided at the geometric center of capsules and at core midplane, as a function of irradiation time for the Braidwood Unit 2 reactor. Similar data presented in terms of iron atom displacement rate (dpa/s) and integrated iron atom displacements (dpa) are given in Table 2-21.

In Table 2-22, lead factors associated with surveillance capsules are provided as a function of operating time for the Braidwood Unit 2 reactor. The lead factor is defined as the ratio of the neutron fluence (E > 1.0 MeV) at the geometric center of the surveillance capsule to the maximum neutron fluence (E > 1.0 MeV) at the pressure vessel clad/base metal interface.

All surveillance capsules have been removed from Braidwood Unit 2, so fluence data at the surveillance positions beyond Cycle 14 is unnecessary (because there are no capsules receiving any fluence). However, if any capsules were to be re-inserted, it would be necessary to know the fast fluence rate at the surveillance capsule holder positions. To allow determination of potential fast fluence accumulation, projected fast fluence rate ($E > 1.0$ MeV) at each surveillance capsule location is provided in Table 2-23. Projections of future operation are based on an average of Cycles 17, 18, and 19. The additional fast fluence accumulated for any re-inserted capsule can be determined by multiplying the fast fluence rate value in Table 2-23 for the appropriate capsule position times the irradiation duration in EFPS.

2.3 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Braidwood Unit 1 and Unit 2 pressure vessels is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology used in the Braidwood Unit 1 and Unit 2 reactor pressure vessels neutron exposure evaluations was carried out in the following four stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator (NUREG/CR-6454, "Pool Critical Assembly Pressure Vessel Facility Benchmark" [12]) at the Oak Ridge National Laboratory (ORNL).

2. Comparison of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment (NUREG/CR-6453, "H.B. Robinson-2 Pressure Vessel Benchmark" [13]).

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments.

4. Comparison of the calculations with all available dosimetry results from measurement programs carried out at the Braidwood Units 1 and 2 reactors.

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross sections. This phase, however, did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational and geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations as well as to a lack of knowledge relative to various plant-specific parameters. The overall calculational uncertainty applicable to the Braidwood Units 1 and 2 analyses were established from the results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Braidwood Units 1 and 2 measurements) was used solely to demonstrate the adequacy of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used to bias the final results in any way.

Table 2-24 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in WCAP-14040-A. The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in WCAP-18245-NP and WCAP-18345-NP [7 and 8] support these uncertainty assessments for Braidwood Unit 1 and Unit 2.

Table 2-1 Braidwood Unit 1 and Unit 2 RPV Material Locations

Material	Axial Location[a] [cm]	Azimuthal Location[b] [degrees]
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[c]		
Nozzle 1	270.12	22
Nozzle 2	270.12	158
Nozzle 3	270.12	202
Nozzle 4	270.12	338
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent[c]		
Nozzle 1	265.99	67
Nozzle 2	265.99	113
Nozzle 3	265.99	247
Nozzle 4	265.99	293
Nozzle Shell Forging – Lowest Extent	187.29	0 to 360
Nozzle Shell to Intermediate Shell Circumferential Weld	187.29	0 to 360
Intermediate Shell Forging	-61.15 to 187.29	0 to 360
Intermediate Shell to Lower Shell Circumferential Weld	-61.15	0 to 360
Lower Shell Forging	-309.59 to -61.15	0 to 360
Lower Shell to Lower Vessel Head Circumferential Weld	-309.59	0 to 360

Notes:

(a) Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

(b) Azimuthal locations are indexed to θ = 0.0 as shown on reactor vessel general assembly drawings.

(c) No credit is taken for the azimuthal location of the nozzle welds – the azimuthal angle providing the maximum exposure is reported

Table 2-2 Reactor Core Power Level – Braidwood Unit 1

Cycle	Core Power [MWt]
1	3411.0
2	3411.0
3	3411.0
4	3411.0
5	3411.0
6	3411.0
7	3411.0
8	3411.0
9	3458.0
10	3586.6
11	3586.6
12	3586.6
13	3586.6
14	3586.6
15	3586.6
16	3586.6
17	3586.6
18	3630.3[a]
19	3658.0[b]

Notes:

(a) With a mid-cycle uprate from 3586.6 MWt to 3645 MWt on February 14, 2014.

(b) The uprate to 3645 MWt is modeled as 3658 MWt to account for calorimetric uncertainty.

Table 2-3 Reactor Core Power Level – Braidwood Unit 2

Cycle	Core Power [MWt]
1	3411.0
2	3411.0
3	3411.0
4	3411.0
5	3411.0
6	3411.0
7	3411.0
8	3411.0
9	3528.0
10	3586.6
11	3586.6
12	3586.6
13	3586.6
14	3586.6
15	3586.6
16	3586.6
17	3597.2[(a)]
18	3658.0[(b)]
19	3658.0[(b)]

Notes:

(a) With a mid-cycle uprate from 3586 6 MWt to 3645 MWt on February 12, 2014.

(b) The uprate to 3645 MWt is modeled as 3658 MWt to account for calorimetric uncertainty.

Table 2-4 Calculated Maximum Fast Neutron Fluence Rate (E > 1.0 MeV) at the Braidwood Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.16	1.16	1.30E+10	2.10E+10	2.37E+10	2.64E+10	2.64E+10	45°
2	0.86	2.02	1.19E+10	1.67E+10	1.65E+10	1.83E+10	1.83E+10	45°
3	1.13	3.15	9.73E+09	1.58E+10	1.93E+10	2.28E+10	2.28E+10	45°
4	1.15	4.30	9.96E+09	1.53E+10	1.68E+10	1.73E+10	1.80E+10	22°
5	1.22	5.52	9.99E+09	1.51E+10	1.74E+10	1.98E+10	1.98E+10	45°
6	1.04	6.55	1.06E+10	1.60E+10	1.67E+10	1.53E+10	1.86E+10	22°
7	1.24	7.79	9.25E+09	1.36E+10	1.76E+10	2.04E+10	2.04E+10	45°
8	1.29	9.08	7.35E+09	1.12E+10	1.31E+10	1.38E+10	1.38E+10	45°
9	1.44	10.52	8.62E+09	1.22E+10	1.39E+10	1.65E+10	1.65E+10	45°
10	1.49	12.01	1.04E+10	1.53E+10	1.63E+10	1.49E+10	1.77E+10	22°
11	1.42	13.43	1.01E+10	1.47E+10	1.59E+10	1.46E+10	1.70E+10	22°
12	1.45	14.88	1.00E+10	1.46E+10	1.58E+10	1.45E+10	1.68E+10	22°
13	1.40	16.28	9.84E+09	1.38E+10	1.43E+10	1.43E+10	1.53E+10	22°
14	1.41	17.69	1.03E+10	1.45E+10	1.57E+10	1.51E+10	1.66E+10	22°
15	1.43	19.11	9.68E+09	1.47E+10	1.62E+10	1.58E+10	1.72E+10	22°
16	1.43	20.55	9.88E+09	1.55E+10	1.68E+10	1.55E+10	1.82E+10	22°
17	1.30	21.84	1.07E+10	1.56E+10	1.66E+10	1.56E+10	1.78E+10	22°
18	1.49	23.33	1.00E+10	1.59E+10	1.73E+10	1.60E+10	1.88E+10	22°
19	1.43	24.76	1.04E+10	1.61E+10	1.80E+10	1.65E+10	1.92E+10	23°

Table 2-5 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Braidwood Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.16	1.16	4.77E+17	7.70E+17	8.70E+17	9.67E+17	9.67E+17	45°
2	0.86	2.02	8.00E+17	1.22E+18	1.32E+18	1.46E+18	1.46E+18	45°
3	1.13	3.15	1.14E+18	1.77E+18	1.99E+18	2.26E+18	2.26E+18	45°
4	1.15	4.30	1.50E+18	2.33E+18	2.60E+18	2.88E+18	2.88E+18	45°
5	1.22	5.52	1.88E+18	2.90E+18	3.26E+18	3.64E+18	3.64E+18	45°
6	1.04	6.55	2.22E+18	3.41E+18	3.80E+18	4.13E+18	4.13E+18	45°
7	1.24	7.79	2.56E+18	3.92E+18	4.45E+18	4.89E+18	4.89E+18	45°
8	1.29	9.08	2.86E+18	4.37E+18	4.98E+18	5.44E+18	5.44E+18	45°
9	1.44	10.52	3.23E+18	4.89E+18	5.58E+18	6.16E+18	6.16E+18	45°
10	1.49	12.01	3.69E+18	5.57E+18	6.29E+18	6.81E+18	6.81E+18	45°
11	1.42	13.43	4.14E+18	6.23E+18	7.01E+18	7.47E+18	7.47E+18	45°
12	1.45	14.88	4.60E+18	6.90E+18	7.73E+18	8.13E+18	8.13E+18	45°
13	1.40	16.28	5.03E+18	7.50E+18	8.36E+18	8.76E+18	8.76E+18	45°
14	1.41	17.69	5.49E+18	8.15E+18	9.06E+18	9.43E+18	9.47E+18	22°
15	1.43	19.11	5.93E+18	8.80E+18	9.78E+18	1.01E+19	1.02E+19	22°
16	1.43	20.55	6.37E+18	9.49E+18	1.05E+19	1.08E+19	1.10E+19	22°
17	1.30	21.84	6.80E+18	1.01E+19	1.12E+19	1.15E+19	1.18E+19	22°
18	1.49	23.33	7.26E+18	1.09E+19	1.20E+19	1.22E+19	1.26E+19	22°
19(a)	1.43	24.76	7.72E+18	1.16E+19	1.28E+19	1.29E+19	1.35E+19	22°
		28.00	8.77E+18	1.32E+19	1.45E+19	1.46E+19	1.54E+19	22°
		32.00	1.01E+19	1.52E+19	1.67E+19	1.66E+19	1.77E+19	22°
		36.00	1.13E+19	1.71E+19	1.89E+19	1.86E+19	2.00E+19	22°
		40.00	1.26E+19	1.91E+19	2.10E+19	2.06E+19	2.23E+19	22°
		44.00	1.39E+19	2.11E+19	2.32E+19	2.26E+19	2.46E+19	22°
		48.00	1.52E+19	2.31E+19	2.53E+19	2.46E+19	2.70E+19	22°
		52.00	1.65E+19	2.51E+19	2.75E+19	2.66E+19	2.93E+19	22°
		54.00	1.72E+19	2.60E+19	2.85E+19	2.76E+19	3.04E+19	22°
		57.00	1.81E+19	2.75E+19	3.02E+19	2.91E+19	3.22E+19	22°
		60.00	1.91E+19	2.90E+19	3.18E+19	3.06E+19	3.39E+19	22°

Note:
(a) Values beyond EOC 19 are projected.

Table 2-6 Calculated Maximum Iron Atom Displacement Rate at the Braidwood Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.16	1.16	2.03E-11	3.23E-11	3.67E-11	4.18E-11	4.18E-11	45°
2	0.86	2.02	1.85E-11	2.57E-11	2.55E-11	2.89E-11	2.89E-11	45°
3	1.13	3.15	1.52E-11	2.44E-11	2.99E-11	3.61E-11	3.61E-11	45°
4	1.15	4.30	1.55E-11	2.36E-11	2.59E-11	2.74E-11	2.76E-11	22°
5	1.22	5.52	1.55E-11	2.32E-11	2.69E-11	3.13E-11	3.13E-11	45°
6	1.04	6.55	1.64E-11	2.45E-11	2.57E-11	2.43E-11	2.85E-11	22°
7	1.24	7.79	1.44E-11	2.10E-11	2.73E-11	3.22E-11	3.22E-11	45°
8	1.29	9.08	1.15E-11	1.73E-11	2.02E-11	2.18E-11	2.18E-11	45°
9	1.44	10.52	1.34E-11	1.87E-11	2.14E-11	2.62E-11	2.62E-11	45°
10	1.49	12.01	1.62E-11	2.36E-11	2.51E-11	2.35E-11	2.71E-11	22°
11	1.42	13.43	1.56E-11	2.27E-11	2.45E-11	2.31E-11	2.60E-11	22°
12	1.45	14.88	1.56E-11	2.25E-11	2.43E-11	2.29E-11	2.57E-11	22°
13	1.40	16.28	1.53E-11	2.12E-11	2.22E-11	2.26E-11	2.34E-11	22°
14	1.41	17.69	1.61E-11	2.24E-11	2.42E-11	2.39E-11	2.55E-11	22°
15	1.43	19.11	1.51E-11	2.26E-11	2.50E-11	2.51E-11	2.63E-11	22°
16	1.43	20.55	1.54E-11	2.38E-11	2.60E-11	2.45E-11	2.79E-11	22°
17	1.30	21.84	1.66E-11	2.40E-11	2.56E-11	2.47E-11	2.73E-11	22°
18	1.49	23.33	1.56E-11	2.45E-11	2.66E-11	2.54E-11	2.88E-11	22°
19	1.43	24.76	1.61E-11	2.48E-11	2.77E-11	2.62E-11	2.94E-11	23°

Table 2-7 Calculated Maximum Iron Atom Displacements at the Braidwood Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacements (dpa)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.16	1.16	7.42E-04	1.18E-03	1.34E-03	1.53E-03	1.53E-03	45°
2	0.86	2.02	1.24E-03	1.88E-03	2.04E-03	2.32E-03	2.32E-03	45°
3	1.13	3.15	1.77E-03	2.73E-03	3.08E-03	3.58E-03	3.58E-03	45°
4	1.15	4.30	2.33E-03	3.58E-03	4.01E-03	4.56E-03	4.56E-03	45°
5	1.22	5.52	2.93E-03	4.47E-03	5.05E-03	5.76E-03	5.76E-03	45°
6	1.04	6.55	3.45E-03	5.25E-03	5.86E-03	6.53E-03	6.53E-03	45°
7	1.24	7.79	3.98E-03	6.03E-03	6.88E-03	7.74E-03	7.74E-03	45°
8	1.29	9.08	4.45E-03	6.73E-03	7.70E-03	8.61E-03	8.61E-03	45°
9	1.44	10.52	5.02E-03	7.54E-03	8.62E-03	9.75E-03	9.75E-03	45°
10	1.49	12.01	5.74E-03	8.58E-03	9.73E-03	1.08E-02	1.08E-02	45°
11	1.42	13.43	6.44E-03	9.59E-03	1.08E-02	1.18E-02	1.18E-02	45°
12	1.45	14.88	7.16E-03	1.06E-02	1.19E-02	1.29E-02	1.29E-02	45°
13	1.40	16.28	7.83E-03	1.16E-02	1.29E-02	1.39E-02	1.39E-02	45°
14	1.41	17.69	8.55E-03	1.26E-02	1.40E-02	1.49E-02	1.49E-02	45°
15	1.43	19.11	9.22E-03	1.36E-02	1.51E-02	1.61E-02	1.61E-02	45°
16	1.43	20.55	9.90E-03	1.46E-02	1.63E-02	1.71E-02	1.71E-02	45°
17	1.30	21.84	1.06E-02	1.56E-02	1.73E-02	1.81E-02	1.81E-02	45°
18	1.49	23.33	1.13E-02	1.67E-02	1.85E-02	1.93E-02	1.94E-02	22°
19(a)	1.43	24.76	1.20E-02	1.78E-02	1.98E-02	2.05E-02	2.07E-02	22°
		28.00	1.36E-02	2.03E-02	2.25E-02	2.31E-02	2.36E-02	22°
		32.00	1.57E-02	2.34E-02	2.58E-02	2.62E-02	2.71E-02	22°
		36.00	1.77E-02	2.64E-02	2.91E-02	2.94E-02	3.07E-02	22°
		40.00	1.97E-02	2.95E-02	3.25E-02	3.26E-02	3.42E-02	22°
		44.00	2.17E-02	3.25E-02	3.58E-02	3.57E-02	3.78E-02	22°
		48.00	2.37E-02	3.56E-02	3.91E-02	3.89E-02	4.14E-02	22°
		52.00	2.57E-02	3.86E-02	4.24E-02	4.21E-02	4.49E-02	22°
		54.00	2.67E-02	4.01E-02	4.41E-02	4.37E-02	4.67E-02	22°
		57.00	2.82E-02	4.24E-02	4.66E-02	4.61E-02	4.94E-02	22°
		60.00	2.97E-02	4.47E-02	4.91E-02	4.84E-02	5.20E-02	22°

Note:

(a) Values beyond EOC 19 are projected.

Table 2-8 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Braidwood Unit 1 Pressure Vessel Welds and Shells

Material	Fast Neutron Fluence (n/cm^2)			
	24.76 EFPY	28 EFPY	32 EFPY	36 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	3.82E+16	4.38E+16	5.07E+16	5.76E+16
Inlet Nozzle Forging to Vessel Shell Welds	5.06E+16	5.80E+16	6.71E+16	7.63E+16
Nozzle Shell Forging[(a)]	4.65E+18	5.32E+18	6.14E+18	6.96E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	4.65E+18	5.32E+18	6.14E+18	6.96E+18
Intermediate Shell Forging	1.35E+19	1.54E+19	1.77E+19	2.00E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.29E+19	1.47E+19	1.69E+19	1.91E+19
Lower Shell Forging	1.32E+19	1.51E+19	1.73E+19	1.96E+19
Lower Shell to Lower Vessel Head Circumferential Weld	5.95E+15	6.77E+15	7.79E+15	8.81E+15

Material	Fast Neutron Fluence (n/cm^2)			
	48 EFPY	54 EFPY	57 EFPY	60 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	7.83E+16	8.86E+16	9.38E+16	9.90E+16
Inlet Nozzle Forging to Vessel Shell Welds	1.04E+17	1.17E+17	1.24E+17	1.31E+17
Nozzle Shell Forging[(a)]	9.42E+18	1.06E+19	1.13E+19	1.19E+19
Nozzle Shell to Intermediate Shell Circumferential Weld	9.42E+18	1.06E+19	1.13E+19	1.19E+19
Intermediate Shell Forging	2.70E+19	3.04E+19	3.22E+19	3.39E+19
Intermediate Shell to Lower Shell Circumferential Weld	2.56E+19	2.89E+19	3.06E+19	3.22E+19
Lower Shell Forging	2.64E+19	2.97E+19	3.14E+19	3.31E+19
Lower Shell to Lower Vessel Head Circumferential Weld	1.19E+16	1.34E+16	1.41E+16	1.49E+16

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld

Table 2-9 Calculated Maximum Iron Atom Displacements at the Braidwood Unit 1 Pressure Vessel Welds and Shells

Material	Displacements per atom (dpa)			
	24.76 EFPY	28 EFPY	32 EFPY	36 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	9.72E-05	1.10E-04	1.26E-04	1.43E-04
Inlet Nozzle Forging to Vessel Shell Welds	1.08E-04	1.22E-04	1.40E-04	1.59E-04
Nozzle Shell Forging[a]	7.13E-03	8.15E-03	9.40E-03	1.07E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	7.13E-03	8.15E-03	9.40E-03	1.07E-02
Intermediate Shell Forging	2.07E-02	2.36E-02	2.71E-02	3.07E-02
Intermediate Shell to Lower Shell Circumferential Weld	1.98E-02	2.26E-02	2.59E-02	2.93E-02
Lower Shell Forging	2.03E-02	2.31E-02	2.66E-02	3.00E-02
Lower Shell to Lower Vessel Head Circumferential Weld	3.77E-05	4.25E-05	4.85E-05	5.47E-05

Material	Displacements per atom (dpa)			
	48 EFPY	54 EFPY	57 EFPY	60 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.93E-04	2.18E-04	2.31E-04	2.43E-04
Inlet Nozzle Forging to Vessel Shell Welds	2.15E-04	2.42E-04	2.56E-04	2.70E-04
Nozzle Shell Forging[a]	1.44E-02	1.63E-02	1.72E-02	1.82E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	1.44E-02	1.63E-02	1.72E-02	1.82E-02
Intermediate Shell Forging	4.14E-02	4.67E-02	4.94E-02	5.20E-02
Intermediate Shell to Lower Shell Circumferential Weld	3.94E-02	4.45E-02	4.70E-02	4.95E-02
Lower Shell Forging	4.04E-02	4.56E-02	4.82E-02	5.08E-02
Lower Shell to Lower Vessel Head Circumferential Weld	7.35E-05	8.29E-05	8.76E-05	9.23E-05

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld

Table 2-10 Calculated Maximum Fast Neutron Fluence Rate (E > 1.0 MeV) at the Braidwood Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	1.27E+10	2.04E+10	2.28E+10	2.55E+10	2.55E+10	45°
2	1.12	2.30	1.01E+10	1.48E+10	1.79E+10	2.11E+10	2.11E+10	45°
3	1.12	3.42	1.02E+10	1.54E+10	1.75E+10	1.94E+10	1.94E+10	45°
4A[(a)]	0.82	4.24	1.04E+10	1.55E+10	1.72E+10	1.90E+10	1.90E+10	45°
4B	0.34	4.58	1.04E+10	1.56E+10	1.74E+10	1.93E+10	1.93E+10	45°
5	1.27	5.85	9.69E+09	1.43E+10	1.58E+10	1.68E+10	1.68E+10	45°
6	1.34	7.19	9.38E+09	1.42E+10	1.59E+10	1.72E+10	1.72E+10	45°
7	1.37	8.56	8.00E+09	1.41E+10	1.63E+10	1.47E+10	1.75E+10	23°
8	1.40	9.96	9.16E+09	1.38E+10	1.47E+10	1.30E+10	1.58E+10	22°
9	1.36	11.33	9.08E+09	1.21E+10	1.32E+10	1.35E+10	1.36E+10	22°
10	1.45	12.78	8.25E+09	1.12E+10	1.27E+10	1.28E+10	1.30E+10	27°
11	1.38	14.15	9.34E+09	1.25E+10	1.37E+10	1.41E+10	1.41E+10	45°
12	1.44	15.60	9.04E+09	1.38E+10	1.53E+10	1.46E+10	1.62E+10	22°
13	1.45	17.04	8.80E+09	1.35E+10	1.49E+10	1.35E+10	1.57E+10	22°
14	1.37	18.41	9.80E+09	1.42E+10	1.58E+10	1.51E+10	1.64E+10	23°
15	1.43	19.85	9.50E+09	1.47E+10	1.61E+10	1.51E+10	1.73E+10	22°
16	1.41	21.26	9.92E+09	1.51E+10	1.63E+10	1.50E+10	1.76E+10	22°
17	1.45	22.71	9.48E+09	1.49E+10	1.62E+10	1.49E+10	1.76E+10	22°
18	1.36	24.07	1.06E+10	1.46E+10	1.52E+10	1.56E+10	1.61E+10	22°
19	1.49	25.56	9.71E+09	1.48E+10	1.67E+10	1.57E+10	1.76E+10	23°

Notes.

(a) 4A indicates time at which Capsule X was withdrawn

Table 2-11 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Braidwood Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm²)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	4.71E+17	7.56E+17	8.48E+17	9.47E+17	9.47E+17	45°
2	1.12	2.30	8.17E+17	1.26E+18	1.46E+18	1.67E+18	1.67E+18	45°
3	1.12	3.42	1.18E+18	1.80E+18	2.07E+18	2.35E+18	2.35E+18	45°
4A[(a)]	0.82	4.24	1.45E+18	2.21E+18	2.52E+18	2.84E+18	2.84E+18	45°
4B	0.34	4.58	1.56E+18	2.37E+18	2.70E+18	3.05E+18	3.05E+18	45°
5	1.27	5.85	1.94E+18	2.94E+18	3.33E+18	3.72E+18	3.72E+18	45°
6	1.34	7.19	2.34E+18	3.54E+18	4.00E+18	4.45E+18	4.45E+18	45°
7	1.37	8.56	2.69E+18	4.15E+18	4.71E+18	5.09E+18	5.09E+18	45°
8	1.40	9.96	3.09E+18	4.76E+18	5.36E+18	5.66E+18	5.66E+18	45°
9	1.36	11.33	3.48E+18	5.28E+18	5.92E+18	6.23E+18	6.23E+18	45°
10	1.45	12.78	3.85E+18	5.79E+18	6.50E+18	6.82E+18	6.82E+18	45°
11	1.38	14.15	4.26E+18	6.33E+18	7.10E+18	7.43E+18	7.43E+18	45°
12	1.44	15.60	4.65E+18	6.93E+18	7.76E+18	8.06E+18	8.09E+18	22°
13	1.45	17.04	5.05E+18	7.54E+18	8.44E+18	8.68E+18	8.81E+18	22°
14	1.37	18.41	5.47E+18	8.16E+18	9.12E+18	9.33E+18	9.52E+18	22°
15	1.43	19.85	5.90E+18	8.82E+18	9.85E+18	1.00E+19	1.03E+19	22°
16	1.41	21.26	6.35E+18	9.50E+18	1.06E+19	1.07E+19	1.11E+19	22°
17	1.45	22.71	6.78E+18	1.02E+19	1.13E+19	1.14E+19	1.19E+19	22°
18	1.36	24.07	7.23E+18	1.08E+19	1.20E+19	1.20E+19	1.26E+19	22°
19[(b)]	1.49	25.56	7.69E+18	1.15E+19	1.28E+19	1.28E+19	1.34E+19	22°
		32.00	9.71E+18	1.45E+19	1.60E+19	1.59E+19	1.69E+19	22°
		36.00	1.10E+19	1.64E+19	1.80E+19	1.78E+19	1.90E+19	22°
		40.00	1.22E+19	1.82E+19	2.01E+19	1.98E+19	2.12E+19	22°
		44.00	1.35E+19	2.01E+19	2.21E+19	2.17E+19	2.33E+19	22°
		48.00	1.47E+19	2.20E+19	2.41E+19	2.37E+19	2.55E+19	22°
		52.00	1.60E+19	2.38E+19	2.62E+19	2.56E+19	2.77E+19	22°
		54.00	1.66E+19	2.48E+19	2.72E+19	2.66E+19	2.87E+19	22°
		57.00	1.76E+19	2.62E+19	2.87E+19	2.80E+19	3.03E+19	22°
		60.00	1.85E+19	2.76E+19	3.02E+19	2.95E+19	3.20E+19	22°

Notes:

(a) 4A indicates time at which Capsule X was withdrawn

(b) Values beyond EOC 19 are projected

Table 2-12 Calculated Maximum Iron Atom Displacement Rate at the Braidwood Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	1.97E-11	3.13E-11	3.53E-11	4.04E-11	4.04E-11	45°
2	1.12	2.30	1.57E-11	2.28E-11	2.76E-11	3.33E-11	3.33E-11	45°
3	1.12	3.42	1.59E-11	2.37E-11	2.70E-11	3.07E-11	3.07E-11	45°
4A[(a)]	0.82	4.24	1.61E-11	2.39E-11	2.66E-11	3.01E-11	3.01E-11	45°
4B	0.34	4.58	1.62E-11	2.41E-11	2.69E-11	3.05E-11	3.05E-11	45°
5	1.27	5.85	1.51E-11	2.20E-11	2.44E-11	2.66E-11	2.66E-11	45°
6	1.34	7.19	1.46E-11	2.19E-11	2.46E-11	2.71E-11	2.71E-11	45°
7	1.37	8.56	1.25E-11	2.17E-11	2.52E-11	2.34E-11	2.68E-11	23°
8	1.40	9.96	1.43E-11	2.12E-11	2.27E-11	2.06E-11	2.43E-11	22°
9	1.36	11.33	1.41E-11	1.87E-11	2.05E-11	2.14E-11	2.14E-11	45°
10	1.45	12.78	1.28E-11	1.73E-11	1.97E-11	2.02E-11	2.02E-11	45°
11	1.38	14.15	1.45E-11	1.93E-11	2.12E-11	2.24E-11	2.24E-11	45°
12	1.44	15.60	1.41E-11	2.13E-11	2.36E-11	2.31E-11	2.48E-11	22°
13	1.45	17.04	1.37E-11	2.08E-11	2.30E-11	2.14E-11	2.42E-11	22°
14	1.37	18.41	1.52E-11	2.18E-11	2.44E-11	2.39E-11	2.52E-11	23°
15	1.43	19.85	1.48E-11	2.27E-11	2.48E-11	2.40E-11	2.65E-11	22°
16	1.41	21.26	1.54E-11	2.32E-11	2.52E-11	2.38E-11	2.70E-11	22°
17	1.45	22.71	1.48E-11	2.30E-11	2.51E-11	2.36E-11	2.69E-11	22°
18	1.36	24.07	1.65E-11	2.25E-11	2.36E-11	2.46E-11	2.48E-11	22°
19	1.49	25.56	1.51E-11	2.28E-11	2.58E-11	2.48E-11	2.69E-11	23°

Notes
(a) 4A indicates time at which Capsule X was withdrawn

Table 2-13 Calculated Maximum Iron Atom Displacements at the Braidwood Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacements (dpa)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	7.33E-04	1.16E-03	1.31E-03	1.50E-03	1.50E-03	45°
2	1.12	2.30	1.27E-03	1.94E-03	2.25E-03	2.64E-03	2.64E-03	45°
3	1.12	3.42	1.83E-03	2.78E-03	3.20E-03	3.72E-03	3.72E-03	45°
4A[(a)]	0.82	4.24	2.25E-03	3.40E-03	3.89E-03	4.50E-03	4.50E-03	45°
4B	0.34	4.58	2.42E-03	3.65E-03	4.18E-03	4.82E-03	4.82E-03	45°
5	1.27	5.85	3.02E-03	4.53E-03	5.15E-03	5.88E-03	5.88E-03	45°
6	1.34	7.19	3.64E-03	5.45E-03	6.19E-03	7.03E-03	7.03E-03	45°
7	1.37	8.56	4.18E-03	6.39E-03	7.28E-03	8.04E-03	8.04E-03	45°
8	1.40	9.96	4.81E-03	7.33E-03	8.28E-03	8.96E-03	8.96E-03	45°
9	1.36	11.33	5.41E-03	8.12E-03	9.15E-03	9.86E-03	9.86E-03	45°
10	1.45	12.78	5.99E-03	8.91E-03	1.00E-02	1.08E-02	1.08E-02	45°
11	1.38	14.15	6.62E-03	9.75E-03	1.10E-02	1.18E-02	1.18E-02	45°
12	1.44	15.60	7.23E-03	1.07E-02	1.20E-02	1.28E-02	1.28E-02	45°
13	1.45	17.04	7.86E-03	1.16E-02	1.30E-02	1.37E-02	1.37E-02	45°
14	1.37	18.41	8.51E-03	1.26E-02	1.41E-02	1.48E-02	1.48E-02	45°
15	1.43	19.85	9.18E-03	1.36E-02	1.52E-02	1.58E-02	1.58E-02	45°
16	1.41	21.26	9.87E-03	1.46E-02	1.63E-02	1.69E-02	1.70E-02	22°
17	1.45	22.71	1.05E-02	1.57E-02	1.75E-02	1.80E-02	1.82E-02	22°
18	1.36	24.07	1.13E-02	1.66E-02	1.85E-02	1.90E-02	1.93E-02	22°
19[(b)]	1.49	25.56	1.20E-02	1.77E-02	1.97E-02	2.02E-02	2.06E-02	22°
		32.00	1.51E-02	2.23E-02	2.47E-02	2.52E-02	2.59E-02	22°
		36.00	1.71E-02	2.52E-02	2.79E-02	2.82E-02	2.92E-02	22°
		40.00	1.90E-02	2.81E-02	3.10E-02	3.13E-02	3.25E-02	22°
		44.00	2.10E-02	3.10E-02	3.41E-02	3.44E-02	3.58E-02	22°
		48.00	2.29E-02	3.38E-02	3.73E-02	3.75E-02	3.91E-02	22°
		52.00	2.49E-02	3.67E-02	4.04E-02	4.05E-02	4.24E-02	22°
		54.00	2.58E-02	3.81E-02	4.20E-02	4.21E-02	4.41E-02	22°
		57.00	2.73E-02	4.03E-02	4.43E-02	4.44E-02	4.65E-02	22°
		60.00	2.88E-02	4.24E-02	4.67E-02	4.67E-02	4.90E-02	22°

Notes:

(a) 4A indicates time at which Capsule X was withdrawn

(b) Values beyond EOC 19 are projected

Table 2-14 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Braidwood Unit 2 Pressure Vessel Welds and Shells

Material	Fast Neutron Fluence (n/cm^2)			
	24.07 EFPY	**25.56 EFPY**	**32 EFPY**	**36 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	3.23E+16	3.44E+16	4.37E+16	4.95E+16
Inlet Nozzle Forging to Vessel Shell Welds	4.29E+16	4.58E+16	5.81E+16	6.57E+16
Nozzle Shell Forging[(a)]	4.09E+18	4.36E+18	5.50E+18	6.21E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	4.09E+18	4.36E+18	5.50E+18	6.21E+18
Intermediate Shell Forging	1.24E+19	1.32E+19	1.66E+19	1.86E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.21E+19	1.29E+19	1.62E+19	1.83E+19
Lower Shell Forging	1.26E+19	1.34E+19	1.69E+19	1.90E+19
Lower Shell to Lower Vessel Head Circumferential Weld	5.66E+15	6.03E+15	7.61E+15	8.59E+15

Material	Fast Neutron Fluence (n/cm^2)			
	48 EFPY	**54 EFPY**	**57 EFPY**	**60 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	6.67E+16	7.54E+16	7.97E+16	8.40E+16
Inlet Nozzle Forging to Vessel Shell Welds	8.87E+16	1.00E+17	1.06E+17	1.12E+17
Nozzle Shell Forging[(a)]	8.34E+18	9.41E+18	9.94E+18	1.05E+19
Nozzle Shell to Intermediate Shell Circumferential Weld	8.34E+18	9.41E+18	9.94E+18	1.05E+19
Intermediate Shell Forging	2.49E+19	2.80E+19	2.95E+19	3.11E+19
Intermediate Shell to Lower Shell Circumferential Weld	2.44E+19	2.75E+19	2.90E+19	3.06E+19
Lower Shell Forging	2.55E+19	2.87E+19	3.03E+19	3.20E+19
Lower Shell to Lower Vessel Head Circumferential Weld	1.15E+16	1.30E+16	1.37E+16	1.45E+16

Note

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld.

Table 2-15 Calculated Maximum Iron Atom Displacements at the Braidwood Unit 2 Pressure Vessel Welds and Shells

Material	Displacements (dpa)			
	24.07 EFPY	25.56 EFPY	32 EFPY	36 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	8.60E-05	9.13E-05	1.14E-04	1.29E-04
Inlet Nozzle Forging to Vessel Shell Welds	9.54E-05	1.01E-04	1.27E-04	1.43E-04
Nozzle Shell Forging[(a)]	6.27E-03	6.68E-03	8.43E-03	9.52E-03
Nozzle Shell to Intermediate Shell Circumferential Weld	6.27E-03	6.68E-03	8.43E-03	9.52E-03
Intermediate Shell Forging	1.91E-02	2.03E-02	2.54E-02	2.86E-02
Intermediate Shell to Lower Shell Circumferential Weld	1.87E-02	1.99E-02	2.50E-02	2.81E-02
Lower Shell Forging	1.93E-02	2.06E-02	2.59E-02	2.92E-02
Lower Shell to Lower Vessel Head Circumferential Weld	3.57E-05	3.79E-05	4.75E-05	5.34E-05

Material	Displacements (dpa)			
	48 EFPY	54 EFPY	57 EFPY	60 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.72E-04	1.94E-04	2.05E-04	2.15E-04
Inlet Nozzle Forging to Vessel Shell Welds	1.91E-04	2.15E-04	2.27E-04	2.39E-04
Nozzle Shell Forging[(a)]	1.28E-02	1.44E-02	1.52E-02	1.61E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	1.28E-02	1.44E-02	1.52E-02	1.61E-02
Intermediate Shell Forging	3.82E-02	4.29E-02	4.53E-02	4.77E-02
Intermediate Shell to Lower Shell Circumferential Weld	3.76E-02	4.23E-02	4.47E-02	4.70E-02
Lower Shell Forging	3.91E-02	4.41E-02	4.65E-02	4.90E-02
Lower Shell to Lower Vessel Head Circumferential Weld	7.15E-05	8.06E-05	8.52E-05	8.97E-05

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld.

Table 2-16 Calculated Fast Neutron Fluence Rate and Fluence (E > 1.0 MeV) at Braidwood Unit 1 Surveillance Capsule Positions[(a)]

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)			Fluence (n/cm^2)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.16	1.16	9.76E+10	1.06E+11	1.04E+11	3.58E+18	3.88E+18	3.82E+18
2	0.86	2.02	6.77E+10	7.39E+10	7.29E+10	5.41E+18	5.89E+18	5.80E+18
3	1.13	3.15	7.92E+10	8.83E+10	8.72E+10	8.24E+18	9.04E+18	8.92E+18
4	1.15	4.30	7.01E+10	7.47E+10	7.37E+10	1.08E+19	1.17E+19	1.16E+19
5	1.22	5.52	7.18E+10	7.87E+10	7.77E+10	1.35E+19	1.48E+19	1.46E+19
6	1.04	6.55	6.58E+10	6.82E+10	6.72E+10	1.57E+19	1.70E+19	1.68E+19
7	1.24	7.79	7.02E+10	7.93E+10	7.83E+10	1.84E+19	2.01E+19	1.98E+19
8	1.29	9.08	5.14E+10	5.55E+10	5.48E+10	2.05E+19	2.24E+19	2.20E+19
9	1.44	10.52	5.56E+10	6.25E+10	6.17E+10	2.30E+19	2.52E+19	2.49E+19
10	1.49	12.01	6.30E+10	6.60E+10	6.50E+10	2.60E+19	2.83E+19	2.79E+19
11	1.42	13.43	6.43E+10	6.78E+10	6.68E+10	2.89E+19	3.13E+19	3.09E+19
12	1.45	14.88	6.36E+10	6.72E+10	6.62E+10	3.18E+19	3.44E+19	3.39E+19
13	1.40	16.28	5.73E+10	6.16E+10	6.07E+10	3.43E+19	3.71E+19	3.66E+19
14	1.41	17.69	6.33E+10	6.74E+10	6.64E+10	3.71E+19	4.01E+19	3.96E+19

Note:

(a) Dual 29° applies to surveillance Capsules V and Y. Dual 31 5° applies to surveillance Capsules U and X and Single 31 5° applies to surveillance Capsules W and Z.

Table 2-17 Calculated Iron Atom Displacement Rate and Iron Atom Displacements at Braidwood Unit 1 Surveillance Capsule Positions[a]

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)			Displacements (dpa)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.16	1.16	1.93E-10	2.09E-10	2.06E-10	7.07E-03	7.66E-03	7.54E-03
2	0.86	2.02	1.33E-10	1.45E-10	1.43E-10	1.07E-02	1.16E-02	1.14E-02
3	1.13	3.15	1.55E-10	1.73E-10	1.71E-10	1.62E-02	1.78E-02	1.75E-02
4	1.15	4.30	1.37E-10	1.46E-10	1.44E-10	2.12E-02	2.31E-02	2.27E-02
5	1.22	5.52	1.41E-10	1.54E-10	1.52E-10	2.66E-02	2.90E-02	2.86E-02
6	1.04	6.55	1.29E-10	1.34E-10	1.31E-10	3.08E-02	3.34E-02	3.29E-02
7	1.24	7.79	1.37E-10	1.55E-10	1.53E-10	3.62E-02	3.94E-02	3.88E-02
8	1.29	9.08	1.00E-10	1.08E-10	1.07E-10	4.03E-02	4.38E-02	4.32E-02
9	1.44	10.52	1.09E-10	1.22E-10	1.21E-10	4.52E-02	4.94E-02	4.87E-02
10	1.49	12.01	1.23E-10	1.29E-10	1.27E-10	5.10E-02	5.54E-02	5.46E-02
11	1.42	13.43	1.26E-10	1.32E-10	1.30E-10	5.66E-02	6.14E-02	6.04E-02
12	1.45	14.88	1.24E-10	1.31E-10	1.29E-10	6.23E-02	6.74E-02	6.63E-02
13	1.40	16.28	1.12E-10	1.20E-10	1.18E-10	6.72E-02	7.27E-02	7.15E-02
14	1.41	17.69	1.24E-10	1.32E-10	1.29E-10	7.27E-02	7.85E-02	7.73E-02

Note

(a) Dual 29° applies to surveillance Capsules V and Y. Dual 31.5° applies to surveillance Capsules U and X and Single 31 5° applies to surveillance Capsules W and Z.

Table 2-18 Calculated Braidwood Unit 1 Surveillance Capsule Lead Factors

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Lead Factor		
			Dual 29°	Dual 31.5°	Single 31.5°
1	1.16	1.16	3.70	4.01[(a)]	3.96
2	0.86	2.02	3.70	4.03	3.97
3	1.13	3.15	3.65	4.00	3.95
4	1.15	4.30	3.74	4.07[(b)]	4.02
5	1.22	5.52	3.72	4.06	4.00
6	1.04	6.55	3.80	4.12	4.06
7	1.24	7.79	3.77	4.11	4.06[(c)]
8	1.29	9.08	3.77	4.11	4.05
9	1.44	10.52	3.74	4.09	4.04
10	1.49	12.01	3.82[(e)]	4.15	4.10[(f)]
11	1.42	13.43	3.87	4.20	4.14
12	1.45	14.88	3.91	4.23	4.17
13	1.4	16.28	3.92	4.24	4.18
14[(g)]	1.41	17.69	3.92[(d)]	4.24	4.18

Notes:

(a) Applicable to Surveillance Capsule U (withdrawn at EOC 1)
(b) Applicable to Surveillance Capsule X (withdrawn at EOC 4).
(c) Applicable to Surveillance Capsule W (withdrawn at EOC 7).
(d) Applicable to Surveillance Capsule V (withdrawn at EOC 14)
(e) Applicable to Surveillance Capsule Y (withdrawn at EOC 10).
(f) Applicable to Surveillance Capsule Z (withdrawn at EOC 10)
(g) Lead factors were not calculated beyond Cycle 14, as all surveillance capsules have been removed.

Table 2-19 Projected Fast Neutron Fluence Rate (E > 1.0 MeV) at Braidwood Unit 1 Surveillance Capsule Positions (Future Operation)

Capsule Position	Fluence Rate (n/cm^2-s)
Dual 29°	6.87E+10
Dual 31.5°	7.23E+10
Single 31.5°	7.12E+10

Table 2-20 Calculated Fast Neutron Fluence Rate and Fluence (E > 1.0 MeV) at Braidwood Unit 2 Surveillance Capsule Positions[(a)]

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)			Fluence (n/cm^2)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	9.61E+10	1.04E+11	1.03E+11	3.57E+18	3.87E+18	3.81E+18
2	1.12	2.30	6.98E+10	7.86E+10	7.76E+10	6.04E+18	6.65E+18	6.57E+18
3	1.12	3.42	7.27E+10	7.91E+10	7.80E+10	8.60E+18	9.44E+18	9.31E+18
4A	0.82	4.24	7.16E+10	7.79E+10	7.68E+10	1.05E+19	1.15E+19	1.13E+19
4B	0.34	4.58	7.23E+10	7.87E+10	7.76E+10	1.12E+19	1.23E+19	1.21E+19
5	1.27	5.85	6.34E+10	6.84E+10	6.74E+10	1.38E+19	1.50E+19	1.48E+19
6	1.34	7.19	6.31E+10	6.87E+10	6.78E+10	1.64E+19	1.79E+19	1.77E+19
7	1.37	8.56	6.65E+10	6.92E+10	6.81E+10	1.93E+19	2.09E+19	2.07E+19
8	1.40	9.96	5.88E+10	6.16E+10	6.07E+10	2.19E+19	2.37E+19	2.33E+19
9	1.36	11.33	5.41E+10	5.88E+10	5.80E+10	2.43E+19	2.62E+19	2.58E+19
10	1.45	12.78	5.08E+10	5.53E+10	5.46E+10	2.66E+19	2.87E+19	2.83E+19
11	1.38	14.15	5.45E+10	5.94E+10	5.86E+10	2.89E+19	3.13E+19	3.09E+19
12	1.44	15.60	6.19E+10	6.57E+10	6.47E+10	3.18E+19	3.43E+19	3.38E+19
13	1.45	17.04	6.11E+10	6.45E+10	6.36E+10	3.46E+19	3.72E+19	3.67E+19
14	1.37	18.41	6.26E+10	6.69E+10	6.60E+10	3.73E+19	4.01E+19	3.96E+19

Note:

(a) Dual 29° applies to surveillance Capsules V and Y. Dual 31 5° applies to surveillance Capsules U and X and Single 31.5° applies to surveillance Capsules W and Z.

Table 2-21 Calculated Iron Atom Displacement Rate and Iron Atom Displacements at Braidwood Unit 2 Surveillance Capsule Positions[a]

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)			Displacements (dpa)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	1.89E-10	2.05E-10	2.02E-10	7.03E-03	7.62E-03	7.50E-03
2	1.12	2.30	1.37E-10	1.54E-10	1.52E-10	1.19E-02	1.31E-02	1.29E-02
3	1.12	3.42	1.42E-10	1.55E-10	1.53E-10	1.69E-02	1.85E-02	1.83E-02
4A	0.82	4.24	1.40E-10	1.53E-10	1.50E-10	2.06E-02	2.25E-02	2.22E-02
4B	0.34	4.58	1.42E-10	1.54E-10	1.52E-10	2.21E-02	2.41E-02	2.38E-02
5	1.27	5.85	1.24E-10	1.34E-10	1.32E-10	2.70E-02	2.95E-02	2.91E-02
6	1.34	7.19	1.23E-10	1.34E-10	1.32E-10	3.22E-02	3.52E-02	3.47E-02
7	1.37	8.56	1.30E-10	1.35E-10	1.33E-10	3.79E-02	4.10E-02	4.04E-02
8	1.40	9.96	1.15E-10	1.20E-10	1.18E-10	4.30E-02	4.63E-02	4.56E-02
9	1.36	11.33	1.06E-10	1.15E-10	1.13E-10	4.75E-02	5.13E-02	5.05E-02
10	1.45	12.78	9.90E-11	1.08E-10	1.06E-10	5.20E-02	5.62E-02	5.53E-02
11	1.38	14.15	1.06E-10	1.16E-10	1.14E-10	5.66E-02	6.12E-02	6.03E-02
12	1.44	15.60	1.21E-10	1.28E-10	1.26E-10	6.21E-02	6.71E-02	6.60E-02
13	1.45	17.04	1.19E-10	1.26E-10	1.24E-10	6.76E-02	7.28E-02	7.17E-02
14	1.37	18.41	1.22E-10	1.30E-10	1.28E-10	7.29E-02	7.84E-02	7.72E-02

Note:

(a) Dual 29° applies to surveillance Capsules V and Y. Dual 31.5° applies to surveillance Capsules U and X and Single 31.5° applies to surveillance Capsules W and Z.

Table 2-22 Calculated Braidwood Unit 2 Surveillance Capsule Lead Factors

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Lead Factor		
			Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	3.77	4.08(a)	4.03
2	1.12	2.30	3.62	3.99	3.94
3	1.12	3.42	3.66	4.02	3.96
4A	0.82	4.24	3.68	4.03(b)	3.98
4B	0.34	4.58	3.68	4.03	3.98
5	1.27	5.85	3.70	4.04	3.99
6	1.34	7.19	3.70	4.04	3.98
7	1.37	8.56	3.80	4.12	4.06(c)
8	1.40	9.96	3.87	4.18	4.12
9	1.36	11.33	3.89	4.20	4.14
10	1.45	12.78	3.90(e)	4.21	4.15(f)
11	1.38	14.15	3.89	4.21	4.15
12	1.44	15.60	3.93	4.24	4.18
13	1.45	17.04	3.92	4.23	4.17
14(g)	1.37	18.41	3.92(d)	4.22	4.16

Notes:

(a) Applicable to Surveillance Capsule U (withdrawn at EOC 1).
(b) Applicable to Surveillance Capsule X (withdrawn during a mid-cycle outage in Cycle 4).
(c) Applicable to Surveillance Capsule W (withdrawn at EOC 7).
(d) Applicable to Surveillance Capsule V (withdrawn at EOC 14)
(e) Applicable to Surveillance Capsule Y (withdrawn at EOC 10).
(f) Applicable to Surveillance Capsule Z (withdrawn at EOC 10).
(g) Lead factors were not calculated beyond Cycle 14, as all surveillance capsules have been removed.

Table 2-23 Projected Fast Neutron Fluence Rate (E > 1.0 MeV) at Braidwood Unit 2 Surveillance Capsule Positions (Future Operation)

Capsule Position	Fluence Rate (n/cm^2-s)
Dual 29°	6.32E+10
Dual 31.5°	6.71E+10
Single 31.5°	6.62E+10

Table 2-24 Calculational Uncertainties

Description	Uncertainty	
	Capsule	Vessel Inner Radius
PCA Comparisons	3%	3%
H. B. Robinson Comparisons	3%	3%
Analytical Sensitivity Studies	10%	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%	5%
Net Calculational Uncertainty	12%	13%

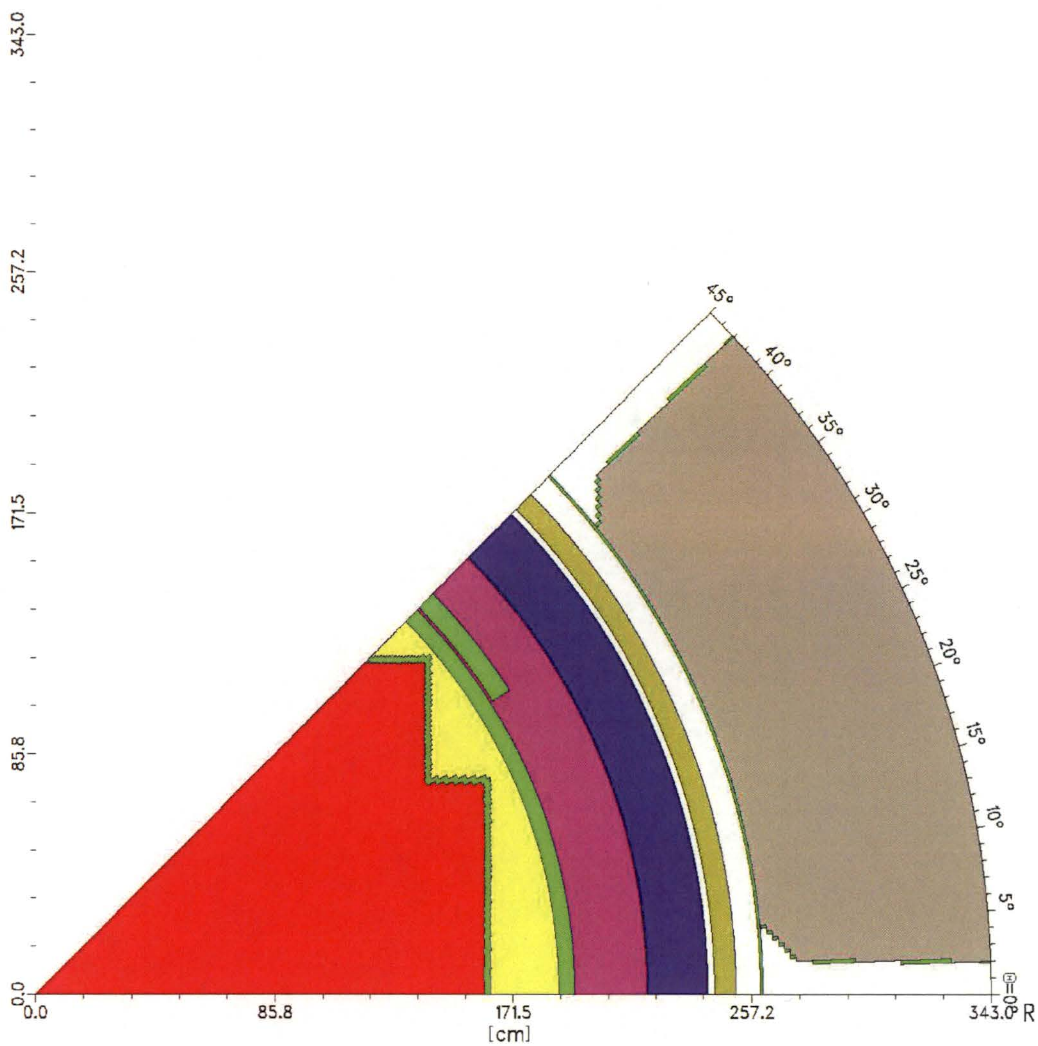

Figure 2-1 Braidwood Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 12.5-Degree Neutron Pad Configuration

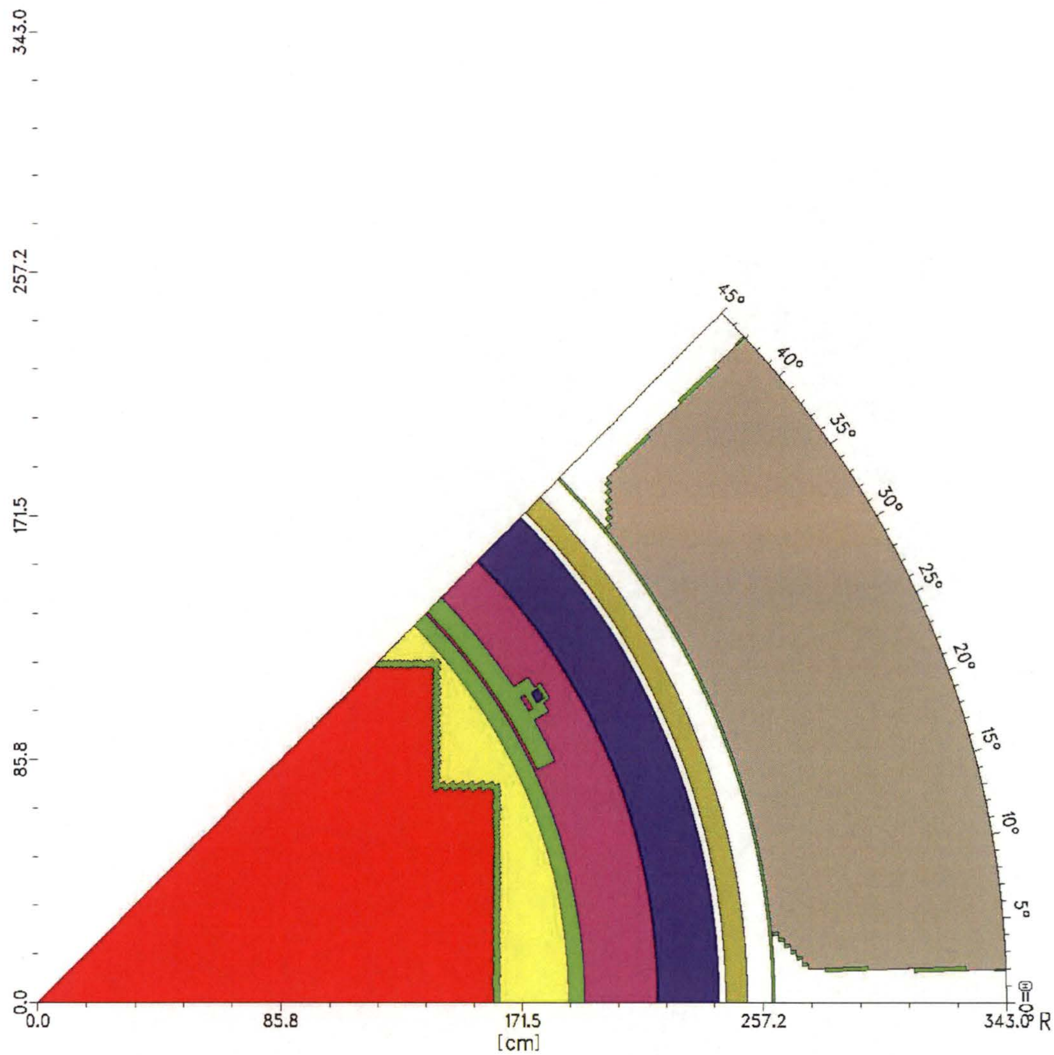

Figure 2-2 Braidwood Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 20.0-Degree Neutron Pad Configuration

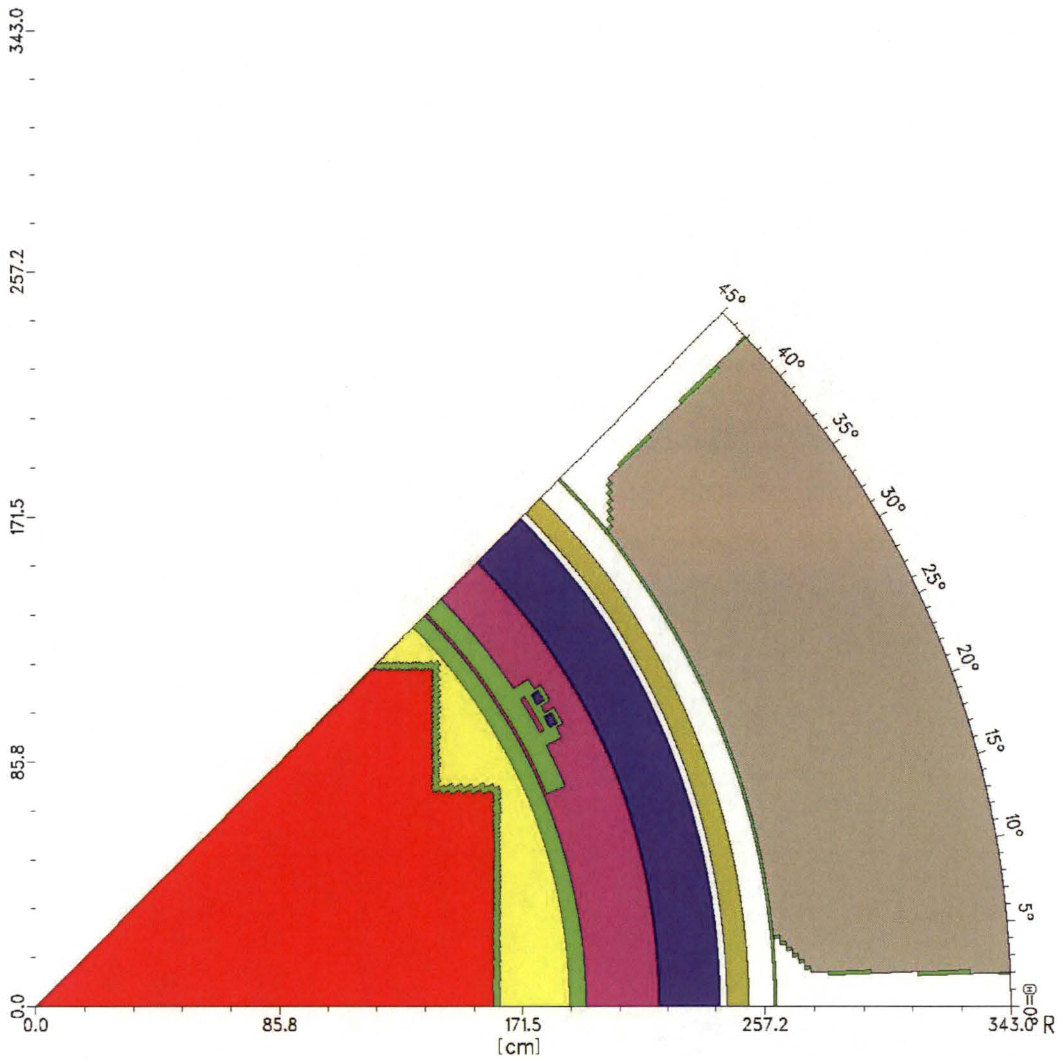

Figure 2-3 Braidwood Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 22.5-Degree Neutron Pad Configuration

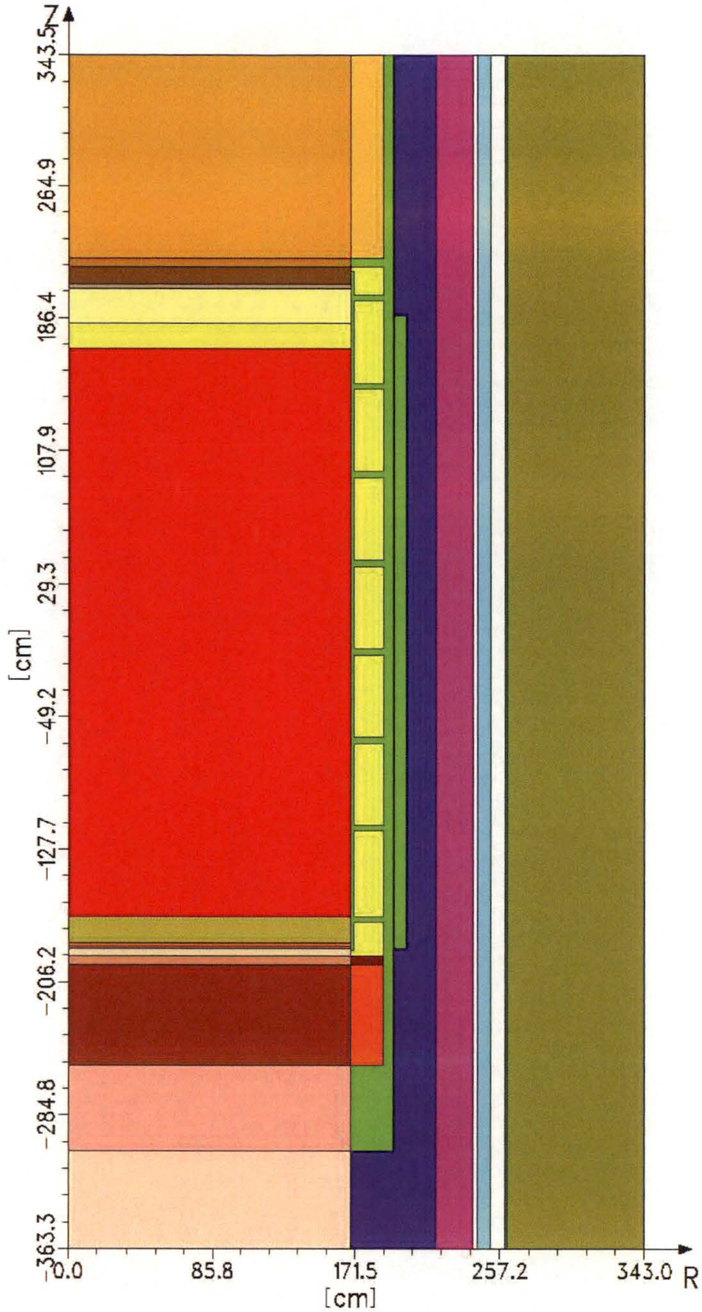

Figure 2-4 Braidwood Units 1 and 2 r, z Reactor Geometry

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [4]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *"the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Braidwood Units 1 and 2 beltline materials traditionally included the Nozzle Shell Forgings, Intermediate Shell Forgings, Lower Shell Forgings, Nozzle to Intermediate Shell Circumferential Welds, and Intermediate to Lower Shell Circumferential Welds; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [14], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. As seen from Table 2-8 and Table 2-14 of this report, the fluence for the inlet nozzle to nozzle shell welds are greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 57 EFPY. Thus, these materials are a part of the extended beltline. Since the fluence values for the inlet nozzle to nozzle shell forging welds are conservatively applied to the inlet nozzle forgings, the inlet nozzle forgings are also considered a part of the extended beltline. Although the reactor vessel outlet nozzle to nozzle shell forging welds have projected fluence values less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period, these materials are evaluated with the extended beltline materials herein. Per NRC RIS 2014-11, the nozzle forging materials must be evaluated for their potential effect on P-T limit curves regardless of exposure. See Appendix B for more details.

A summary of the best-estimate copper (Cu) and nickel (Ni) contents, in units of weight percent (wt. %), as well as the initial RT_{NDT} values for the reactor vessel beltline and extended beltline materials are provided in Table 3-1 and Table 3-2 for Braidwood Units 1 and 2, respectively. Table 3-3 provides the initial RT_{NDT} values for the reactor vessel closure head and vessel flange materials for Braidwood Units 1 and 2. These values are taken from References [45] and [46].

"Master Curve" Fracture Toughness Properties

As part of this P-T limit curve development effort, the initial RT_{NDT} for the Units 1 and 2 inlet/outlet nozzle to upper shell forging welds were redefined in order to take advantage of the "Master Curve" method ($RT_{NDT} = T_0 + 35°F$) in BAW-2308 Revision 1-A Safety Evaluation (SE) and Revision 2-A SE [6]. When using these Master Curve-generated initial RT_{NDT} values, the chemistry factor (CF) and σ_Δ terms will be adjusted to a minimum of 167°F and 28°F, respectively; however, if the material-specific CF value is greater than 167°F, the material-specific value will be used. The values of 167°F and 28°F comply with the "Conditions and Limitation" placed on the use of "Master Curve" fracture toughness properties in Section 5.0 of Revision 1-A SE of [6].

The use of the Master Curve is a departure from the ASME Code, Section III, Subsection NB-2300 method required by 10 CFR 50, Appendix G; therefore, it requires the submittal and NRC approval of a 10 CFR 50.12 exemption for use of [6] in P-T limit curve development. All of the technical requirements outlined in the SEs for [6] have been met and justification for the use of the Master Curve method in [6] at Braidwood Units 1 and 2 is provided in Appendix G.

Table 3-1 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Braidwood Unit 1 Reactor Vessel Materials[a]

Reactor Vessel Material	Heat Number	Wt. % Cu[b]	Wt. % Ni[b]	$RT_{NDT(u)}$[c] (°F)
Reactor Vessel Beltline Materials				
Nozzle Shell Forging	5P-7016	0.04	0.73	10
Intermediate Shell Forging	[49D383/49C344]-1-1	0.05	0.73	-30
Lower Shell Forging	[49D867/49C813]-1-1	0.05	0.74	-20
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498 (Linde 80 flux type, Lot # 0261)	0.04	0.46	-25
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011 (Linde 80 flux type, Lot # 8061)	0.03	0.67	40
Surveillance Weld Material – Braidwood Unit 1	442011 (Linde 80 flux type, Lot # 8061)	0.03	0.67	---
Reactor Vessel Extended Beltline Materials				
Inlet Nozzle 01-001	21-3257	0.09	0.82	-20
Inlet Nozzle 01-002	21-3257	0.09	0.81	-10
Inlet Nozzle 02-001	22-3313	0.07	0.78	-10
Inlet Nozzle 02-002	22-3313	0.07	0.80	0
Outlet Nozzle 01-001	22-3025	0.13	0.83	-10
Outlet Nozzle 01-003	11-5226	0.09	0.84	-10
Outlet Nozzle 02-001	4-3329	0.08	0.82	-20
Outlet Nozzle 02-002	4-3383	0.08	0.81	-20
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403 (Linde 80 flux type, Lot # 0852)	0.29	0.56	-48.6[d]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403 (Linde 80 flux type, Lot # 0852)	0.29	0.56	-48.6[d]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-588	41403 (Linde 80 flux type, Lot # 8119)	0.29	0.63	-48.6[d]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-579	442010 (Linde 80 flux type, Lot # 8119)	0.25	0.63	-48.6[d]

Notes

(a) All data taken from Section 3 of [16], unless noted.

(b) The chemical compositions are based on Braidwood Unit 1 CMTR data.

(c) The initial RT_{NDT} values are based on measured data for all beltline and extended beltline materials. Initial RT_{NDT} values for the forging materials are based on Charpy test data for specimens oriented in the "weak" direction per Section B.1.1 of NUREG-0800 Branch Technical Position 5-3 [25]

(d) Generic value taken from Table 9 of [6] Revision 2-A with an associated σ_I of 18.0°F. See Appendix G for more details. Use of these values must also meet certain conditions when used in safety evaluations per [6].

Table 3-2 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Braidwood Unit 2 Reactor Vessel Materials[a]

Reactor Vessel Material	Heat Number	Wt. % Cu[b]	Wt. % Ni[b]	$RT_{NDT(u)}$[c] (°F)
Reactor Vessel Beltline Materials				
Nozzle Shell Forging	5P-7056	0.04	0.90	30
Intermediate Shell Forging	[49D963/49C904]-1-1	0.03	0.71	-30
Lower Shell Forging	[50D102/50C97]-1-1	0.06	0.76	-30
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498 (Linde 80 flux type, Lot # 0261)	0.04	0.46	-25
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011 (Linde 80 flux type, Lot # 8061)	0.03	0.67	40
Surveillance Weld Material – Braidwood Unit 2	442011 (Linde 80 flux type, Lot # 0344)	0.03	0.71	---
Reactor Vessel Extended Beltline Materials				
Inlet Nozzle 01-001	41-5414	0.07	0.83	-10
Inlet Nozzle 01-002	41-5414	0.07	0.85	-10
Inlet Nozzle 02-001	42-5417	0.09	0.88	-10
Inlet Nozzle 02-002	42-5417	0.09	0.89	-10
Outlet Nozzle 01-002	11-5266	0.09	0.86	10
Outlet Nozzle 01-003	11-5226	0.09	0.88	-10
Outlet Nozzle 02-001	4-3481	0.07	0.84	-10
Outlet Nozzle 02-002	4-3502	0.09	0.78	-10
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404 (Linde 80 flux type, Lot # 0261)	0.18	0.52	-48.6[d]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404 (Linde 80 flux type, Lot # 0261)	0.18	0.52	-48.6[d]

Notes:

(a) All data taken from Section 3 of [16], unless noted
(b) The chemical compositions are based on Braidwood Unit 2 CMTR data.
(c) The initial RT_{NDT} values are based on measured data for all beltline and extended beltline materials. Initial RT_{NDT} values for the forging materials are based on Charpy test data for specimens oriented in the "weak" direction per Section B.1.1 of NUREG-0800 Branch Technical Position 5-3 [25]
(d) Generic value taken from Table 9 of [6] Revision 2-A with an associated σ_I of 18.0°F. See Appendix G for more details. Use of these values must also meet certain conditions when used in safety evaluations per [6].

Table 3-3 Initial RT_{NDT} Values for the Braidwood Units 1 and 2 Reactor Vessel Closure Head and Vessel Flange Materials

Reactor Vessel Material	Unit 1 Initial RT_{NDT} (°F)	Unit 2 Initial RT_{NDT} (°F)
Closure Head	-20	20
Vessel Flange	-10	20

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [1], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister plant' data.

The surveillance capsule forging material for Braidwood Units 1 and 2 is from Lower Shell Forging [49D867/49C813]-1-1 and Lower Shell Forging [50D102/50C97]-1-1, respectively. Per Appendix D, the surveillance data are deemed credible for Braidwood Unit 1 and non-credible for Braidwood Unit 2; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7 and the pressurized thermal shock (PTS) calculations contained in Appendix E for the Braidwood Unit 1 Lower Shell Forging and a full margin term will be utilized for the Braidwood Unit 2 Lower Shell Forging.

The surveillance capsule weld material for Braidwood Units 1 and 2 is Heat # 442011, which is applicable to the intermediate to lower shell circumferential weld at each unit. Per Appendix D, the surveillance weld data are deemed credible; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7 and the PTS calculations contained in Appendix E for Heat # 442011.

Table 4-1 and Table 4-2 summarize the Braidwood Units 1 and 2 surveillance data for the plate and weld material that will be used in the calculation of the Position 2.1 chemistry factor values for the relevant materials.

Table 4-1 Braidwood Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.388	5.6
	X	1.17	37.9
	W	1.98	24.0
	V	3.71	51.3
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.388	0.0[c]
	X	1.17	29.3
	W	1.98	37.1
	V	3.71	39.7
Surveillance Weld Material (Heat # 442011)	U	0.388	17.4
	X	1.17	29.8
	W	1.98	49.0
	V	3.71	62.8

Notes:

(a) Fluence values are from Section 2.

(b) Measured ΔRT_{NDT} values are from [17].

(c) Per [17], this ΔRT_{NDT} value was calculated to be negative (-15 8°F). Physically, this should not occur; therefore, a conservative value of 0°F is shown in this table.

Table 4-2 Braidwood Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)
Lower Shell Forging [50D102/50C97]-1-1 (Tangential)	U	0.387	0.0[c]
	X	1.15	0.0[c]
	W	2.07	4.6
	V	3.73	28.4
Lower Shell Forging [50D102/50C97]-1-1 (Axial)	U	0.387	0.0[c]
	X	1.15	33.8
	W	2.07	33.1
	V	3.73	63.3
Surveillance Weld Material (Heat # 442011)	U	0.387	0.0[c]
	X	1.15	26.1
	W	2.07	23.7
	V	3.73	45.6

Notes.

(a) Fluence values are from Section 2.

(b) Measured ΔRT_{NDT} values are from [18].

(c) Per [18], these ΔRT_{NDT} values were calculated to be negative. The actual values of ΔRT_{NDT} are -9 6°F (Capsule U Tangential), -9.6°F (Capsule X Tangential), -0.1°F (Capsule U Axial), and -0.8°F (Capsule U Weld) Physically, this should not occur; therefore, a conservative value of 0°F is shown in this table.

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using Regulatory Guide 1.99, Revision 2 [1], Positions 1.1 and 2.1. Position 1.1 CFs for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of [1]. The best-estimate copper and nickel weight percent values for the Braidwood Units 1 and 2 reactor vessel materials are provided in Table 3-1 and Table 3-2 of this report.

The Position 2.1 CFs are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in [1]. The Braidwood Units 1 and 2 surveillance data was summarized in Section 4 of this report, and will be utilized in the Position 2.1 CF calculations in this Section. Table 5-1 through Table 5-3 calculate the Braidwood Units 1 and 2 Position 2.1 CFs.

Position 1.1 and Position 2.1 CFs are summarized in Table 5-4 and Table 5-5 for Braidwood Units 1 and 2, respectively. Adjustment of the ΔRT_{NDT} values due to chemistry and temperature differences between the surveillance and vessel material per [1] was not required, because the surveillance material has the same Position 1.1 CF as both intermediate shell to lower shell forging weld seams. Since the CFs are equal, the chemistry adjustment factor would be equal to 1. In addition, no temperature adjustments are needed since Braidwood Units 1 and 2 have similar operating temperatures.

Table 5-1 Braidwood Unit 1 Reactor Vessel Forgings Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.388	0.738	5.6	4.13	0.54
	X	1.17	1.044	37.9	39.56	1.09
	W	1.98	1.186	24.0	28.48	1.41
	V	3.71	1.340	51.3	68.73	1.79
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.388	0.738	0.0[d]	0.00	0.54
	X	1.17	1.044	29.3	30.58	1.09
	W	1.98	1.186	37.1	44.02	1.41
	V	3.71	1.340	39.7	53.19	1.79
				SUM:	268.68	9.67
	$CF_{[49D867/49C813]-1-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (268.68) \div (9.67) =$ **27.8°F**					

Notes

(a) The calculated fluence values are from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10 \cdot \log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [17].

(d) A negative ΔRT_{NDT} value was calculated (-15.8°F). Physically, this should not occur; therefore, a conservative value of zero was used in this calculation.

Table 5-2 Braidwood Unit 2 Reactor Vessel Forgings Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] ($x\ 10^{19}$ n/cm², E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [50D102/50C97]-1-1 (Tangential)	U	0.387	0.737	0.0[d]	0.00	0.54
	X	1.15	1.039	0.0[d]	0.00	1.08
	W	2.07	1.198	4.6	5.51	1.44
	V	3.73	1.341	28.4	38.08	1.80
Lower Shell Forging [50D102/50C97]-1-1 (Axial)	U	0.387	0.737	0.0[d]	0.00	0.54
	X	1.15	1.039	33.8	35.12	1.08
	W	2.07	1.198	33.1	39.66	1.44
	V	3.73	1.341	63.3	84.88	1.80
				SUM:	203.25	9.71
	$CF_{[50D102/50C97]-1-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (203.25) \div (9.71) =$ **20.9°F**					

Notes:

(a) The calculated fluence values are from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [18]

(d) A negative ΔRT_{NDT} values was calculated. The actual values of ΔRT_{NDT} are -9.6°F (Capsule U Tangential), -9 6°F (Capsule X Tangential), and -0.1°F (Capsule U Axial) Physically, this should not occur, therefore, a conservative value of zero was used in this calculation

Table 5-3 Braidwood Units 1 and 2 Reactor Vessel Welds Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Braidwood Unit 1 Surveillance Weld Material (Heat # 442011)	U	0.388	0.738	17.4	12.84	0.54
	X	1.17	1.044	29.8	31.11	1.09
	W	1.98	1.186	49.0	58.14	1.41
	V	3.71	1.340	62.8	84.13	1.79
Braidwood Unit 2 Surveillance Weld Material (Heat # 442011)	U	0.387	0.737	0.0[d]	0.00	0.54
	X	1.15	1.039	26.1	27.12	1.08
	W	2.07	1.198	23.7	28.39	1.44
	V	3.73	1.341	45.6	61.14	1.80
				SUM:	302.87	9.69
	$CF_{Weld\ Heat\ \#\ 442011} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (302.87) \div (9.69) =$ **31.2°F**					

Notes:

(a) The calculated fluence values are from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [17] and [18]. The surveillance weld ΔRT_{NDT} values have not been adjusted using the ratio procedure of Reg Guide 1.99, Revision 2 since the ratio is equal to 1 In addition, no temperature adjustments were necessary because the Braidwood Units 1 and 2 surveillance capsules were irradiated at essentially the same temperature.

(d) A negative ΔRT_{NDT} value was calculated (-0 8°F) Physically, this should not occur; therefore, a conservative value of zero was used in this calculation

Table 5-4 Summary of Braidwood Unit 1 Position 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material	Heat Number	Chemistry Factor (°F)	
		Position 1.1[(a)]	Position 2.1[(b)]
Reactor Vessel Beltline Materials			
Nozzle Shell Forging	5P-7016	26	
Intermediate Shell Forging	[49D383/49C344]-1-1	31	
Lower Shell Forging	[49D867/49C813]-1-1	31	27.8
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498	54	
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011	41	31.2
Surveillance Weld Material – Braidwood Unit 1	442011	41	
Reactor Vessel Extended Beltline Materials			
Inlet Nozzle 01-001	21-3257	58	
Inlet Nozzle 01-002	21-3257	58	
Inlet Nozzle 02-001	22-3313	44	
Inlet Nozzle 02-002	22-3313	44	
Outlet Nozzle 01-001	22-3025	96	
Outlet Nozzle 01-003	11-5226	58	
Outlet Nozzle 02-001	4-3329	51	
Outlet Nozzle 02-002	4-3383	51	
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403	185.6[(c)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403	185.6[(c)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-588	41403	195.7[(c)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-579	442010	181.0[(c)]	

Notes:

(a) Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-1 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [1]

(b) Position 2.1 chemistry factors were taken from Table 5-1 and Table 5-3 of this report. As discussed in Appendix D, the surveillance forging data is deemed credible. Also as discussed in Appendix D, the surveillance weld data is deemed credible.

(c) These CF values, calculated using Regulatory Guide 1 99, Revision 2, satisfy the condition stipulated in [6] that the CF be no less than 167°F when the initial RT_{NDT} values from BAW-2308 [6] are used.

Table 5-5 Summary of Braidwood Unit 2 Position 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material	Heat Number	Chemistry Factor (°F)	
		Position 1.1[(a)]	Position 2.1[(b)]
Reactor Vessel Beltline Materials			
Nozzle Shell Forging	5P-7056	26	
Intermediate Shell Forging	[49D963/49C904]-1-1	20	
Lower Shell Forging	[50D102/50C97]-1-1	37	20.9
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498	54	
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011	41	31.2
Surveillance Weld Material – Braidwood Unit 2	442011	41	
Reactor Vessel Extended Beltline Materials			
Inlet Nozzle 01-001	41-5414	44	
Inlet Nozzle 01-002	41-5414	44	
Inlet Nozzle 02-001	42-5417	58	
Inlet Nozzle 02-002	42-5417	58	
Outlet Nozzle 01-002	11-5266	58	
Outlet Nozzle 01-003	11-5226	58	
Outlet Nozzle 02-001	4-3481	44	
Outlet Nozzle 02-002	4-3502	58	
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404	167.0[(c)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404	167.0[(c)]	

Notes

(a) Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-2 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [1].

(b) Position 2 1 chemistry factors were taken from Table 5-2 and Table 5-3 of this report. As discussed in Appendix D, the surveillance forging data is deemed non-credible Also as discussed in Appendix D, the surveillance weld data is deemed credible.

(c) Minimum CF required by [6] as a condition for using values from [6] The actual calculated CF using Regulatory Guide 1 99, Revision 2 is 141.2°F.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) 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 the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code [3]. The K_{Ic} curve is given by the following equation:

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

where,

K_{Ic} (ksi√in.) = 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 Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

6.2 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:

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

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

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

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:

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

Axial Flaw Methodology

For plates, forgings, and longitudinal welds, M_m for an inside axial 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 $\sqrt{t} > 3.464$

and, M_m for an outside axial 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 $\sqrt{t} > 3.464$

Circumferential Flaw Methodology

Similarly, for circumferential welds, M_m for an inside or an outside circumferential surface flaw is given by:

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

Where:

p = internal pressure (ksi), R_i = vessel inner radius (in), and t = vessel wall thickness (in.).

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

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m (4)

The maximum K_I produced by radial thermal gradient for the postulated axial or circumferential inside surface defect of G-2120 is:

$K_{It} = 0.953 \times 10^{-3}$ x CR x $t^{2.5}$ (5)

where CR is the cooldown rate in °F/hr., or for a postulated axial or circumferential outside surface defect

$K_{It} = 0.753 \times 10^{-3}$ x HU x $t^{2.5}$ (6)

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 ASME Code, Section XI, Appendix G, Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Code, 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 Figure 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 1/4T axial or circumferential inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{7}$$

or similarly, K_{It} during heatup for a 1/4T outside axial or circumferential surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{8}$$

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:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{9}$$

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

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2] Section 2.6 (Equations 2.6.2-4 and 2.6.3-1). Finally, the reactor vessel metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A, Revision 4 (Equation 2.6.1-1). This equation is solved utilizing values for thermal diffusivity of 0.518 ft^2/hr at 70°F and 0.379 ft^2/hr at 550°F and a constant convective heat-transfer coefficient value of 7000 Btu/hr-ft^2-°F.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw (the postulated flaw has a depth of 1/4 of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, Paragraph G-2120), the appropriate value for RT_{NDT}, and the reference fracture toughness curve (Equation 1). 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 1/4T flaw of Appendix G to Section XI of 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 vessel wall because the thermal gradients, which increase with increasing cooldown rates, produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Since an inside surface flaw has a higher tensile stresses than as outside flaw and is subject to more neutron embrittlement than an outside surface flaw in the beltline region, postulation of outside flaw for cooldown conditions is unnecessary. Allowable P-T curves are generated for steady-state (zero-rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

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) across the vessel wall 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 could be lower 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 P-T 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 inside 1/4T flaw during heatup is lower than the K_{Ic} for the flaw 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 P-T 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 third portion of the heatup analysis concerns the calculation of the P-T 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 P-T curves for 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 least 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.

6.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [4] addresses the metal temperature of the closure head flange and vessel flange regions. However, per the technical basis in [5], the flange requirement has been eliminated for Braidwood Units 1 and 2. An exemption to 10 CFR Part 50, Appendix G was approved for the Byron and Braidwood Units per [26].

6.4 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [4]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [2], the minimum boltup temperature should be 60°F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. Since the limiting unirradiated RT_{NDT} of this region is 20°F per Table 3-3, the minimum boltup temperature for the Braidwood Units 1 and 2 reactor vessels is 60°F. This limit is shown in Figures 8-1 and 8-2.

7 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{10}$$

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 [15]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided 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 should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \tag{11}$$

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.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \tag{12}$$

where x inches (reactor vessel cylindrical shell 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 11 to calculate the ΔRT_{NDT} at the specific depth.

The projected reactor vessel neutron fluence was updated for this analysis and documented in Section 2 of this report. The evaluation methods used in Section 2 are consistent with the methods presented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2].

Table 7-1 and Table 7-2 contain the surface fluence values at 57 EFPY for Braidwood Units 1 and 2, respectively. These values are used for the development of the P-T limit curves contained in this report. Table 7-1 and Table 7-2 also contain the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2 [1]. The values in this table are used to calculate the 57 EFPY ART values for the Braidwood Units 1 and 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value. When a generic value is used, the σ_I is obtained from the set of data used to establish the mean. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings when surveillance data is not used or is non-credible, and 8.5°F (half the value) for plates or forgings when credible surveillance data is used For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used or is non-credible, and is 14°F (half the value) when credible surveillance capsule data is used. Per [1], the value for σ_Δ need not exceed 0.5 times the mean value of

ΔRT_{NDT}. When using initial RT_{NDT} values based on [6], such as those used for the Braidwood nozzle-to-shell welds, the values of σ_I and σ_Δ are stipulated by [6].

Contained in Tables 7-3 through 7-8 are the 57 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Braidwood Units 1 and 2 heatup and cooldown curves. For the materials listed in Table 7-3 through Table 7-8 circumferential flaws are considered in the circumferential weld materials and axial flaws are considered in all other materials. The limiting ART values for Braidwood Units 1 and 2 are summarized in Table 7-9.

The outlet nozzle forgings and welds for Braidwood Units 1 and 2 have projected fluence values that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY per Table 2-8 and Table 2-14 at the lowest extent of the nozzle weld. However, the inlet nozzle forgings and welds do have projected fluence values that exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY. Consistent with NRC RIS 2014-11 [14], neutron radiation embrittlement need not be considered herein for the outlet nozzle materials. However, ART calculations for both the inlet and outlet nozzle forging materials are conservatively evaluated in this section with consideration of embrittlement effects and without consideration for attenuation of the fluence through the vessel thickness.

Table 7-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the Braidwood Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Region	Surface Fluence, f [(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Surface FF	1/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials						
Nozzle Shell Forging	1.13	1.034	0.679	0.891	0.245	0.619
Intermediate Shell Forging	3.22	1.307	1.93	1.180	0.697	0.899
Lower Shell Forging	3.14	1.302	1.89	1.174	0.680	0.892
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	1.13	1.034	0.679	0.891	0.245	0.619
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	3.06	1.295	1.84	1.167	0.663	0.885
Reactor Vessel Extended Beltline Materials[(d)]						
Inlet Nozzles	0.0124	0.127	Note (c)			
Outlet Nozzles	0.00938[(b)]	0.105				
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.0124	0.127				
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.00938[(b)]	0 105				

Notes:

(a) 57 EFPY surface fluence values are documented in Table 2-8.

(b) The outlet nozzle materials do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY; therefore, per NRC RIS 2014-11 [14], neutron irradiation embrittlement need not be considered for the nozzle materials herein. However, the results are included for information

(c) The inlet and outlet nozzle forgings and welds are conservatively evaluated without consideration for the attenuation of the fluence through the vessel thickness.

(d) The nozzle forging fluence values are conservatively set equal to the fluence at the respective nozzle welds

Table 7-2 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the Braidwood Unit 2 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Region	Surface Fluence, f [(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Surface FF	1/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials						
Nozzle Shell Forging	0.994	0.998	0.597	0.856	0.215	0.587
Intermediate Shell Forging	2.95	1.287	1.77	1.157	0.639	0.874
Lower Shell Forging	3.03	1.293	1.82	1.164	0.656	0.882
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	0.994	0.998	0.597	0.856	0.215	0.587
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	2.90	1.283	1.74	1.153	0.628	0.870
Reactor Vessel Extended Beltline Materials[(d)]						
Inlet Nozzles	0.0106	0.114	Note (c)			
Outlet Nozzles	0.00797[(b)]	0.094				
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.0106	0.114				
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.00797[(b)]	0.094				

Notes:

(a) 57 EFPY surface fluence values are documented in Table 2-14.

(b) The outlet nozzle materials do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY, therefore, per NRC RIS 2014-11 [14], neutron irradiation embrittlement need not be considered for the nozzle materials herein. However, the results are included for information

(c) The inlet and outlet nozzle forgings and welds are conservatively evaluated without consideration for the attenuation of the fluence through the vessel thickness.

(d) The nozzle forging fluence values are conservatively set equal to the fluence at the respective nozzle welds.

Table 7-3 Adjusted Reference Temperature Evaluation for the Braidwood Unit 1 Reactor Vessel Beltline Materials through 57 EFPY at the 1/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF(a) (°F)	1/4T Fluence(b) (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	Margin (°F)	ART(e) (°F)
Nozzle Shell Forging [5P-7016]	1.1	26	0.679	0.891	10	23.2	0	11.6	23.2	56.3
Intermediate Shell Forging [49D383/49C344]-1-1	1.1	31	1.93	1.180	-30	36.6	0	17.0	34.0	40.6
Lower Shell Forging [49D867/49C813]-1-1	1.1	31	1.89	1.174	-20	36.4	0	17.0	34.0	50.4
Lower Shell Forging Using Credible Braidwood Unit 1 Surveillance Data	2.1	27.8	1.89	1.174	-20	32.6	0	8.5	17.0	29.6
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645 [Heat # H4498, Linde 80 flux type, Lot # 0261]	1.1	54	0.679	0.891	-25	48.1	0	24.1	48.1	71.3
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 [Heat #442011, Linde 80 flux type, Lot # 8061]	1.1	41	1.84	1.167	40	47.8	0	23.9	47.8	135.7
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	1.84	1.167	40	36.4	0	14.0	28.0	104.4

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix D, the lower shell forging material and the intermediate to lower shell weld material surveillance data were determined to be credible. Therefore, per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 data and σ_Δ = 14°F for Position 2 1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-4 Adjusted Reference Temperature Evaluation for the Braidwood Unit 1 Reactor Vessel Beltline Materials through 57 EFPY at the 3/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART[(e)] (°F)
Nozzle Shell Forging [5P-7016]	1.1	26	0.245	0.619	10	16.1	0	8.0	16.1	42.2
Intermediate Shell Forging [49D383/49C344]-1-1	1.1	31	0.697	0.899	-30	27.9	0	13.9	27.9	25.7
Lower Shell Forging [49D867/49C813]-1-1	1.1	31	0.680	0.892	-20	27.6	0	13.8	27.6	35.3
Lower Shell Forging Using Credible Braidwood Unit 1 Surveillance Data	2.1	27.8	0.680	0.892	-20	24.8	0	8.5	17.0	21.8
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645 [Heat # H4498, Linde 80 flux type, Lot # 0261]	1.1	54	0.245	0.619	-25	33.4	0	16.7	33.4	41.8
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 [Heat #442011, Linde 80 flux type, Lot # 8061]	1.1	41	0.663	0.885	40	36.3	0	18.1	36.3	112.5
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	0.663	0.885	40	27.6	0	13.8	27.6	95.2

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix D, the lower shell forging material and the intermediate to lower shell weld material surveillance data were determined to be credible. Therefore, per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8 5°F for Position 2.1 with credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 data and σ_Δ = 14°F for Position 2 1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-5 Adjusted Reference Temperature Evaluation for the Braidwood Unit 1 Reactor Vessel Extended Beltline Materials through 57 EFPY at the Surface Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART[g] (°F)
Inlet Nozzle 01-001 [Heat # 21-3257]	1.1	58	0.0124	0.127	-20	7.3	0	3.7	7.3	-5.3
Inlet Nozzle 01-002 [Heat # 21-3257]	1.1	58	0.0124	0.127	-10	7.3	0	3.7	7.3	4.7
Inlet Nozzle 02-001 [Heat # 22-3313]	1.1	44	0.0124	0.127	-10	5.6	0	2.8	5.6	1.1
Inlet Nozzle 02-002 [Heat # 22-3313]	1.1	44	0.0124	0.127	0	5.6	0	2.8	5.6	11.1
Outlet Nozzle 01-001 [Heat # 22-3025]	1.1	96	0.00938	0.105	-10	10.1	0	5.0	10.1	10.2
Outlet Nozzle 01-003 [Heat # 11-5226]	1.1	58	0.00938	0.105	-10	6.1	0	3.0	6.1	2.2
Outlet Nozzle 02-001 [Heat # 4-3329]	1.1	51	0.00938	0.105	-20	5.4	0	2.7	5.4	-9.3
Outlet Nozzle 02-002 [Heat # 4-3329]	1.1	51	0.00938	0.105	-20	5.4	0	2.7	5.4	-9.3
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598 [Heat # 41403, Linde 80 flux type, Lot # 0852]	1.1	185.6	0.0124	0.127	-48.6	23.5	18.0[d]	28.0[f]	66.6	41.5
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598 [Heat # 41403, Linde 80 flux type, Lot # 0852]	1.1	185.6	0.00938	0.105	-48.6	19.5	18.0[d]	28.0[f]	66.6	37.5
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-588 [Heat # 41403, Linde 80 flux type, Lot # 8119]	1.1	195.7	0.00938	0.105	-48.6	20.5	18.0[d]	28.0[f]	66.6	38.5
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-579 [Heat # 442010, Linde 80 flux type, Lot # 8119]	1.1	181	0.00938	0.105	-48.6	19.0	18.0[d]	28.0[f]	66.6	37.0

Notes:

(a) Values are taken from Table 5-4

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1

(d) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(u)}$ value.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1, and the weld metal σ_Δ = 28°F for Position 1.1 However, σ_Δ need not exceed 0 5*ΔRT_{NDT} per regulatory guidance in [1].

(f) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A

(g) ART values are calculated in accordance with [1].

Table 7-6 Adjusted Reference Temperature Evaluation for the Braidwood Unit 2 Reactor Vessel Beltline Materials through 57 EFPY at the 1/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART[(e)] (°F)
Nozzle Shell Forging [5P-7056]	1.1	26	0.597	0.856	30	22.2	0	11.1	22.2	74.5
Intermediate Shell Forging [49D963/49C904]-1-1	1.1	20	1.77	1.157	-30	23.1	0	11.6	23.1	16.3
Lower Shell Forging [50D102/50C97]-1-1	1.1	37	1.82	1.164	-30	43.1	0	17.0	34.0	47.1
Lower Shell Forging Using Non-Credible Braidwood Unit 2 Surveillance Data	2.1	20.9	1.82	1.164	-30	24.3	0	12.2	24.3	18.7
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645 [Heat # H4498, Linde 80 flux type, Lot # 0261]	1.1	54	0.597	0.856	-25	46.2	0	23.1	46.2	67.4
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 [Heat # 442011, Linde 80 flux type, Lot # 8061]	1.1	41	1.74	1.153	40	47.3	0	23.6	47.3	134.5
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data [Heat # 442011, Linde 80 flux type, Lot # 0344]	2.1	31 2	1.74	1.153	40	36.0	0	14.0	28.0	104.0

Notes:

(a) Values are taken from Table 5-5.

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the intermediate to lower shell weld material surveillance data was determined to be credible. Per the guidance of regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1 1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 and σ_Δ = 14°F for Position 2.1 with credible surveillance data However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1]

Table 7-7 Adjusted Reference Temperature Evaluation for the Braidwood Unit 2 Reactor Vessel Beltline Materials through 57 EFPY at the 3/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART[(e)] (°F)
Nozzle Shell Forging [5P-7056]	1.1	26	0.215	0.587	30	15.3	0	7.6	15.3	60.5
Intermediate Shell Forging [49D963/49C904]-1-1	1.1	20	0.639	0.874	-30	17.5	0	8.7	17.5	5.0
Lower Shell Forging [50D102/50C97]-1-1	1.1	37	0.656	0.882	-30	32.6	0	16.3	32.6	35.3
Lower Shell Forging Using Non-Credible Braidwood Unit 2 Surveillance Data	2.1	20.9	0.656	0.882	-30	18.4	0	9.2	18.4	6.9
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645 [Heat # H4498, Linde 80 flux type, Lot # 0261]	1.1	54	0.215	0.587	-25	31.7	0	15.9	31.7	38.4
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 [Heat # 442011, Linde 80 flux type, Lot # 8061]	1.1	41	0.628	0.870	40	35.7	0	17.8	35.7	111.3
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data [Heat # 442011, Linde 80 flux type, Lot # 0344]	2.1	31.2	0.628	0.870	40	27.1	0	13.6	27.1	94.3

Notes:

(a) Values are taken from Table 5-5.

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the intermediate to lower shell weld material surveillance data was determined to be credible. Per the guidance of regulatory Guide 1 99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and for Position 2 1 with non-credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-8 Adjusted Reference Temperature Evaluation for the Braidwood Unit 2 Reactor Vessel Extended Beltline Materials through 57 EFPY at the Surface Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	Surface Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART[(h)] (°F)
Inlet Nozzle 01-001 [Heat # 41-5414]	1.1	44	0.0106	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 01-002 [Heat # 41-5414]	1.1	44	0.0106	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 02-001 [Heat # 42-5417]	1.1	58	0.0106	0.114	-10	6.6	0	3.3	6.6	3.2
Inlet Nozzle 02-002 [Heat # 42-5417]	1.1	58	0.0106	0.114	-10	6.6	0	3.3	6.6	3.2
Outlet Nozzle 01-002 [Heat # 11-5266]	1.1	58	0.00797	0.094	10	5.4	0	2.7	5.4	20.9
Outlet Nozzle 01-003 [Heat # 11-5226]	1.1	58	0.00797	0.094	-10	5.4	0	2.7	5.4	0.9
Outlet Nozzle 02-001 [Heat # 4-3481]	1.1	44	0.00797	0.094	-10	4.1	0	2.1	4.1	-1.7
Outlet Nozzle 02-002 [Heat # 4-3502]	1.1	58	0.00797	0.094	-10	5.4	0	2.7	5.4	0.9
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654 [Heat # 41404, Linde 80 flux type, Lot # 0261]	1.1	167.0[(f)]	0.0106	0.114	-48.6	19.0	18.0[(d)]	28.0[(g)]	66.6	37.0
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654 [Heat # 41404, Linde 80 flux type, Lot # 0261]	1.1	167.0[(f)]	0.00797	0.094	-48.6	15.7	18.0[(d)]	28.0[(g)]	66.6	33.6

Notes:

(a) Values are taken from Table 5-5.
(b) Values are taken from Table 7-2.
(c) Values are taken from Table 3-2.
(d) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(u)}$ value.
(e) Per the guidance of regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1, and the weld metal σ_Δ = 28°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].
(f) Value is required minimum per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.
(g) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.
(h) ART values are calculated in accordance with [1].

Table 7-9 Limiting ART Values for Braidwood Units 1 and 2 at 57 EFPY[a]

	Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)	Limiting Material
"Axial Flaw" Method	74.5	60.5	Braidwood Unit 2 Nozzle Shell Forging 5P-7056
"Circumferential Flaw" Method	104.4[b]	95.2[b]	Braidwood Unit 1 Intermediate to Lower Shell Forging Circumferential Weld Seam (Heat # 442011) using Credible Braidwood Unit 1 Surveillance Data

Note:

(a) Values are the limiting values from Tables 7-3 through 7-8 For the materials listed in Table 7-3 through Table 7-8 circumferential flaws are considered in the circumferential weld materials and axial flaws are considered in all other materials.

(b) For the Intermediate to Lower Shell Weld materials, the ART values calculated using Position 2.1 CFs are used instead of the Position 1 1 results because credible surveillance data is available.

8 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 cylindrical beltline region using the methods discussed in Sections 6 and 7 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [2].

The highest ART values for Braidwood Units 1 and 2 correspond to the Braidwood Unit 1 Intermediate to Lower Shell Forging Circumferential Weld (Heat # 442011). However, since this material is a "Circumferential Flaw" material, the applied membrane (pressure) stress and resulting stress intensity factor at the postulated flaw location are much lower than for the most limiting "Axial Flaw" material. Consequently, this material does not produce the most limiting P-T limit curves. The most limiting P-T limit curves for Braidwood Units 1 and 2 are produced by using the "Axial Flaw" methodology and the limiting "Axial Flaw" material ART values. Thus, the limiting ART values for Braidwood Units 1 and 2 used in the generation of the P-T limit curves are based on the Braidwood Unit 2 Nozzle Shell Forging (Position 1.1) from Table 7-9. For P-T limit curve development, the limiting ART values are conservatively rounded up as shown below in Table 8-1.

Table 8-1 ART Values to be used in P-T Limit Curves Development for Braidwood Units 1 and 2 at 57 EFPY(a)

Limiting Material	Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)
Braidwood Unit 2 Nozzle Shell Forging 5P-7056	75	61

Note

(a) Values correspond to the limiting "Axial Flaw" Method ART values in Table 7-9 rounded up to the nearest whole number

Figure 8-1 presents the limiting heatup curves without margins for possible instrumentation errors using a heatup rate of 100°F/hr applicable for 57 EFPY, without the flange requirements and using the "Axial Flaw" methodology. Figure 8-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0°, 25°, 50°, and 100°F/hr applicable for 57 EFPY, without the flange requirements and using the "Axial Flaw" methodology. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G. As discussed in Section 6, the use of the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values produce the most limiting P-T limit curves for Braidwood Units 1 and 2. The exclusion of the 10 CFR 50, Appendix G flange requirements is justified by WCAP-16143-P and approved by the NRC in [26]. The NRC originally approved the exemption from the Appendix G flange requirements in [26] based on Revision 0 of WCAP-16143-P. Reference [44] revised the exemption to account for Revision 1 of WCAP-16143-P, which considers a 53 stud configuration.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 8-1 and 8-2. 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 P-T combinations are to the right of the criticality limit line shown in Figure 8-1 (heatup curve only). 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 the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

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

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 6-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$ [see page 6-1 Equation (1)],

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies P-T limits for core operation in order to provide additional margin during actual power production. The P-T limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding P-T curve for heatup and cooldown calculated as described in Section 6 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the Braidwood Units 1 and 2 reactor vessels at 57 EFPY is 135°F; this temperature value is calculated based on Equation (13). The vertical line drawn from these points on the P-T curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 8-1 and 8-2 define all of the above limits for ensuring prevention of non-ductile failure for the Braidwood Units 1 and 2 reactor vessels for 57 EFPY without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 8-1 and 8-2 are presented in Tables 8-2 and 8-3. Vacuum refill limits for the Reactor Coolant System (RCS) are displayed on Figures 8-1 and 8-2 by showing a minimum pressure of 0 psia. This is consistent with the current Braidwood Units 1 and 2 Pressure and Temperature Limit Reports (PTLRs) [27 and 28].

As discussed in Appendix B, the P-T limits developed for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for Braidwood Units 1 and 2 at 57 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Braidwood Unit 2 Nozzle Shell Forging 5P-7056 using Regulatory Guide 1.99 Position 1.1 data

LIMITING ART VALUES AT 57 EFPY: 1/4T, 75°F (Axial Flaw)
3/4T, 61°F (Axial Flaw)

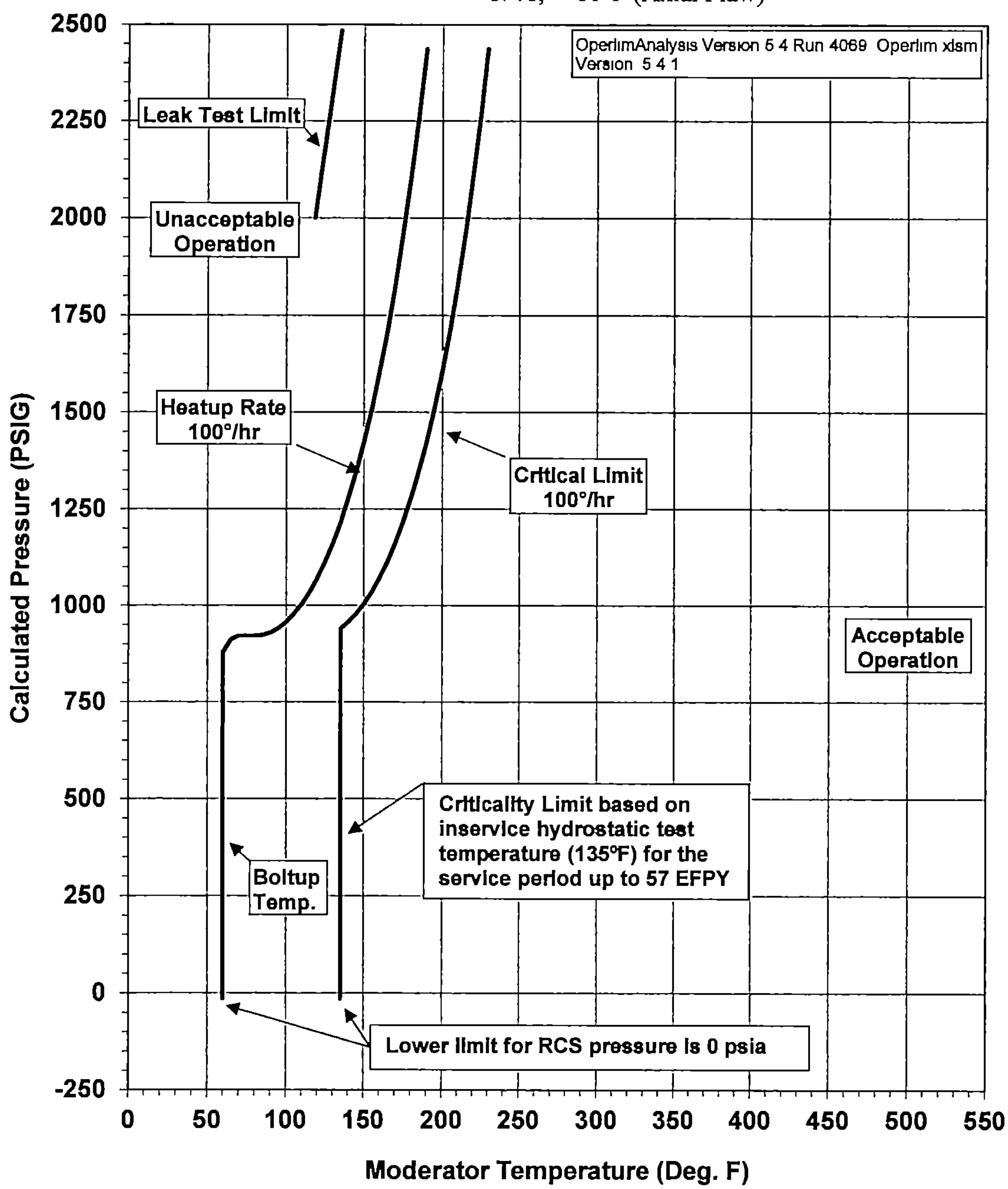

Figure 8-1 Braidwood Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for 57 EFPY (without Flange Requirements and <u>without</u> Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Braidwood Unit 2 Nozzle Shell Forging 5P-7056 using Regulatory Guide 1.99 Position 1.1 data

LIMITING ART VALUES AT 57 EFPY: 1/4T, 75°F (Axial Flaw)
3/4T, 61°F (Axial Flaw)

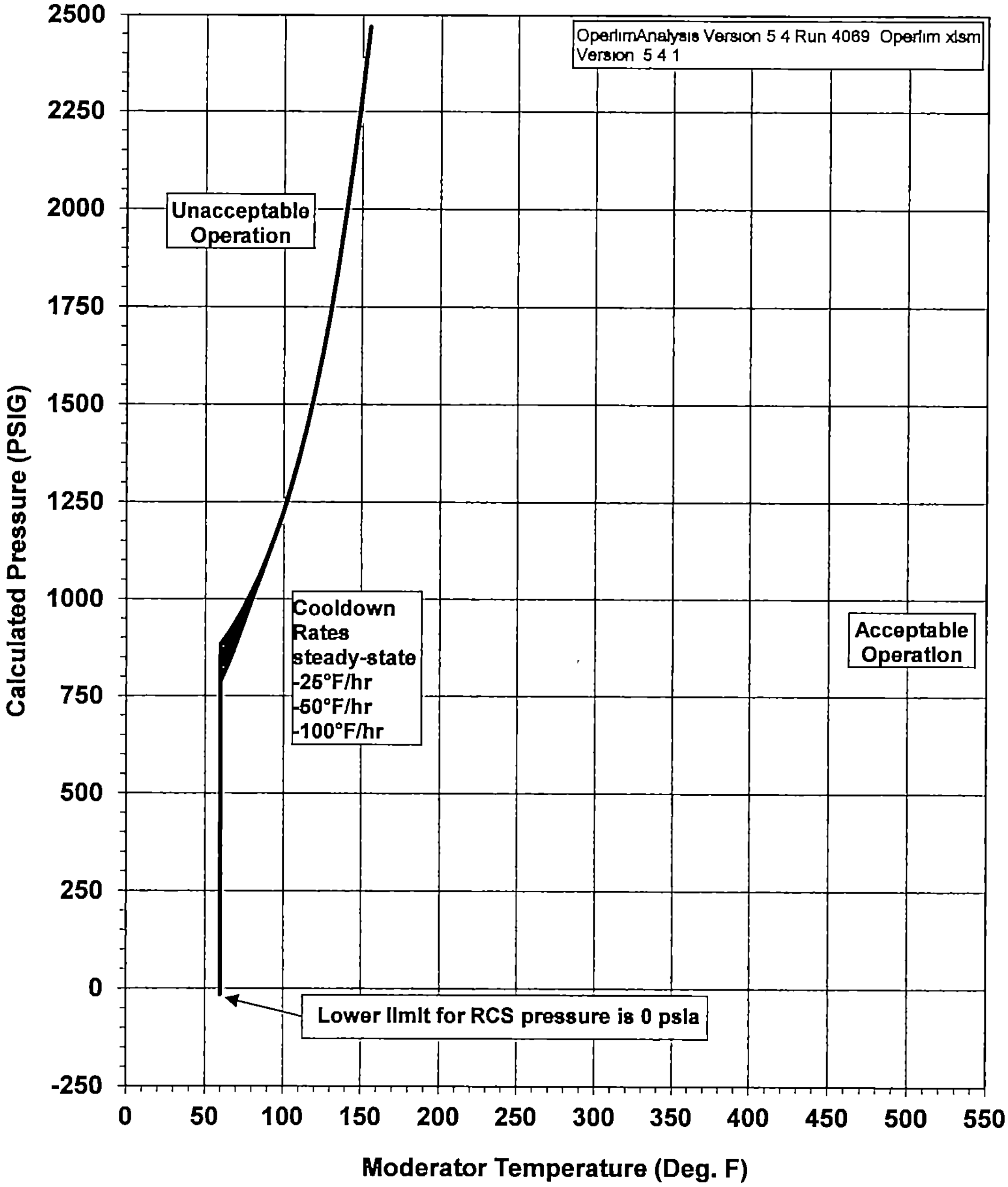

Figure 8-2 Braidwood Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, and 100°F/hr) Applicable for 57 EFPY (without Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-2 Braidwood Units 1 and 2 57 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/o Flange Requirements, and w/o Margins for Instrumentation Errors)

100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)
60	Note (a)	135	Note (a)
60	879	135	940
65	912	140	957
70	921	145	978
75	921	150	1004
80	921	155	1035
85	923	160	1071
90	929	165	1113
95	940	170	1161
100	957	175	1215
105	978	180	1277
110	1004	185	1345
115	1035	190	1422
120	1071	195	1508
125	1113	200	1604
130	1161	205	1710
135	1215	210	1827
140	1277	215	1957
145	1345	220	2102
150	1422	225	2261
155	1508	230	2437
160	1604	-	-
165	1710	-	-
170	1827	-	-
175	1957	-	-
180	2102	-	-
185	2261	-	-
190	2437	-	-

Leak Test Limit	
T (°F)	P (psig)
118	2000
135	2485

Note.

(a) The minimum acceptable pressure is 0 psia.

Table 8-3 Braidwood Units 1 and 2 57 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/o Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		25°F/hr Cooldown		50°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note (a)	60	Note (a)	60	Note (a)	60	Note (a)
60	882	60	854	60	828	60	788
65	912	65	886	65	864	65	835
70	944	70	923	70	905	70	887
75	980	75	963	75	950	75	944
80	1020	80	1007	80	1000	80	1000
85	1063	85	1056	85	1055	85	1055
90	1112	90	1110	90	1110	90	1110
95	1165	95	1165	95	1165	95	1165
100	1224	100	1224	100	1224	100	1224
105	1290	105	1290	105	1290	105	1290
110	1362	110	1362	110	1362	110	1362
115	1442	115	1442	115	1442	115	1442
120	1530	120	1530	120	1530	120	1530
125	1627	125	1627	125	1627	125	1627
130	1735	130	1735	130	1735	130	1735
135	1854	135	1854	135	1854	135	1854
140	1986	140	1986	140	1986	140	1986
145	2131	145	2131	145	2131	145	2131
150	2292	150	2292	150	2292	150	2292
155	2469	155	2469	155	2469	155	2469

Note
(a) The minimum acceptable pressure is 0 psia.

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APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) and vessel temperatures for the maximum heatup and cooldown rates at 57 EFPY for Braidwood Units 1 and 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 88.75 inches
- 3/4T Radius = 93.00 inches

Table A-1 K_{It} and Vessel Temperature Values for Braidwood Units 1 and 2 at 57 EFPY 100°F/hr Heatup Curves (w/o Flange Requirements and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.008	-0.994	55.046	0.477
65	58.618	-2.441	55.313	1.442
70	61.705	-3.684	56.014	2.421
75	65.007	-4.860	57.197	3.337
80	68.582	-5.872	58.805	4.153
85	72.269	-6.794	60.798	4.881
90	76.140	-7.591	63.135	5.523
95	80.110	-8.318	65.772	6.097
100	84.212	-8.952	68.677	6.604
105	88.400	-9.527	71.814	7.056
110	92.684	-10.031	75.155	7.459
115	97.039	-10.491	78.674	7.820
120	101.464	-10.896	82.350	8.144
125	105.948	-11.267	86.163	8 435
130	110.483	-11.595	90.098	8.697
135	115.065	-11.897	94.140	8.934
140	119.685	-12.166	98.275	9.147
145	124.343	-12.415	102.494	9.342
150	129.029	-12.637	106.785	9.518
155	133.746	-12.845	111.140	9.680
160	138.484	-13.032	115.552	9.828
165	143.247	-13.207	120.013	9.964
170	148.025	-13.366	124.518	10.089
175	152.822	-13.516	129.062	10.206
180	157.632	-13.653	133.640	10 314
185	162.457	-13.784	138.248	10.415
190	167.291	-13.905	142.883	10.509
195	172.137	-14.020	147.541	10.598
200	176.990	-14.128	152.219	10.682
205	181.852	-14.232	156.916	10.762
210	186.720	-14.329	161.629	10.838

Table A-2 K_{It} and Vessel Temperature Values for Braidwood Units 1 and 2 at 57 EFPY 100°F/hr Cooldown Curves (w/o Flange Requirements and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	236.227	16.489
205	231.144	16.422
200	226.059	16.356
195	220.975	16.289
190	215.891	16.223
185	210.806	16.155
180	205.721	16.089
175	200.636	16.021
170	195.551	15.955
165	190.466	15.887
160	185.381	15.820
155	180.295	15.753
150	175.210	15.686
145	170.125	15.619
140	165.040	15.552
135	159.955	15.485
130	154.870	15.419
125	149.785	15.352
120	144.700	15.285
115	139.615	15.219
110	134.530	15.152
105	129.445	15.086
100	124.361	15.020
95	119.276	14.954
90	114.192	14.888
85	109.108	14.822
80	104.024	14.757
75	98.940	14.691
70	93.856	14.626
65	88.772	14.560
60	83.690	14.495

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES

As described in NRC Regulatory Issue Summary (RIS) 2014-11 [14], reactor vessel non-beltline materials may define pressure-temperature (P-T) limit curves that are more limiting than those calculated for the reactor vessel cylindrical shell beltline materials. Reactor vessel nozzles, penetrations, and other discontinuities have complex geometries that can exhibit significantly higher stresses than those for the reactor vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperatures (RT_{NDT}) for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries.

The methodology contained in WCAP-14040-A, Revision 4 [2] was used in the main body of this report to develop P-T limit curves for the limiting Braidwood Units 1 and 2 cylindrical shell beltline material; however, Reference [2] does not consider ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. Due to the geometric discontinuity, the inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in Appendix B herein. P-T limit curves are determined for the reactor vessel nozzle corner region for Braidwood Units 1 and 2 and compared to the P-T limit curves for the reactor vessel traditional beltline region in order to determine if the nozzles can be more limiting than the reactor vessel beltline as the plant ages and the vessel accumulates more neutron fluence. The increase in neutron fluence as the plant ages causes a concern for embrittlement of the reactor vessel above the beltline region. Therefore, the P-T limit curves are developed for the nozzle inside corner region since the geometric discontinuity results in high stresses due to internal pressure and the cooldown transient. The cooldown transient is analyzed as it results in tensile stresses at the inside surface of the nozzle corner.

A flaw is postulated at the inside surface of the reactor vessel nozzle corner, and stress intensity factors are determined based on the rounded curvature of the nozzle geometry. The allowable pressure is then calculated based on the fracture toughness of the nozzle material and the stress intensity factors for the postulated flaw.

B.1 CALCULATION OF ADJUSTED REFERENCE TEMPERATURES

The fracture toughness (K_{Ic}) used for the inlet and outlet nozzle material is defined in Appendix G of the Section XI ASME Code, as discussed in Section 6 of this report. The K_{Ic} fracture toughness curve is dependent on the Adjusted Reference Temperature (ART) value for irradiated materials. The ART values for the inlet and outlet nozzle materials are determined using the methodology contained in Regulatory Guide 1.99, Revision 2 [1], which is described in Section 7 of this report, as well as weight percent (wt. %) copper (Cu) values, wt. % nickel (Ni) values, initial RT_{NDT} values, and projected neutron fluence as inputs. The material properties for each of the reactor vessel inlet and outlet nozzle forging materials are documented in Table 3-1 and Table 3-2. A summary of the limiting inlet and outlet nozzle ART values used in the nozzle P-T limit curves analysis for Braidwood Units 1 and 2 is presented in Table B-1.

Nozzle Material Properties

The Braidwood Units 1 and 2 nozzle material properties are provided in Table 3-1 and Table 3-2, respectively. Cu and Ni weight percent (wt. %) values were obtained from the Braidwood Units 1 and 2 CMTRs for each of the Braidwood Units 1 and 2 reactor vessel inlet and outlet nozzles.

ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3) [15] was used to determine all initial RT_{NDT} values for the inlet and outlet nozzles. All initial RT_{NDT} values are consistent with those reported in WCAP-17607-NP [16]. The weak-direction Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was identified in CMTRs for Braidwood Units 1 and 2, and these weak-direction results were utilized in determining initial RT_{NDT} values.

Nozzle Calculated Neutron Fluence Values

The maximum fast neutron (E > 1.0 MeV) exposure of the Braidwood Units 1 and 2 reactor vessel materials is discussed in Section 2 of this report. The fluence values used in the inlet and outlet nozzle ART calculations were calculated at the lowest extent of the nozzles (i.e., the nozzle to nozzle shell weld locations) and are therefore taken at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle, for conservatism.

Per Table 2-8 and Table 2-14, the inlet nozzles are determined to receive a projected maximum fluence of 1.24×10^{17} n/cm^2 (E > 1.0 MeV) and the outlet nozzle are determined to receive a projected maximum fluence of 0.938×10^{17} n/cm^2 (E > 1.0 MeV) at the lowest extent of the nozzles at 57 EFPY. Thus, the maximum neutron fluence values for the outlet nozzle materials are not projected to exceed a fluence of 1×10^{17} n/cm^2 at 57 EFPY, but the inlet nozzle materials are projected to exceed the threshold. Per NRC RIS 2014-11 [14], embrittlement of reactor vessel materials, with projected fluence values less than 1×10^{17} n/cm^2, does not need to be considered. Thus, embrittlement of the outlet nozzle materials does not need to be considered. In Section 7, embrittlement of the outlet nozzle materials is conservatively considered for comparison with the beltline ART values; however, in this appendix, embrittlement of the outlet nozzle materials is not considered, consistent with RIS 2014-11. In addition, the fluence values utilized for the nozzle evaluations conservatively do not consider attenuation through the reactor vessel thickness.

The limiting nozzle ART values used for determination of the nozzle P-T limit curves are summarized in Table B-1.

Table B-1 Summary of the Limiting ART Values for the Braidwood Units 1 and 2 Inlet and Outlet Nozzle Materials

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
57	Braidwood Unit 1 Inlet Nozzles	11.1[(a)]
	Braidwood Unit 1 Outlet Nozzles	-10[(b)]
	Braidwood Unit 2 Inlet Nozzles	3.2[(c)]
	Braidwood Unit 2 Outlet Nozzles	10[(d)]

Notes:

(a) Limiting value of all Braidwood Unit 1 Inlet Nozzles from Table 7-5. Limiting value corresponds to Braidwood Unit 1 Inlet Nozzle 02-002.

(b) Limiting value of all Braidwood Unit 1 Outlet Nozzle initial RT_{NDT} values from Table 7-5. Limiting value corresponds to Braidwood Unit 1 Outlet Nozzles 01-001 and 01-003. Since the projected fluence is less than 1 x 10^{17} n/cm^2 at 57 EFPY for the outlet nozzles, embrittlement need not be considered consistent with RIS 2014-11.

(c) Limiting value of all Braidwood Unit 2 Inlet Nozzles from Table 7-8. Limiting value corresponds to Braidwood Unit 2 Inlet Nozzles 02-001 and 02-002.

(d) Limiting value of all Braidwood Unit 2 Outlet Nozzle initial RT_{NDT} values from Table 7-8 Limiting value corresponds to Braidwood Unit 2 Outlet Nozzle 01-002 Since the projected fluence is less than 1 x 10^{17} n/cm^2 at 57 EFPY for the outlet nozzles, embrittlement need not be considered consistent with RIS 2014-11.

B.2 NOZZLE COOLDOWN PRESSURE-TEMPERATURE LIMITS

Allowable pressures are determined for a given temperature based on the fracture toughness of the limiting nozzle material along with the appropriate pressure and thermal stress intensity factors. The Braidwood Units 1 and 2 nozzle fracture toughness used to determine the P-T limits is calculated using the limiting inlet and outlet nozzle ART values from Table B-1. The stress intensity factor correlations used for the nozzle corners are provided in ORNL study, ORNL/TM-2010/246 [19], and are consistent with ASME PVP2011-57015 [20]. The methodology includes postulating an inside surface nozzle corner flaw, and calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

For the inlet and outlet nozzles, a 3 inch flaw was used for the generation of the nozzle P-T limit curves per guidance from Article G-2120 of the ASME Section XI Code [3]. Article G-2223 of the ASME Section XI Code states that a flaw depth smaller than 1/4T can be used. Therefore, in lieu of using a 1/4T circular corner flaw depth for the limiting inlet and outlet nozzles, Article G-2120 states that for sections greater than 12 inch thick, the postulated flaw depth based on the 12 inch section may be used. Thus, a 1/4T flaw for a 12 inch section is used (1/4 * 12" = 3") for the generation of the nozzle P-T limit curves for the inlet and outlet nozzles.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

where,

K_I = stress intensity factor for a circular corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with a 3 inch flaw

The Braidwood Units 1 & 2 reactor vessel inlet nozzle P-T limit curves are shown in Figure B-1, while the outlet nozzle P-T limit curves are shown in Figure B-2. The nozzle P-T limit curves are based on the

stress intensity factor expression discussed above; also shown in these figures are the traditional beltline cooldown P-T limit curves from Figure 8-2. The nozzle P-T limit curves are provided for a cooldown rate of 100°F/hr, along with a steady-state curve.

An outside surface flaw in the nozzle was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is also not provided since it would be less limiting than the cooldown nozzle P-T limit curve in Figures B-1 and B-2 for an inside surface flaw. Additionally, the cooldown transient is more limiting than the heatup transient since it results in tensile stresses (rather than compressive stresses) at the inside surface of the nozzle corner.

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical shell beltline curves. Therefore, the P-T limits provided in Section 8 for 57 EFPY remain limiting for the beltline and non-beltline reactor vessel components.

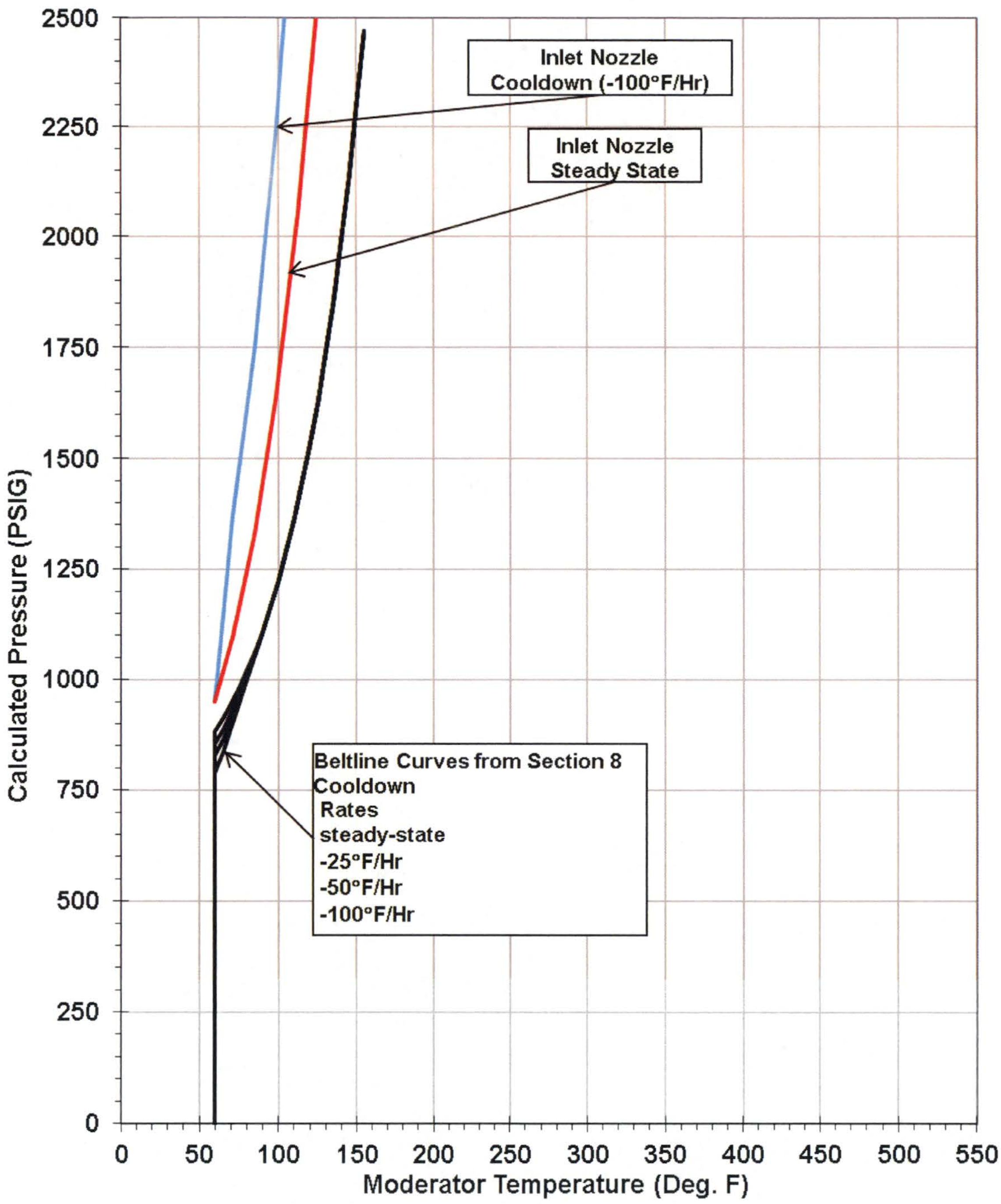

Figure B-1 Comparison of Braidwood Units 1 and 2 57 EFPY Beltline P-T Limits to 57 EFPY Limiting Inlet Nozzle P-T Limits, without Margins for Instrumentation Error

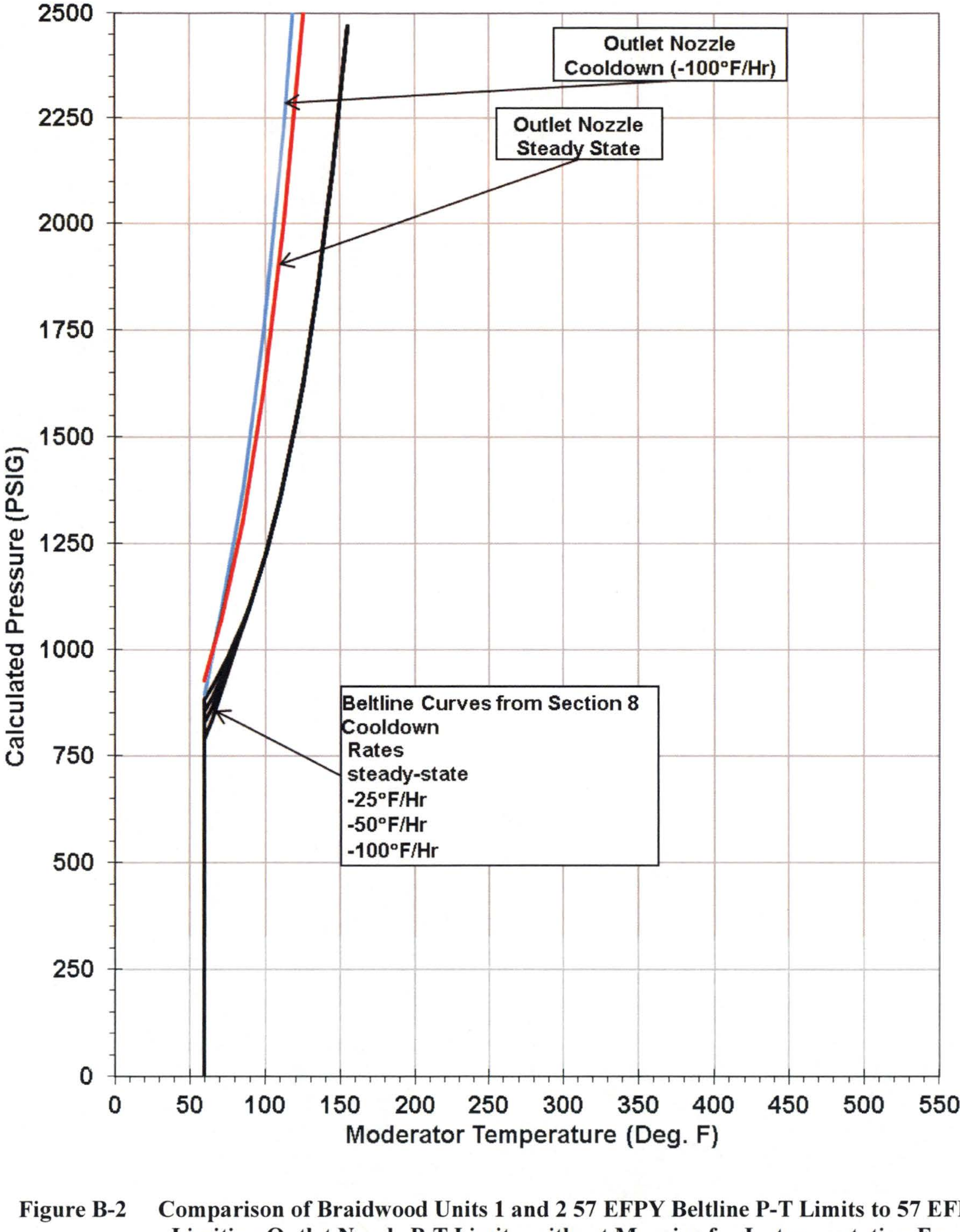

Figure B-2 Comparison of Braidwood Units 1 and 2 57 EFPY Beltline P-T Limits to 57 EFPY Limiting Outlet Nozzle P-T Limits, without Margins for Instrumentation Error

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G [4] requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature (LST) requirement for all RCPB components, which is specified in NB-2332(b) and NB-3211 of the ASME Code, Section III [15], is the relevant requirement that would affect the P-T limits. This requirement is applicable to ferritic materials outside of the RV with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps and valves [15].

The Braidwood Units 1 and 2 reactor coolant systems do not have ferritic materials in the Class 1 piping, pumps, and valves (fabricated instead with stainless steel). Therefore, the LST requirements of the ASME Code, Section III, NB-2332(b) and NB-3211 [15] for these components do not need to be considered.

RIS 2014-11 [14] also addresses other ferritic components of the reactor coolant system relative to P-T limit, and states the following:

> *As specified in Sections I and IV.A of 10 CFR Part 50, Appendix G, ferritic RCPB components outside of the reactor vessel must meet the applicable requirements of ASME Code, Section III, "Rules for Construction of Nuclear Facility Components."*

The other ferritic RCPB components that are not part of the RV beltline or extended beltline for Braidwood Units 1 and 2 consist of the RV closure head, steam generators, and pressurizer. The Braidwood Units 1 and 2 primary system components are analyzed to the following ASME Code Section III Editions, and met all applicable requirements at the time of construction. Therefore, no further consideration of these components is necessary.

- Reactor Vessel Closure Head – ASME Code Section III 1971 Edition through Summer 1973 Addenda. This component was previously analyzed in WCAP-16143-P [5].
- Replacement Steam Generator for Unit 1 – ASME Code Section III 1986 Edition
- Steam Generator for Unit 2 – ASME Code Section III 1971 Edition through Summer 1972 and Winter 1974 Addenda
- Pressurizer – ASME Code Section III 1971 Edition through Summer 1973 Addenda

APPENDIX D BRAIDWOOD UNITS 1 AND 2 SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

Regulatory Guide 1.99, Revision 2 [1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position 2.1 of [1], describes the method for calculating the adjusted reference temperature of reactor vessel beltline materials using surveillance capsule data. The methods of Position 2.1 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been four surveillance capsules removed and tested from each of the Braidwood Units 1 and 2 reactor vessels. To use the surveillance data, the data must be shown to be credible. In accordance with [1], the credibility of the surveillance data will be judged based on five criteria.

The purpose of this evaluation is to apply the credibility requirements of [1], to the Braidwood Units 1 and 2 reactor vessel surveillance data, including fluence values updated in Section 2, to determine if the surveillance data is credible.

D.1 BRAIDWOOD UNIT 1 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [4], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Braidwood Unit 1 reactor vessel beltline region consists of the following materials:

1. Nozzle Shell Forging 5P-7016
2. Intermediate Shell Forging [49D383/49C344]-1-1
3. Lower Shell Forging [49D867/49C813]-1-1
4. Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld Seam WF-645 (Weld Wire Heat # H4498, Linde 80 Flux Type, Flux Lot # 0261)
5. Intermediate Shell Forging to Lower Shell Forging Circumferential Weld Seams WF-562 (Weld Wire Heat # 442011, Linde 80 Flux Type, Flux Lot # 8061)

The Braidwood Unit 1 surveillance program utilizes tangential and axial test specimens from the Lower Shell Forging. The surveillance weld metal was fabricated with weld wire Heat # 442011, Flux Type Linde 80, Lot # 8061.

At the time when the Braidwood Unit 1 surveillance program material was selected, it was believed that copper and phosphorus were the elements most important to the embrittlement of reactor vessel steels and the Nozzle Shell Forging was not considered a "beltline" material. The Lower Shell Forging had an initial RT_{NDT} that was 10°F higher than the Intermediate Shell Forging initial RT_{NDT}. In addition, the Lower and Intermediate Shell Forgings had essentially the same copper and phosphorus content. Based on this comparison of the beltline forging materials (Intermediate and Lower Shell Forgings), the Lower Shell Forging was chosen for the surveillance program.

Weld seam WF-562, on the other hand, was considered the only weld in the beltline region and therefore was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 442011), the same type of flux (Linde 80), and the same flux lot (# 8061) as the Intermediate to Lower Shell Forging Circumferential Weld Seam.

Therefore, the materials selected for use in the Braidwood Unit 1 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

Based on the discussion above, Criterion 1 is met for the Braidwood Unit 1 surveillance program.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-18092, Revision 1 [17].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Braidwood Unit 1 surveillance materials unambiguously.

Hence, the Braidwood Unit 1 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [21].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the forging.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of [1]. Braidwood Unit 1 has one circumferential weld that will be evaluated for credibility. This weld is Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 and is fabricated from weld wire Heat # 442011, Linde 80 type flux, Lot # 8061. This weld metal heat is contained in both the Braidwood Unit 1 and the Braidwood Unit 2 surveillance programs. Since the welds in question utilize data from other surveillance programs, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [22]. At this meeting the NRC presented five cases. Of the five cases, Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situation listed above for Braidwood Unit 1 surveillance weld metal. Note that for the forging materials, the straightforward method in [1] will be followed.

Following the NRC Case 4 guidelines, only the Braidwood Unit 1 data will be evaluated first. Table D-1 provides the calculation of the interim CF for Braidwood Unit 1. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this

calculation is based on the actual surveillance weld metal measured shift values. In addition, only Braidwood Unit 1 data is being considered; therefore, no temperature adjustment is required.

Table D-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Braidwood Unit 1 Surveillance Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.388	0.738	5.6	4.13	0.54
	X	1.17	1.044	37.9	39.56	1.09
	W	1.98	1.186	24.0	28.48	1.41
	V	3.71	1.340	51.3	68.73	1.79
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.388	0.738	-15.8[d]	-11.66	0.54
	X	1.17	1.044	29.3	30.58	1.09
	W	1.98	1.186	37.1	44.02	1.41
	V	3.71	1.340	39.7	53.19	1.79
				SUM:	257.03	9.67
	$CF_{[49D867/49C813]-1-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (257.03) \div (9.67) =$ **26.6°F**					
Braidwood Unit 1 Surveillance Weld Material (Heat # 442011)	U	0.388	0.738	17.4	12.84	0.54
	X	1.17	1.044	29.8	31.11	1.09
	W	1.98	1.186	49.0	58.14	1.41
	V	3.71	1.340	62.8	84.13	1.79
				SUM:	186.22	4.84
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (186.22) \div (4.84) =$ **38.5°F**					

Notes

(a) Taken from Table 5-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$

(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17]

(d) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-2.

Table D-2 Braidwood Unit 1 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[a] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[b] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Forging (Tangential)	U	26.6	0.388	0.738	5.6	19.6	14.0	Yes
	X		1.17	1.044	37.9	27.7	10.2	Yes
	W		1.98	1.186	24.0	31.5	7.5	Yes
	V		3.71	1.340	51.3	35.6	15.7	Yes
Lower Shell Forging (Axial)	U		0.388	0.738	-15.8[c]	19.6	35.4	No
	X		1.17	1.044	29.3	27.7	1.6	Yes
	W		1.98	1.186	37.1	31.5	5.6	Yes
	V		3.71	1.340	39.7	35.6	4.1	Yes
Braidwood Unit 1 Surveillance Weld Metal	U	38.5	0.388	0.738	17.4	28.4	11.0	Yes
	X		1.17	1.044	29.8	40.2	10.4	Yes
	W		1.98	1.186	49.0	45.7	3.3	Yes
	V		3.71	1.340	62.8	51.6	11.2	Yes

Notes

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17]

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}]

(c) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

From a statistical point of view, ± 1σ would be expected to encompass 68% of the data. The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 17°F for base metal. Table D-2 indicates that only one of the eight surveillance data points fall outside the ± 1σ of 17°F scatter band for surveillance base metals (87.5% within the scatter band); <u>therefore, the forging data is deemed "credible" per the third criterion</u>.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-2 indicates that all four surveillance data points (100%) fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; <u>therefore, the weld material is deemed "credible" per the third criterion when only the Braidwood Unit 1 data is considered</u>.

Next, data from all sources for weld material with the same heat number is considered in order to evaluate the credibility of the weld metal using the NRC Case 4 guidelines. Data for the Braidwood Unit 1 surveillance weld material is also available from the Braidwood Unit 2 surveillance weld material. Since data are from multiple sources, the data may need to be adjusted for chemical and irradiation environment differences.

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. This is performed as follows:

Braidwood Unit 1 surveillance weld metal

Cu Wt. % = 0.03, Ni Wt. % = 0.67, Position 1.1 CF = 41°F (from Table 3-1 and Table 5-4)

Braidwood Unit 2 surveillance weld metal

Cu Wt. % = 0.03, Ni Wt. % = 0.71, Position 1.1 CF = 41°F (from Table 3-2 and Table 5-5)

The mean chemical composition is: Cu Wt. % = 0.03, Ni Wt. % = 0.69, Position 1.1 CF = 41°F

Since the mean chemical composition yields the same Position 1.1 CF as the Braidwood Units 1 and 2 surveillance weld materials, chemistry adjustments are not needed. Furthermore, as described in Section 5, since Braidwood Units 1 and 2 have similar operating temperatures, temperature adjustments are also not needed. Table D-3 provides the summary of the weld interim CF considering all available data.

Table D-3 Calculation of Interim Weld Chemistry Factor for the Credibility Evaluation Using all Available Surveillance Data

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Braidwood Unit 1 Surveillance Weld Material (Heat # 442011)	U	0.388	0.738	17.4	12.84	0.54
	X	1.17	1.044	29.8	31.11	1.09
	W	1.98	1.186	49.0	58.14	1.41
	V	3.71	1.340	62.8	84.13	1.79
Braidwood Unit 2 Surveillance Weld Material (Heat # 442011)	U	0.387	0.737	-0.80[(d)]	-0.59	0.54
	X	1.15	1.039	26.1	27.12	1.08
	W	2.07	1.198	23.7	28.39	1.44
	V	3.73	1.341	45.6	61.14	1.80
				SUM:	302.28	9.69
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (302.28) \div (9.69) =$ **31.2°F**					

Notes:

(a) Taken from Table 5-3.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$

(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18]

(d) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-4.

Table D-4 Braidwood Unit 1 Calculated Surveillance Weld Metal Data Scatter about the Best-Fit Line using all Available Surveillance Data

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured(a) ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}(b) (°F)	<17°F (Base Metal) <28°F (Weld)
Braidwood Unit 1 Surveillance Weld Metal	U	31.2	0 388	0.738	17.4	23.0	5.6	Yes
	X		1.17	1.044	29.8	32.6	2.8	Yes
	W		1.98	1.186	49.0	37.0	12.0	Yes
	V		3.71	1.340	62.8	41.8	21.0	Yes
Braidwood Unit 2 Surveillance Weld Metal	U		0.387	0.737	-0.80(c)	23.0	23.8	Yes
	X		1.15	1.039	26.1	32.4	6.3	Yes
	W		2.07	1.198	23.7	37.4	13.7	Yes
	V		3.73	1.341	45.6	41.8	3.8	Yes

Notes·

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18].

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}]

(c) Even though a reduction should not occur, usıng the negative measured ΔRT_{NDT} value produces the most conservative results for this credıbility evaluation.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-4 indicates that all eight surveillance data points (100%) fall within the ± 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when all available data for the Braidwood Unit 1 weld is considered.

Hence, Criterion 3 is met for the Braidwood Unit 1 surveillance program materials.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The Braidwood Unit 1 capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions and will not differ by more than 25°F.

Hence, Criterion 4 is met for the Braidwood Unit 1 surveillance program.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Braidwood Unit 1 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Braidwood Unit 1 surveillance program.

Hence, Criterion 5 is met for the Braidwood Unit 1 surveillance program.

Conclusion: Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The Braidwood Unit 1 surveillance forging data are deemed "credible"
- The Braidwood Unit 1 surveillance weld data are deemed "credible"

D.2 BRAIDWOOD UNIT 2 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [4], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Braidwood Unit 2 reactor vessel beltline region consists of the following materials:

1. Nozzle Shell Forging 5P-7056
2. Intermediate Shell Forging [49D963/49C904]-1-1
3. Lower Shell Forging [50D102/50C97]-1-1
4. Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld Seam WF-645 (Weld Wire Heat # H4498, Linde 80 Flux Type, Flux Lot # 0261)
5. Intermediate Shell Forging to Lower Shell Forging Circumferential Weld Seams WF-562 (Weld Wire Heat # 442011, Linde 80 Flux Type, Flux Lot # 8061)

The Braidwood Unit 2 surveillance program utilizes tangential and axial test specimens from the Lower Shell Forging. The surveillance weld metal was fabricated with weld wire Heat # 442011, Flux Type Linde 80, Lot # 0344.

At the time when the Braidwood Unit 2 surveillance program material was selected, it was believed that copper and phosphorus were the elements most important to the embrittlement of reactor vessel steels and the Nozzle Shell Forging was not considered a "beltline" material. The Lower Shell Forging had an initial RT_{NDT} that was equal to the Intermediate Shell Forging initial RT_{NDT}, but had higher copper content. Based on this comparison of the beltline forging materials (Intermediate and Lower Shell Forgings), the Lower Shell Forging was chosen for the surveillance program.

Weld seam WF-562, on the other hand, was considered the only weld in the beltline region and therefore was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 442011), the same type of flux (Linde 80), as the Intermediate to Lower Shell Forging Circumferential Weld Seam.

Therefore, the materials selected for use in the Braidwood Unit 2 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

<u>Based on the discussion above, Criterion 1 is met for the Braidwood Unit 2 surveillance program.</u>

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-18107, Revision 1 [18].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Braidwood Unit 2 surveillance materials unambiguously.

Hence, the Braidwood Unit 2 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [21].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the forging.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of [1]. Braidwood Unit 2 has one circumferential weld that will be evaluated for credibility. This weld is Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562 and is fabricated from weld wire Heat # 442011, Linde 80 type flux, Lot # 8061. This weld metal heat is contained in both the Braidwood Unit 1 and the Braidwood Unit 2 surveillance programs. Since the welds in question utilize data from other surveillance programs, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [22]. At this meeting the NRC presented five cases. Of the five cases, Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situation listed above for Braidwood Unit 2 surveillance weld metal. Note that for the forging materials, the straightforward method in [1] will be followed.

Following the NRC Case 4 guidelines, only the Braidwood Unit 2 data will be evaluated first. Table D-5 provides the calculation of the interim CF for Braidwood Unit 2. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values. In addition, only Braidwood Unit 2 data is being considered; therefore, no temperature adjustment is required.

Table D-5 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Braidwood Unit 2 Surveillance Data

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [50D102/50C97]-1-1 (Tangential)	U	0.387	0.737	-9.6[(d)]	-7.08	0.54
	X	1.15	1.039	-9.6[(d)]	-9.97	1.08
	W	2.07	1.198	4.6	5.51	1.44
	V	3.73	1.341	28.4	38.08	1.80
Lower Shell Forging [50D102/50C97]-1-1 (Axial)	U	0.387	0.737	-0.10[(d)]	-0.07	0.54
	X	1.15	1.039	33.8	35.12	1.08
	W	2.07	1.198	33.1	39.65	1.44
	V	3.73	1.341	63.3	84.89	1.80
				SUM:	186.12	9.71
	$CF_{[50D102/50C97]-1-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (186.12) \div (9.71) =$ **19.2°F**					
Braidwood Unit 2 Surveillance Weld Material (Heat # 442011)	U	0.387	0.737	-0.80[(d)]	-0.59	0.54
	X	1.15	1.039	26.1	27.12	1.08
	W	2.07	1.198	23.7	28.39	1.44
	V	3.73	1.341	45.6	61.14	1.80
				SUM:	116.07	4.86
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (116.07) \div (4.86) =$ **23.9°F**					

Notes

(a) Taken from Table 5-2.
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [18].
(d) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-6.

Table D-6 Braidwood Unit 2 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[(a)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Forging (Tangential)	U	19.2	0.387	0.737	-9.6[(c)]	14.1	23.7	No
	X		1.15	1.039	-9.6[(c)]	19.9	29.5	No
	W		2.07	1.198	4.6	23.0	18.4	No
	V		3.73	1.341	28.4	25.7	2.7	Yes
Lower Shell Forging (Axial)	U		0.387	0.737	-0.10[(c)]	14.1	14.2	Yes
	X		1.15	1.039	33.8	19.9	13.9	Yes
	W		2.07	1.198	33.1	23.0	10.1	Yes
	V		3.73	1.341	63.3	25.7	37.6	No
Braidwood Unit 2 Surveillance Weld Metal	U	23.9	0.387	0.737	-0.80[(c)]	17.6	18.4	Yes
	X		1.15	1.039	26.1	24.8	1.3	Yes
	W		2.07	1.198	23.7	28.6	4.9	Yes
	V		3.73	1.341	45.6	32.0	13.6	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [18].

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

(c) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation

From a statistical point of view, ± 1σ would be expected to encompass 68% of the data. The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 17°F for base metal. Table D-6 indicates that four of the eight surveillance data points (50%) fall outside the ± 1σ of 17°F scatter band for surveillance base metals; therefore, the forging data is deemed "non-credible" per the third criterion.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-6 indicates that all four surveillance data points (100%) fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when only the Braidwood Unit 2 data is considered.

Next, data from all sources is considered in order to evaluate the credibility of the weld metal using the NRC Case 4 guidelines. Data for the Braidwood Unit 2 surveillance weld material is also available from the Braidwood Unit 1 surveillance weld material. Since data are from multiple sources, the data may need to be adjusted for chemical and irradiation environment differences.

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. This is performed as follows:

Braidwood Unit 1 surveillance weld metal

Cu Wt. % = 0.03, Ni Wt. % = 0.67, Position 1.1 CF = 41°F (from Table 3-1 and Table 5-4)

Braidwood Unit 2 surveillance weld metal

Cu Wt. % = 0.03, Ni Wt. % = 0.71, Position 1.1 CF = 41°F (from Table 3-2 and Table 5-5)

The mean chemical composition is: Cu Wt. % = 0.03, Ni Wt. % = 0.69, Position 1.1 CF = 41°F

Since the mean chemical composition yields the same Position 1.1 CF as the Braidwood Units 1 and 2 surveillance weld materials, chemistry adjustments are not needed. Furthermore, as described in Section 5, since Braidwood Units 1 and 2 have similar operating temperatures temperature adjustments are also not needed. Table D-7 provides the summary of the weld interim CF considering all available data.

Table D-7 Calculation of Interim Weld Chemistry Factor for the Credibility Evaluation Using all Available Surveillance Data

Material	Capsule	Capsule Fluence[(a)] ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Braidwood Unit 1 Surveillance Weld Material (Heat # 442011)	U	0.388	0.738	17.4	12.84	0.54
	X	1.17	1.044	29.8	31.11	1.09
	W	1.98	1.186	49.0	58.14	1.41
	V	3.71	1.340	62.8	84.13	1.79
Braidwood Unit 2 Surveillance Weld Material (Heat # 442011)	U	0.387	0.737	-0.80[(d)]	-0.59	0.54
	X	1.15	1.039	26.1	27.12	1.08
	W	2.07	1.198	23.7	28.39	1.44
	V	3.73	1.341	45.6	61.14	1.80
				SUM:	302.28	9.69
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (302.28) \div (9.69) =$ **31.2°F**					

Notes

(a) Taken from Table 5-3
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18]
(d) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-8.

Table D-8 Braidwood Unit 2 Calculated Surveillance Weld Metal Data Scatter about the Best-Fit Line using all Available Surveillance Data

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[(a)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
Braidwood Unit 1 Surveillance Weld Metal	U	31.2	0.388	0.738	17.4	23.0	5.6	Yes
	X		1.17	1.044	29.8	32.6	2.8	Yes
	W		1.98	1.186	49.0	37.0	12.0	Yes
	V		3.71	1.340	62.8	41.8	21.0	Yes
Braidwood Unit 2 Surveillance Weld Metal	U		0.387	0.737	-0.80[(c)]	23.0	23.8	Yes
	X		1.15	1.039	26.1	32.4	6.3	Yes
	W		2.07	1.198	23.7	37.4	13.7	Yes
	V		3.73	1.341	45.6	41.8	3.8	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18].
(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].
(c) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-8 indicates that all eight surveillance data points (100%) fall within the ± 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when all available data for the Braidwood Unit 2 weld is considered.

Hence, Criterion 3 is met for the Braidwood Unit 2 surveillance program materials.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The Braidwood Unit 2 capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions and will not differ by more than 25°F.

Hence, Criterion 4 is met for the Braidwood Unit 2 surveillance program.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Braidwood Unit 2 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Braidwood Unit 2 surveillance program.

Hence, Criterion 5 is met for the Braidwood Unit 2 surveillance program.

Conclusion: Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The Braidwood Unit 2 surveillance forging data are deemed "non-credible"
- The Braidwood Unit 2 surveillance weld data are deemed "credible"

APPENDIX E PRESSURIZED THERMAL SHOCK EVALUATION

Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the reactor pressure vessel (RPV) under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [23]) that established screening criteria on Pressurized Water Reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. The RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 [1].

These accepted methods were used with the clad/base metal interface fluence values of Section 2 to calculate the following RT_{PTS} values for the Braidwood Units 1 and 2 RPV materials at 57 EFPY (EOLE). The EOLE RT_{PTS} calculations are summarized in Table E-1 and Table E-2.

PTS Conclusion

All of the beltline and extended beltline materials in the Braidwood Units 1 and 2 reactor vessels are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds at 57 EFPY. Therefore, all materials are acceptable.

Table E-1 RT_{PTS} Calculations for the Braidwood Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Heat Number	CF[(a)] (°F)	Surface Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(d)] (°F)	Margin (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials										
Nozzle Shell Forging	5P-7016	26.0	1.13	1.034	10	26.9	0	13.4	26.9	63.8
Intermediate Shell Forging	[49D383/49C344]-1-1	31.0	3.22	1.307	-30	40.5	0	17.0	34.0	44.5
Lower Shell Forging	[49D867/49C813]-1-1	31.0	3.14	1.302	-20	40.3	0	17.0	34.0	54.3
Lower Shell Forging Using Credible Braidwood Unit 1 Surveillance Data	[49D867/49C813]-1-1	27.8	3.14	1.302	-20	36.2	0	8.5	17.0	33.2
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498	54.0	1.13	1.034	-25	55.8	0	27.9	55.8	86.7
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011	41.0	3.06	1.295	40	53.1	0	26.6	53.1	146.2
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	442011	31.2	3.06	1.295	40	40.4	0	14.0	28.0	108.4
Reactor Vessel Extended Beltline Materials										
Inlet Nozzle 01-001	21-3257	58.0	0.0124	0.127	-20	7.3	0	3.7	7.3	-5.3
Inlet Nozzle 01-002	21-3257	58.0	0.0124	0.127	-10	7.3	0	3.7	7.3	4.7
Inlet Nozzle 02-001	22-3313	44.0	0.0124	0.127	-10	5.6	0	2.8	5.6	1.1
Inlet Nozzle 02-002	22-3313	44.0	0.0124	0.127	0	5.6	0	2.8	5.6	11.1
Outlet Nozzle 01-001	22-3025	96.0	0.00938	0.105	-10	10.1	0	5.0	10.1	10.2
Outlet Nozzle 01-003	11-5226	58.0	0.00938	0.105	-10	6.1	0	3.0	6.1	2.2
Outlet Nozzle 02-001	4-3329	51.0	0.00938	0.105	-20	5.4	0	2.7	5.4	-9.3
Outlet Nozzle 02-002	4-3383	51.0	0.00938	0.105	-20	5.4	0	2.7	5.4	-9.3
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403	185.6	0.0124	0.127	-48.6	23.5	18[(f)]	28.0[(e)]	66.6	41.5
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-598	41403	185.6	0.00938	0.105	-48.6	19.5	18[(f)]	28.0[(e)]	66.6	37.5

Table E-1 RT_{PTS} Calculations for the Braidwood Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Heat Number	CF(a) (°F)	Surface Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ(d) (°F)	Margin (°F)	RT_{PTS} (°F)
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-588	41403	195.7	0.00938	0.105	-48.6	20.5	18(f)	28.0(e)	66.6	38.5
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-579	442010	181.0	0.00938	0.105	-48.6	19.0	18(f)	28.0(e)	66.6	37.0

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix D, the lower shell forging material and the intermediate to lower shell weld material surveillance data were determined to be credible. Therefore, per the guidance of 10 CFR 50.61 [23], the base metal σ_Δ = 17°F when surveillance data not used to determine the CF and σ_Δ = 8.5°F when credible surveillance data is used to determine the CF, and the weld metal σ_Δ = 28°F for the when surveillance data not used, and σ_Δ = 14°F when credible surveillance data is used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per [23].

(e) Value is required per condition from [6] This condition must be met in order to use values from Table 9 of [6] Revision 2-A

(f) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(u)}$ value.

Table E-2 RT_{PTS} Calculations for the Braidwood Unit 2 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Material ID/Heat Number	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials										
Nozzle Shell Forging	5P-7056	26.0	0.994	0.998	30	26.0	0	13.0	26.0	81.9
Intermediate Shell Forging	[49D963/49C904]-1-1	20.0	2.95	1.287	-30	25.7	0	12.9	25.7	21.5
Lower Shell Forging	[50D102/50C97]-1-1	37.0	3.03	1.293	-30	47.8	0	17.0	34.0	51.8
Lower Shell Forging Using Non-Credible Braidwood Unit 2 Surveillance Data	[50D102/50C97]-1-1	20.9	3.03	1.293	-30	27.0	0	13.5	27.0	24.1
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-645	H4498	54.0	0.994	0.998	-25	53.9	0	27.0	53.9	82.8
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-562	442011	41.0	2.90	1.283	40	52.6	0	26.3	52.6	145.2
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	442011	31.2	2.90	1.283	40	40.0	0	14.0	28.0	108.0
Reactor Vessel Extended Beltline Materials										
Inlet Nozzle 01-001	41-5414	44.0	0.0106	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 01-002	41-5414	44.0	0.0106	0.114	-10	5.0	0	2.5	5.0	0.0
Inlet Nozzle 02-001	42-5417	58.0	0.0106	0.114	-10	6.6	0	3.3	6.6	3.2
Inlet Nozzle 02-002	42-5417	58.0	0.0106	0.114	-10	6.6	0	3.3	6.6	3.2
Outlet Nozzle 01-002	11-5266	58.0	0.00797	0.094	10	5.4	0	2.7	5.4	20.9
Outlet Nozzle 01-003	11-5226	58.0	0.00797	0.094	-10	5.4	0	2.7	5.4	0.9
Outlet Nozzle 02-001	4-3481	44.0	0.00797	0.094	-10	4.1	0	2.1	4.1	-1.7
Outlet Nozzle 02-002	4-3502	58.0	0.00797	0.094	-10	5.4	0	2.7	5.4	0.9
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404	167.0[e]	0.0106	0.114	-48.6	19.0	18[g]	28.0[f]	66.6	37.0
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-654	41404	167.0[e]	0.00797	0.094	-48.6	15.7	18[g]	28.0[f]	66.6	33.6

Notes (continued on next page):

(a) Values are taken from Table 5-5.

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the intermediate to lower shell weld material surveillance data was determined to be credible. Per the guidance of 10 CFR 50.61 [23], the base metal σ_Δ = 17°F when surveillance data is non-credible or not used to determine the CF, and the weld metal σ_Δ = 28°F when surveillance data is not used and σ_Δ = 14°F when credible surveillance data is used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per [23].

(e) Value is required minimum per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(f) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(g) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(u)}$ value

APPENDIX F VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

F.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for all surveillance capsules withdrawn from service to date at Braidwood Unit 1 and Unit 2 are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [9]. One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 2 of this report.

F.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the four in-vessel neutron sensor sets and two sets of Ex-Vessel Neutron Dosimetry (EVND) dosimetry sets analyzed to date as part of the Braidwood Unit 1 and Unit 2 Reactor Vessels Materials Surveillance Program are presented.

Six irradiation capsules attached to the neutron pads were included in the Braidwood Unit 1 reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 58.5°, 121.5°, 238.5°, and 301.5° (31.5° from the core cardinal axis) and 61° and 241° (29° from the core cardinal axis). The irradiation history of each of these six in-vessel surveillance capsules is summarized as follows:

Capsule	Location	Irradiation History
U	31.5° Dual	Cycle 1 (withdrawn for analysis)
X	31.5° Dual	Cycles 1–4 (withdrawn for analysis)
W	31.5° Single	Cycles 1–7 (withdrawn for analysis)
V	29° Dual	Cycles 1–14 (withdrawn for analysis)
Y	29° Dual	Cycles 1–10 (withdrawn for storage)
Z	31.5° Single	Cycles 1–10 (withdrawn for storage)

In addition to the four previously analyzed in-vessel surveillance capsules, two sets of EVND dosimetry sensors have been analyzed after withdrawal from Braidwood Unit 1. The location and time of irradiation for EVND sensor sets analyzed at Braidwood Unit 1 are summarized as follows:

Capsule ID	Azimuthal Location from Cardinal Axis	Gradient Chain ID Tag on which EVND Capsule is Located	Axial Elevation	Cycles of Irradiation
A	0.5°	1S-1 00	Active Core Midplane	14
B	14.5°	1S-1 15	Active Core Midplane	14
C	29.5°	1S-1 30	Active Core Midplane	14
E	44.5°	1S-1 45	Active Core Midplane	14
G	0.5°	1S-2 00	Active Core Midplane	15–19
H	14.5°	1S-2 15	Active Core Midplane	15–19
I	29.5°	1S-2 30	Active Core Midplane	15–19
K	44.5°	1S-2 45	Active Core Midplane	15–19
D	44.5°	1S-1 45	Active Core Top	14
F	44.5°	1S-1 45	Active Core Bottom	14
J	44.5°	1S-2 45	Active Core Top	15–19
L	44.5°	1S-2 45	Active Core Bottom	15–19

Similarly, six irradiation capsules attached to the neutron pads were included in the Braidwood Unit 2 reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 58.5°, 121.5°, 238.5°, 301.5° (31.5° from the core cardinal axis), 61.0° and 241.0° (29.0° from the core cardinal axis). The irradiation history of each of these six in-vessel surveillance capsules is summarized as follows:

Capsule	Location	Irradiation History
U	31.5° Dual	Cycle 1 (withdrawn for analysis)
X	31.5° Dual	Cycles 1-4A* (withdrawn for analysis)
W	31.5° Single	Cycles 1-7 (withdrawn for analysis)
V	29.0° Dual	Cycles 1-14 (withdrawn for analysis)
Z	31.5° Single	Cycles 1-10 (withdrawn for storage)
Y	29.0° Dual	Cycles 1-10 (withdrawn for storage)

Notes: * 4A indicates time at which Capsule X was withdrawn

In addition to the four previously analyzed in-vessel surveillance capsules, two sets of EVND dosimetry sensors have been analyzed after withdrawal from Braidwood Unit 2. The location and time of irradiation for EVND sensor sets analyzed at Braidwood Unit 2 are summarized as follows:

Capsule ID	Azimuthal Location from Cardinal Axis	Gradient Chain ID Tag on which EVND Capsule is Located	Axial Elevation	Cycles of Irradiation
A	0.5°	2S-1 00	Active Core Midplane	14
B	14.5°	2S-1 15	Active Core Midplane	14
C	29.5°	2S-1 30	Active Core Midplane	14
E	44.5°	2S-1 45	Active Core Midplane	14
G	0.5°	2S-2 00	Active Core Midplane	15–19
H	14.5°	2S-2 15	Active Core Midplane	15–19
I	29.5°	2S-2 30	Active Core Midplane	15–19
K	44.5°	2S-2 45	Active Core Midplane	15–19
D	44.5°	2S-1 45	Active Core Top	14
F	44.5°	2S-1 45	Active Core Bottom	14
J	44.5°	2S-2 45	Active Core Top	15–19
L	44.5°	2S-2 45	Active Core Bottom	15–19

The azimuthal locations included in the above tabulation represent the first octant equivalent azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of Braidwood Unit 1 and Unit 2 Surveillance Capsules U, X, W, and V are summarized as follows:

Sensor Material	Reaction Of Interest	Capsule U	Capsule X	Capsule W	Capsule V
Copper	$^{63}Cu(n,\alpha)^{60}Co$	X	X	X	X
Iron	$^{54}Fe(n,p)^{54}Mn$	X	X	X	X
Nickel	$^{58}Ni(n,p)^{58}Co$	X	X	X	
Uranium-238	$^{238}U(n,f)^{137}Cs$	X	X	X	X
Neptunium-237	$^{237}Np(n,f)\ ^{137}Cs$	X	X	X	X
Cobalt-Aluminum*	$^{59}Co(n,\gamma)^{60}Co$	X	X	X	X
*The cobalt-aluminum measurements include both bare wire and cadmium-covered sensors.					

Since all of the dosimetry monitors were located at the radial center of the material test specimen array, radial gradient corrections were not required for these reaction rates. Pertinent physical and nuclear characteristics of the in-vessel surveillance capsule passive neutron sensors are listed in Table F-1. The pertinent physical and nuclear characteristics of the EVND capsule passive neutron sensors are listed in Table F-9.

The use of passive monitors such as those listed above does not yield a direct measure of the energy-dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of

the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- The physical characteristics of each monitor,
- The operating history of the reactor,
- The energy response of each monitor, and
- The neutron energy spectrum at the monitor location.

Results from the radiometric counting of the neutron sensors from Braidwood Unit 1 Capsules U, X, W, and V are documented in [29, 30, 31, and 17], respectively, and re-evaluated in [7]. Results from the radiometric counting of the neutron sensors from Braidwood Unit 2 Capsules U, X, W, and V are documented in [32, 33, 34, and 18], respectively, and re-evaluated in [8]. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by Capsules U, X, W, and V was based on the monthly power generation of Braidwood Unit 1 from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The startup and shutdown dates for each cycle at Braidwood Unit 1 are given in Table F-2. Similarly, the startup and shutdown dates for each cycle at Braidwood Unit 2 are given in Table F-3.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum \frac{P_j}{P_{ref}} C_j \, [1 - e^{-\lambda t_j}] \, [e^{-\lambda t_{d,j}}]}$$

where:

R = Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).

A = Measured specific activity (dps/g).

N_0 = Number of target element atoms per gram of sensor.

F = Atom fraction of the target isotope in the target element.

Y = Number of product atoms produced per reaction.

P_j = Average core power level during irradiation period j (MW).

P_{ref} = Maximum or reference power level of the reactor (MW).

C_j = Calculated ratio of $\phi(E > 1.0\ \text{MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0\ \text{MeV})$ over the entire irradiation period.

λ = Decay constant of the product isotope (1/sec).

t_j = Length of irradiation period j (sec).

t_{d_j} = Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology discussed in Section 2, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel-cycle-specific neutron flux values are used to compute values for cycle-dependent C_j values at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets at both Braidwood Unit 1 and Unit 2, additional corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U and ^{237}Np sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the Braidwood Unit 1 fission sensor reaction rates are summarized as follows:

Correction	Capsule U	Capsule X	Capsule W	Capsule V
^{235}U Impurity/Pu Build-in	0.869	0.839	0.810	0.752
$^{238}U(\gamma,f)$	0.966	0.967	0.970	0.968
Net ^{238}U Correction	0.839	0.811	0.786	0.728
$^{237}Np(\gamma,f)$	0.990	0.990	0.991	0.991

Similarly, the correction factors applied to the Braidwood Unit 2 fission sensor reaction rates are summarized as follows:

Correction	Capsule U	Capsule X	Capsule W	Capsule V
^{235}U Impurity/Pu Build-in	0.869	0.840	0.807	0.751
$^{238}U(\gamma,f)$	0.967	0.967	0.970	0.968
Net ^{238}U Correction	0.840	0.812	0.783	0.727
$^{237}Np(\gamma,f)$	0.990	0.990	0.991	0.991

These factors were applied in a multiplicative fashion to the decay corrected uranium and neptunium fission sensor reaction rates.

Results of the sensor reaction rate determinations for Braidwood Unit 1 Capsules U, X, W, and V are given in Table F-4 through Table F-7. Results of the sensor reaction rate determinations for Braidwood Unit 2 Capsules U, X, W, and V are given in Table F-22 through Table F-25. In Table F-4 through Table F-7 and Table F-22 through Table F-25, the measured specific activities, decay-corrected saturated specific activities, and computed reaction rates for each sensor indexed to the radial center of the capsule are listed. The fission sensor reaction rates are listed both with and without the applied corrections for ^{238}U impurities, plutonium build-in, and gamma-ray-induced fission effects.

Table F-8 lists the EVND capsule designations for the Braidwood Unit 1 and Unit 2 EVND capsules. Similarly, the nuclear parameters listed in Table F-9 are used to evaluate the Braidwood Unit 1 and Unit 2 EVND capsule neutron sensors irradiated in Cycle 14 and Cycles 15-19, respectively. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycle 14 at Braidwood Unit 1 are reported in Table F-10 through Table F-15. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycles 15 through 19 at Braidwood Unit 1 are reported in Table F-16 through Table F-21. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycle 14 at Braidwood Unit 2 are reported in Table F-26 through Table F-31. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycles 15 through 19 at Braidwood Unit 2 are reported in Table F-32 through Table F-37.

F.1.2 Least-Squares Evaluation of Sensor Sets

Least-squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations resulting in a Best-Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as $\phi(E > 1.0 \text{ MeV})$ or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\varphi_g \pm \delta_{\varphi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the Braidwood Unit 1 surveillance capsule dosimetry, the FERRET Code [35] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters ($\phi(E > 1.0 \text{ MeV})$ and dpa) along with associated uncertainties for the five in-vessel capsules analyzed to date.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Braidwood Unit 1 application, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section F.1.1. The dosimetry reaction cross sections and uncertainties were obtained from the Sandia National Laboratories Radiation Metrology Laboratory (SNLRML) dosimetry cross-section library [36]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB)" [37].

The uncertainties associated with the measured reaction rates, dosimetry cross sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The

assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" [38].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the Braidwood Unit 1 and Unit 2 surveillance capsule sensor sets.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{46}Ti\ (n,p)\ ^{46}Sc$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	10%
$^{93}Nb\ (n,n')\ ^{93m}Nb$	10%
$^{237}Np(n,f)^{137}Cs$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations, and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the Braidwood Unit 1 surveillance program, the following uncertainties in the fission spectrum averaged cross sections are provided in the SNLRML documentation package.

Reaction	**Uncertainty**
$^{63}Cu(n,\alpha)^{60}Co$	4.08-4.16%
$^{46}Ti(n,p)^{46}Sc$	4.50-4.87%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{238}U(n,f)^{137}Cs$	0.54-0.64%
$^{93}Nb\ (n,n')\ ^{93m}Nb$	6.96-7.23%
$^{237}Np(n,f)^{137}Cs$	10.32-10.97%
$^{59}Co(n,\gamma)^{60}Co$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectra input to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta\, e^{-H}$$

where

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g', and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Braidwood Unit 1 calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

F.1.3 Comparisons of Measurements and Calculations

Braidwood Unit 1 Neutron Dosimetry Benchmark

In this section, comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations are presented for Braidwood Unit 1. These comparisons are provided on two levels. In the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second level, calculated values of neutron exposure rates in terms of fast neutron fluence rate ϕ (E > 1.0 MeV) and iron atom displacements are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table F-38, comparisons of measured-to-calculation (M/C) ratios are listed for the threshold sensors contained in the in-vessel Capsules U, X, W and V. From Table F-38, it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.97 to 1.15 with an overall average of 1.05 and an associated standard deviation of 10.1%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table F-39, similar comparisons are provided for the sensor sets withdrawn from the midplane axial elevation measurement locations in the reactor cavity. From Table F-39, it is noted that for the individual threshold foils, the average M/C ratio ranges from 0.85 to 1.02 with an overall average of 0.93 and an associated standard deviation of 7.1%. From Table F-40, it is also noted that the M/C ratios for the EVND Set 1S-1 and 1S-2 capsules are in good agreement. Finally, as in the case of the in-vessel comparisons, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

Comparisons of the M/C ratios for the two sensor sets withdrawn from the off-midplane measurement locations in the reactor cavity are provided in Table F-40.

In Table F-41 and Table F-42, best-estimate to calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set are provided for the in-vessel and midplane ex-vessel irradiations, respectively. For the in-vessel capsules the average BE/C ratio is seen to be 1.00 with an associated uncertainty of 7.3% for neutron fluence rate (E > 1.0 MeV) and 1.00 with an associated uncertainty of 6.9% for the iron atom displacement rate. The corresponding average BE/C ratio from the midplane ex-vessel irradiations is 0.95 with an uncertainty of 2.8% for neutron fluence rate (E > 1.0 MeV) and 0.99 with an uncertainty of 2.9% for the iron atom displacement rate.

The M/C data sets listed in Table F-38 and Table F-39, as well as the BE/C data sets given in Table F-41 and Table F-42, provide a validation of the plant-specific neutron transport calculations described in Section 2 of this report. Each of these data comparisons shows that for both in-vessel and midplane ex-vessel locations the measurements and calculations agree well within the 20% criterion specified in Regulatory Guide 1.190. In fact, both the average M/C results and BE/C results fall within the 13% (1σ) uncertainty assigned to the fast neutron exposure of the absolute transport calculations.

The measurements to calculation comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	Unit 1 In-Vessel		Unit 1 Ex-Vessel Midplane		Unit 1 Combined	
	Avg. M/C	% Unc. (1σ)	Avg. M/C	% Unc. (1σ)	Avg. M/C	% Unc. (1σ)
^{63}Cu(n,α)	1.15	3.4%	0.92	3.0%	1.04	2.3%
^{46}Ti(n,p)	-	-	0.92	6.4%	-	-
^{54}Fe(n,p)	1.00	5.4%	0.94	3.4%	0.97	3.2%
^{58}Ni(n,p)	0.97	10.6%	0.85	3.9%	0.91	5.9%
^{93}Nb(n,n')	-	-	1.02	4.2%	-	-
^{238}U(Cd) (n,f)	1.12	8.8%	-	-	-	-
^{237}Np(Cd)(n,f)	0.98	8.7%	-	-	-	-
Linear Average	1.05	10.1%	0.93	7.1%	0.99	6.3%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) are summarized as follows:

Parameter	Unit 1 In-Vessel		Unit 1 Ex-Vessel Midplane		Unit 1 Combined	
	Avg. BE/C	% Unc. (1σ)	Avg. BE/C	% Unc. (1σ)	Avg. BE/C	% Unc. (1σ)
Fast Neutron Fluence (E > 1.0 MeV)	1.00	7.3%	0.95	2.8%	0.97	4.0%
dpa/s	1.00	6.9%	0.99	2.9%	1.00	3.8%

These data comparisons show similar and consistent results with the linear average M/C ratio of 0.99 in good agreement with the resultant least-squares BE/C ratios of 0.97 for neutron fluence rate (E > 1.0 MeV) and 1.00 for iron atom displacement rate. The comparisons demonstrate that the calculated results for Braidwood Unit 1 provided in Section 2 of this report are validated within the context of the assigned 13% uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by Reference 9 is easily met.

Braidwood Unit 2 Neutron Dosimetry Benchmark

In this section, comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations are presented for Braidwood Unit 2. These comparisons are provided on two levels. In the first instance, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second instance, calculated values of neutron exposure rates in terms of neutron fluence (E > 1.0 MeV) and iron atom displacements are compared with the best estimate exposure rates obtained from the least-squares evaluation.

In Table F-43, comparisons of M/C ratios are listed for the threshold sensors contained in in-vessel Capsules U, X, W and V. From Table F-43 it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.96 to 1.15 with an overall average of 1.03 and an associated standard deviation of 8.1%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table F-44, similar comparisons are provided for the sensor sets withdrawn from the midplane axial elevation measurement locations in the reactor cavity. From Table F-44, it is noted that for the individual threshold foils, the average M/C ratio ranges from 0.87 to 1.06 with an overall average of 0.95 and an associated standard deviation of 7.4%. From Table F-44, it is also noted that the M/C ratios for the EVND Set 2S-1 and 2S-2 capsules are in good agreement. Finally, as in the case of the in-vessel comparisons, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

Comparisons of the M/C ratios for the two sensor sets withdrawn from the off-midplane measurement locations in the reactor cavity are provided in Table F-45.

In Table F-46 and Table F-47, best-estimate to calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set are provided for the in-vessel and midplane ex-vessel irradiations, respectively. For the in-vessel capsules the average BE/C ratio is seen to be 0.98 with an associated uncertainty of 4.9% for neutron fluence rate (E > 1.0 MeV) and 0.98 with an associated uncertainty of 4.0% for the iron atom displacement rate. The corresponding average BE/C ratio from the midplane ex-vessel irradiations is 0.98 with an uncertainty of 3.2% for neutron fluence rate (E > 1.0 MeV) and 1.01 with an uncertainty of 3.0% for the iron atom displacement rate.

The M/C data sets listed in Table F-43 and Table F-44, as well as the BE/C data sets given in Table F-46 and Table F-47, provide a validation of the plant-specific neutron transport calculations described in Section 2 of this report. Each of these data comparisons shows that for both in-vessel and midplane ex-vessel locations the measurements and calculations agree well within the 20% criterion specified in Regulatory Guide 1.190. In fact, both the average M/C results and BE/C results fall within the 13% (1σ) uncertainty assigned to the fast neutron exposure of the absolute transport calculations.

The measurements to calculation comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	Unit 2 In-Vessel		Unit 2 Midplane Ex-Vessel		Unit 2 Combined	
	Average	% Standard Deviation	Average	% Standard Deviation	Average	% Standard Deviation
	M/C	(1σ)	M/C	(1σ)	M/C	(1σ)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.15	1.1%	0.94	3.3%	1.01	1.7%
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.99	4.3%	0.95	3.5%	0.96	2.8%
$^{58}Ni\ (n,p)\ ^{58}Co$	0.96	8.3%	0.87	4.0%	0.90	4.8%
$^{46}Ti\ (n,p)\ ^{46}Sc$	-	-	0.92	2.7%	0.92	2.7%
$^{93}Nb\ (n,n)\ ^{93m}Nb$	-	-	1.06	4.0%	1.06	4.0%
$^{238}U\ (n,f)\ ^{137}Cs$	1.04	2.9%	-	-	1.04	2.9%
$^{237}Np\ (n,f)\ ^{137}Cs$	0.99	8.2%	-	-	0.99	8.2%
Linear Average	1.03	8.1%	0.95	7.4%	0.99	5.5%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) is summarized as follows:

Parameter	Unit 2 In-Vessel		Unit 2 Midplane Ex-Vessel		Unit 2 Combined	
	Average	% Standard Deviation	Average	% Standard Deviation	Average	% Standard Deviation
	BE/C	(1σ)	BE/C	(1σ)	BE/C	(1σ)
Fluence Rate (E > 1.0 MeV)	0.98	4.9%	0.98	3.2%	0.98	2.9%
dpa/s	0.98	4.0%	1.01	3.0%	0.99	2.5%

These data comparisons show similar and consistent results with the linear average M/C ratio of 0.99 in excellent agreement with the resultant least-squares BE/C ratios of 0.98 and 0.99 for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, respectively. The comparisons demonstrate that the calculated results for Braidwood Unit 2 provided in Section 2 of this report are validated within the context of the assigned 13% calculational uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by Reference 9 is easily met.

Table F-1 Nuclear Parameters Used in the Evaluation of Braidwood Unit 1 and Unit 2 In-Vessel Surveillance Capsule Neutron Sensors

Monitor Material	Reaction of Interest	Target Atom Fraction	90% Response Range (MeV)[(a)]	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	0.6917	4.9 – 11.9	5.272 y	
Iron	^{54}Fe (n,p)	0.0585	2.1 – 8.5	312.1 d	
Nickel	^{58}Ni (n,p)	0.6808	1.5 – 8.3	70.82 d	
Uranium-238	^{238}U (n,f)	1.0000	1.3 – 6.9	30.07 y	6.02
Neptunium-237	^{237}Np (n,f)	1.0000	0.3 – 3.8	30.07 y	6.17
Cobalt-Aluminum	^{59}Co (n,γ)	0.0015	non-threshold	5.272 y	

Note:

(a) The 90% response range is defined such that, in the neutron spectrum characteristic of the Braidwood Unit 1 and Unit 2 surveillance capsules, approximately 90% of the sensor response is due to neutrons in the energy range specified with approximately 5% of the total response due to neutrons with energies below the lower limit and 5% of the total response due to neutrons with energies above the upper limit

Table F-2 Braidwood Unit 1 Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	12-Jul-1987	2-Sep-1989
2	15-Dec-1989	1-Mar-1991
3	18-May-1991	5-Sep-1992
4	8-Nov-1992	4-Mar-1994
5	13-May-1994	30-Sep-1995
6	14-Dec-1995	29-Mar-1997
7	27-May-1997	5-Sep-1998
8	14-Nov-1998	18-Mar-2000
9	5-Apr-2000	22-Sep-2001
10	12-Oct-2001	15-Apr-2003
11	2-May-2003	4-Oct-2004
12	24-Oct-2004	16-Apr-2006
13	3-May-2006	30-Sep-2007
14	26-Oct-2007	29-Mar-2009
15	19-Apr-2009	3-Oct-2010
16	6-Nov-2010	16-Apr-2012
17	19-May-2012	9-Sep-2013
18	30-Sep-2013	30-Mar-2015
19	18-Apr-2015	26-Sep-2016

Table F-3 Braidwood Unit 2 Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	25-May-1988	16-Mar-1990
2	28-May-1990	13-Sep-1991
3	26-Nov-1991	5-Mar-1993
4a	2-May-1993	5-Apr-1994[(a)]
4b	1-May-1994	8-Oct-1994
5	17-Nov-1994	16-Mar-1996
6	14-May-1996	27-Sep-1997
7	14-Nov-1997	24-Apr-1999
8	20-May-1999	21-Oct-2000
9	5-Nov-2000	19-Apr-2002
10	11-May-2002	4-Nov-2003
11	19-Nov-2003	17-Apr-2005
12	5-May-2005	15-Oct-2006
13	2-Nov-2006	20-Apr-2008
14	17-May-2008	12-Oct-2009
15	31-Oct-2009	17-Apr-2011
16	12-May-2011	15-Oct-2012
17	08-Nov-2012	3-May-2014
18	22-May-2014	5-Oct-2015
19	26-Oct-2015	23-Apr-2017

Note:

(a) Capsule X was removed during a mid-cycle outage

Table F-4 Measured Sensor Activities and Reaction Rates for Braidwood Unit 1 Surveillance Capsule U

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
90-1044	^{63}Cu (n,α) ^{60}Co	5.16E+04	4.38E+05	6.68E-17	6.12E-17	6.12E-17
90-1049	^{63}Cu (n,α) ^{60}Co	4.53E+04	3.84E+05	5.86E-17		
90-1054	^{63}Cu (n,α) ^{60}Co	4.50E+04	3.82E+05	5.82E-17		
90-1045	^{54}Fe (n,p) ^{54}Mn	1.08E+06	3.98E+06	6.32E-15	5.93E-15	5.93E-15
90-1050	^{54}Fe (n,p) ^{54}Mn	9.89E+05	3.65E+06	5.79E-15		
90-1055	^{54}Fe (n,p) ^{54}Mn	9.74E+05	3.59E+06	5.70E-15		
90-1046	^{58}Ni (n,p) ^{58}Co	2.65E+06	5.58E+07	7.99E-15	7.53E-15	7.53E-15
90-1051	^{58}Ni (n,p) ^{58}Co	2.43E+06	5.12E+07	7.32E-15		
90-1056	^{58}Ni (n,p) ^{58}Co	2.42E+06	5.09E+07	7.29E-15		
90-1042	^{59}Co (n,γ) ^{60}Co	1.06E+07	8.99E+07	5.87E-12	5.86E-12	5.86E-12
90-1043	^{59}Co (n,γ) ^{60}Co	1.05E+07	8.91E+07	5.81E-12		
90-1047	^{59}Co (n,γ) ^{60}Co	1.08E+07	9.16E+07	5.98E-12		
90-1048	^{59}Co (n,γ) ^{60}Co	1.04E+07	8.82E+07	5.76E-12		
90-1053	^{59}Co (n,γ) ^{60}Co	1.06E+07	8.99E+07	5.87E-12		
90-1052	^{59}Co(Cd) (n,γ) ^{60}Co	5.43E+06	4.61E+07	3.01E-12	3.01E-12	3.01E-12
90-1040	^{238}U (n,f) ^{137}Cs	1.53E+05	6.25E+06	4.11E-14	4.11E-14	3.45E-14
90-1041	^{237}Np (n,f) ^{137}Cs	1.32E+06	5.40E+07	3.44E-13	3.44E-13	3.41E-13

Note:

(a) Measured activity decay corrected to May 31, 1990.

Table F-5 Measured Sensor Activities and Reaction Rates for Braidwood Unit 1 Surveillance Capsule X

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
94-1652	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.36E+05	3.79E+05	5.78E-17	5.43E-17	5.43E-17
94-1657	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.25E+05	3.48E+05	5.31E-17		
94-1661	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.22E+05	3.40E+05	5.19E-17		
94-1650	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.66E+06	3.38E+06	5.35E-15	4.98E-15	4.98E-15
94-1655	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.49E+06	3.03E+06	4.81E-15		
94-1659	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.48E+06	3.01E+06	4.77E-15		
94-1651	$^{58}Ni\ (n,p)\ ^{58}Co$	7.37E+06	5.02E+07	7.18E-15	6.71E-15	6.71E-15
94-1656	$^{58}Ni\ (n,p)\ ^{58}Co$	6.73E+06	4.58E+07	6.56E-15		
94-1660	$^{58}Ni\ (n,p)\ ^{58}Co$	6.56E+06	4.47E+07	6.39E-15		
94-1649	$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.49E+07	6.94E+07	4.53E-12	4.55E-12	4.55E-12
94-1653	$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.57E+07	7.16E+07	4.67E-12		
94-1658	$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.45E+07	6.83E+07	4.45E-12		
94-1661	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	1.26E+07	3.51E+07	2.29E-12	2.29E-12	2.29E-12
94-1650	$^{238}U\ (n,f)\ ^{137}Cs$	5.00E+05	5.57E+06	3.66E-14	3.66E-14	2.97E-14
94-1655	$^{237}Np\ (n,f)\ ^{137}Cs$	3.27E+06	3.64E+07	2.32E-13	2.32E-13	2.30E-13

Note:

(a) Measured activity decay corrected to August 19, 1994.

Table F-6 Measured Sensor Activities and Reaction Rates for Braidwood Unit 1 Surveillance Capsule W

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
99070215	^{63}Cu (n,α) ^{60}Co	1.70E+05	3.53E+05	5.39E-17	4.99E-17	4.99E-17
99070220	^{63}Cu (n,α) ^{60}Co	1.53E+05	3.18E+05	4.85E-17		
99070225	^{63}Cu (n,α) ^{60}Co	1.49E+05	3.10E+05	4.72E-17		
99070216	^{54}Fe (n,p) ^{54}Mn	1.33E+06	3.44E+06	5.46E-15	5.09E-15	5.09E-15
99070221	^{54}Fe (n,p) ^{54}Mn	1.21E+06	3.13E+06	4.96E-15		
99070226	^{54}Fe (n,p) ^{54}Mn	1.18E+06	3.05E+06	4.84E-15		
99070217	^{58}Ni (n,p) ^{58}Co	2.02E+06	5.39E+07	7.71E-15	7.18E-15	7.18E-15
99070222	^{58}Ni (n,p) ^{58}Co	1.86E+06	4.96E+07	7.10E-15		
99070227	^{58}Ni (n,p) ^{58}Co	1.76E+06	4.69E+07	6.72E-15		
99070214	^{59}Co (n,γ) ^{60}Co	3.01E+07	6.25E+07	4.08E-12	4.14E-12	4.14E-12
99070219	^{59}Co (n,γ) ^{60}Co	3.09E+07	6.42E+07	4.19E-12		
99070224	^{59}Co (n,γ) ^{60}Co	3.06E+07	6.36E+07	4.15E-12		
99070213	^{59}Co(Cd) (n,γ) ^{60}Co	2.46E+07	5.11E+07	3.34E-12	3.48E-12	3.48E-12
99070218	^{59}Co(Cd) (n,γ) ^{60}Co	2.67E+07	5.55E+07	3.62E-12		
99070211	^{238}U (n,f) ^{137}Cs	9.50E+05	6.24E+06	4.10E-14	4.10E-14	3.22E-14
99070212	^{237}Np (n,f) ^{137}Cs	6.30E+06	4.14E+07	2.64E-13	2.64E-13	2.62E-13

Note:

(a) Measured activity decay corrected to July 26, 1999.

Table F-7 Measured Sensor Activities and Reaction Rates for Braidwood Unit 1 Surveillance Capsule V

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
30160882005	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	9.70E+04	2.85E+05	4.35E-17	4.03E-17	4.03E-17
30160882010	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	8.67E+04	2.55E+05	3.88E-17		
30160882015	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	8.58E+04	2.52E+05	3.84E-17		
30160882007	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.15E+04	2.50E+06	3.96E-15	3.82E-15	3.82E-15
30160882012	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.09E+04	2.37E+06	3.76E-15		
30160882017	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.08E+04	2.35E+06	3.72E-15		
30160882003	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.58E+07	4.64E+07	3.03E-12	2.91E-12	2.91E-12
30160882004	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.46E+07	4.29E+07	2.80E-12		
30160882008	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.61E+07	4.73E+07	3.09E-12		
30160882013	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.52E+07	4.46E+07	2.91E-12		
30160882009	$^{59}Co\ (n,\gamma)\ ^{60}Co$(b)	1.43E+07	4.20E+07	2.74E-12		
30160882014	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	8.19E+06	2.41E+07	1.57E-12	1.57E-12	1.57E-12
30160882001	$^{238}U\ (n,f)\ ^{137}Cs$	1.37E+06	4.97E+06	3.26E-14	3.26E-14	2.38E-14
30160882002	$^{237}Np\ (n,f)\ ^{137}Cs$	8.67E+06	3.14E+07	2.01E-13	2.01E-13	1.99E-13

Notes:

(a) Measured activity decay corrected to October 1. 2015.

(b) Partially cadmium-covered; treated as bare.

Table F-8 Multiple Foil Sensor Set Locations within the Reactor Cavity for Braidwood Unit 1 and Unit 2

Azimuth (Degrees)	Capsule Identification – EVND Sets 1S-1 and 2S-1		
	Core Top	Core Midplane	Core Bottom
0		A	
15		B	
30		C	
45	D	E	F

Azimuth (Degrees)	Capsule Identification – EVND Sets 1S-2 and 2S-2		
	Core Top	Core Midplane	Core Bottom
0		G	
15		H	
30		I	
45	J	K	L

Table F-9 Nuclear Parameters Used in the Evaluation of Braidwood Unit 1 EVND Capsules Set 1S-1 and 1S-2 and Braidwood Unit 2 EVND Capsules Set 2S-1 and 2S-2

Reaction of Interest	Atomic Weight[(a)] (g/g-atom)	Target Atom Fraction[(b)]	Product Half-life[(b,c)] (days)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	63.546	0.6917	1925.5
$^{46}Ti\ (n,p)\ ^{46}Sc$	47.867	0.0825	83.79
$^{54}Fe\ (n,p)\ ^{54}Mn$	55.845	0.05845	312.11
$^{58}Ni\ (n,p)\ ^{58}Co$	58.693	0.68077	70.82
$^{93}Nb\ (n,n')\ ^{93m}Nb$	92.906	1.0	5890.0
$^{59}Co\ (n,\gamma)\ ^{60}Co$	58.933	0.00438	1925.5
$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	58.933	0.00438	1925.5

Notes:

(a) Atomic weight data were taken from the Chart of the Nuclides 17th Edition, dated 2009 [39]

(b) Half-life and target atom fraction data for ^{63}Cu (n,α), ^{46}Ti(n,p), ^{54}Fe(n,p), ^{58}Ni(n,p), and ^{93}Nb (n,n') reactions were taken from ASTM Standard E 1005-10 [40].

(c) The half-life for the ^{59}Co (n,γ) reaction was taken from ASTM Standard E 1005-10 [40]

Table F-10 Reaction Rates for Braidwood Unit 1 EVND Capsule A; 0.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216005	^{63}Cu (n,α) ^{60}Co	4.29E+02	2.70E+03	4.12E-19	4.12E-19
3014216007	^{46}Ti (n,p) ^{46}Sc	1.31E+03	5.53E+03	5.33E-18	5.33E-18
3014216001	^{54}Fe (n,p) ^{54}Mn	8.81E+03	1.90E+04	3.01E-17	2.98E-17
3014216002	^{54}Fe (n,p) ^{54}Mn	8.63E+03	1.86E+04	2.95E-17	
3014216004	^{58}Ni (n,p) ^{58}Co	4.78E+04	2.60E+05	3.72E-17	3.72E-17
3014216107	^{93}Nb (n,n') ^{93m}Nb	4.88E+04	8.49E+05	1.31E-16	1.31E-16
3014216108	^{59}Co (n,γ) ^{60}Co	3.52E+05	2.22E+06	4.96E-14	4.96E-14
3014216009	^{59}Co(Cd) (n,γ) ^{60}Co	1.78E+05	1.12E+06	2.51E-14	2.51E-14

Note:

(a) Measured activity decay corrected to September 17, 2009.

Table F-11 Reaction Rates for Braidwood Unit 1 EVND Capsule B; 14.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216014	^{63}Cu (n,α) ^{60}Co	5.46E+02	3.44E+03	5.25E-19	5.25E-19
3014216016	^{46}Ti (n,p) ^{46}Sc	1.90E+03	8.02E+03	7.73E-18	7.73E-18
3014216010	^{54}Fe (n,p) ^{54}Mn	1.20E+04	2.59E+04	4.11E-17	4.04E-17
3014216011	^{54}Fe (n,p) ^{54}Mn	1.16E+04	2.50E+04	3.97E-17	
3014216013	^{58}Ni (n,p) ^{58}Co	6.46E+04	3.51E+05	5.03E-17	5.03E-17
3014216108	^{93}Nb (n,n') ^{93m}Nb	6.06E+04	1.05E+06	1.63E-16	1.63E-16
3014216017	^{59}Co (n,γ) ^{60}Co	6.28E+05	3.96E+06	8.84E-14	8.84E-14
3014216018	^{59}Co(Cd) (n,γ) ^{60}Co	2.62E+05	1.65E+06	3.69E-14	3.69E-14

Note.

(a) Measured activity decay corrected to September 17, 2009.

Table F-12 Reaction Rates for Braidwood Unit 1 EVND Capsule C; 29.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216023	^{63}Cu (n,α) ^{60}Co	5.22E+02	3.29E+03	5.02E-19	5.02E-19
3014216025	^{46}Ti (n,p) ^{46}Sc	1.77E+03	7.47E+03	7.20E-18	7.20E-18
3014216019	^{54}Fe (n,p) ^{54}Mn	1.10E+04	2.37E+04	3.76E-17	3.75E-17
3014216020	^{54}Fe (n,p) ^{54}Mn	1.09E+04	2.35E+04	3.73E-17	
3014216022	^{58}Ni (n,p) ^{58}Co	6.73E+04	3.66E+05	5.24E-17	5.24E-17
3014216109	^{93}Nb (n,n') ^{93m}Nb	6.86E+04	1.19E+06	1.84E-16	1.84E-16
3014216026	^{59}Co (n,γ) ^{60}Co	6.79E+05	4.28E+06	9.56E-14	9.56E-14
3014216027	^{59}Co(Cd) (n,γ) ^{60}Co	3.10E+05	1.95E+06	4.36E-14	4.36E-14

Note:

(a) Measured activity decay corrected to September 17, 2009.

Table F-13 Reaction Rates for Braidwood Unit 1 EVND Capsule D; 44.5° at 182.88 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216032	^{63}Cu (n,α) ^{60}Co	1.11E+02	6.99E+02	1.07E-19	1.07E-19
3014216034	^{46}Ti (n,p) ^{46}Sc	4.42E+02	1.87E+03	1.80E-18	1.80E-18
3014216028	^{54}Fe (n,p) ^{54}Mn	2.70E+03	5.82E+03	9.24E-18	9.56E-18
3014216029	^{54}Fe (n,p) ^{54}Mn	2.89E+03	6.23E+03	9.89E-18	
3014216031	^{58}Ni (n,p) ^{58}Co	1.95E+04	1.06E+05	1.52E-17	1.52E-17
3014216110	^{93}Nb (n,n') ^{93m}Nb	2.26E+04	3.93E+05	6.06E-17	6.06E-17
3014216035	^{59}Co (n,γ) ^{60}Co	1.62E+05	1.02E+06	2.28E-14	2.28E-14
3014216036	^{59}Co(Cd) (n,γ) ^{60}Co	9.84E+04	6.20E+05	1.39E-14	1.39E-14

Note

(a) Measured activity decay corrected to September 17, 2009.

Table F-14 Reaction Rates for Braidwood Unit 1 EVND Capsule E; 44.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216041	^{63}Cu (n,α) ^{60}Co	3.64E+02	2.29E+03	3.50E-19	3.50E-19
3014216043	^{46}Ti (n,p) ^{46}Sc	1.19E+03	5.02E+03	4.84E-18	4.84E-18
3014216037	^{54}Fe (n,p) ^{54}Mn	8.76E+03	1.89E+04	3.00E-17	3.00E-17
3014216038	^{54}Fe (n,p) ^{54}Mn	8.78E+03	1.89E+04	3.00E-17	
3014216040	^{58}Ni (n,p) ^{58}Co	5.07E+04	2.76E+05	3.95E-17	3.95E-17
3014216111	^{93}Nb (n,n') ^{93m}Nb	6.57E+04	1.14E+06	1.76E-16	1.76E-16
3014216044	^{59}Co (n,γ) ^{60}Co	3.79E+05	2.39E+06	5.34E-14	5.34E-14
3014216045	^{59}Co(Cd) (n,γ) ^{60}Co	2.27E+05	1.43E+06	3.20E-14	3.20E-14

Note:

(a) Measured activity decay corrected to September 17, 2009.

Table F-15 Reaction Rates for Braidwood Unit 1 EVND Capsule F; 44.5° at -182.88 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3014216050	^{63}Cu (n,α) ^{60}Co	1.49E+02	9.39E+02	1.43E-19	1.43E-19
3014216052	^{46}Ti (n,p) ^{46}Sc	5.64E+02	2.38E+03	2.29E-18	2.29E-18
3014216046	^{54}Fe (n,p) ^{54}Mn	4.20E+03	9.06E+03	1.44E-17	1.37E-17
3014216047	^{54}Fe (n,p) ^{54}Mn	3.78E+03	8.15E+03	1.29E-17	
3014216049	^{58}Ni (n,p) ^{58}Co	2.46E+04	1.34E+05	1.92E-17	1.92E-17
3014216112	^{93}Nb (n,n') ^{93m}Nb	2.62E+04	4.56E+05	7.03E-17	7.03E-17
3014216053	^{59}Co (n,γ) ^{60}Co	2.15E+05	1.36E+06	3.03E-14	3.03E-14
3014216054	^{59}Co(Cd) (n,γ) ^{60}Co	1.24E+05	7.81E+05	1.75E-14	1.75E-14

Note:

(a) Measured activity decay corrected to September 17, 2009.

Table F-16 Reaction Rates for Braidwood Unit 1 EVND Capsule G; 0.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068001	^{54}Fe (n,p) ^{54}Mn	1.58E+04	1.87E+04	2.97E-17	2.99E-17
4068002	^{54}Fe (n,p) ^{54}Mn	1.60E+04	1.90E+04	3.01E-17	
4068003	^{58}Ni (n,p) ^{58}Co	1.34E+05	2.59E+05	3.70E-17	3.70E-17
4068004	^{63}Cu (n,α) ^{60}Co	1.55E+03	2.66E+03	4.05E-19	4.05E-19
4068005	^{46}Ti (n,p) ^{46}Sc	3.09E+03	5.39E+03	5.20E-18	5.20E-18
4068006	^{93}Nb (n,n') ^{93m}Nb	2.25E+05	8.70E+05	1.34E-16	1.34E-16
4068007	^{59}Co (n,γ) ^{60}Co	1.31E+06	2.25E+06	5.02E-14	5.02E-14
4068008	^{59}Co(Cd) (n,γ) ^{60}Co	6.71E+05	1.15E+06	2.57E-14	2.57E-14

Note

(a) Measured activity decay corrected to November 30, 2016.

Table F-17 Reaction Rates for Braidwood Unit 1 EVND Capsule H; 14.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068009	^{54}Fe (n,p) ^{54}Mn	2.30E+04	2.70E+04	4.29E-17	4.24E-17
4068010	^{54}Fe (n,p) ^{54}Mn	2.25E+04	2.64E+04	4.20E-17	
4068011	^{58}Ni (n,p) ^{58}Co	1.98E+05	3.78E+05	5.42E-17	5.42E-17
4068012	^{63}Cu (n,α) ^{60}Co	2.08E+03	3.56E+03	5.43E-19	5.43E-19
4068013	^{46}Ti (n,p) ^{46}Sc	4.41E+03	7.62E+03	7.34E-18	7.34E-18
4068014	^{93}Nb (n,n') ^{93m}Nb	3.13E+05	1.21E+06	1.87E-16	1.87E-16
4068015	^{59}Co (n,γ) ^{60}Co	2.38E+06	4.07E+06	9.10E-14	9.10E-14
4068016	^{59}Co(Cd) (n,γ) ^{60}Co	1.04E+06	1.78E+06	3.98E-14	3.98E-14

Note:

(a) Measured activity decay corrected to November 30, 2016.

Table F-18 Reaction Rates for Braidwood Unit 1 EVND Capsule I; 29.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068017	^{54}Fe (n,p) ^{54}Mn	2.33E+04	2.72E+04	4.31E-17	4.32E-17
4068018	^{54}Fe (n,p) ^{54}Mn	2.34E+04	2.73E+04	4.33E-17	
4068019	^{58}Ni (n,p) ^{58}Co	2.08E+05	3.92E+05	5.61E-17	5.61E-17
4068020	^{63}Cu (n,α) ^{60}Co	2.02E+03	3.45E+03	5.27E-19	5.27E-19
4068021	^{46}Ti (n,p) ^{46}Sc	4.44E+03	7.57E+03	7.29E-18	7.29E-18
4068022	^{93}Nb (n,n') ^{93m}Nb	3.41E+05	1.32E+06	2.03E-16	2.03E-16
4068023	^{59}Co (n,γ) ^{60}Co	2.63E+06	4.50E+06	1.01E-13	1.01E-13
4068024	^{59}Co(Cd) (n,γ) ^{60}Co	1.20E+06	2.05E+06	4.58E-14	4.58E-14

Note:

(a) Measured activity decay corrected to November 30, 2016.

Table F-19 Reaction Rates for Braidwood Unit 1 EVND Capsule J; 44.5° at 182.88 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068025	^{54}Fe (n,p) ^{54}Mn	6.83E+03	7.88E+03	1.25E-17	1.31E-17
4068026	^{54}Fe (n,p) ^{54}Mn	7.43E+03	8.57E+03	1.36E-17	
4068027	^{58}Ni (n,p) ^{58}Co	8.00E+04	1.48E+05	2.12E-17	2.12E-17
4068028	^{63}Cu (n,α) ^{60}Co	5.53E+02	9.46E+02	1.44E-19	1.44E-19
4068029	^{46}Ti (n,p) ^{46}Sc	1.47E+03	2.47E+03	2.38E-18	2.38E-18
4068030	^{93}Nb (n,n') ^{93m}Nb	1.45E+05	5.60E+05	8.64E-17	8.64E-17
4068031	^{59}Co (n,γ) ^{60}Co	6.95E+05	1.19E+06	2.66E-14	2.66E-14
4068032	^{59}Co(Cd) (n,γ) ^{60}Co	4.36E+05	7.46E+05	1.67E-14	1.67E-14

Note:

(a) Measured activity decay corrected to November 30, 2016.

Table F-20 Reaction Rates for Braidwood Unit 1 EVND Capsule K; 44.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068033	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.72E+04	2.02E+04	3.21E-17	3.15E-17
4068034	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.66E+04	1.95E+04	3.10E-17	
4068035	$^{58}Ni\ (n,p)\ ^{58}Co$	1.47E+05	2.80E+05	4.00E-17	4.00E-17
4068036	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.38E+03	2.37E+03	3.61E-19	3.61E-19
4068037	$^{46}Ti\ (n,p)\ ^{46}Sc$	3.01E+03	5.18E+03	4.99E-18	4.99E-18
4068038	$^{93}Nb\ (n,n')\ ^{93m}Nb$	3.14E+05	1.21E+06	1.87E-16	1.87E-16
4068039	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.43E+06	2.45E+06	5.48E-14	5.48E-14
4068040	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	8.81E+05	1.51E+06	3.37E-14	3.37E-14

Note
(a) Measured activity decay corrected to November 30, 2016.

Table F-21 Reaction Rates for Braidwood Unit 1 EVND Capsule L; 44.5° at -182.88 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4068041	$^{54}Fe\ (n,p)\ ^{54}Mn$	6.16E+03	7.47E+03	1.19E-17	1.13E-17
4068042	$^{54}Fe\ (n,p)\ ^{54}Mn$	5.60E+03	6.79E+03	1.08E-17	
4068043	$^{58}Ni\ (n,p)\ ^{58}Co$	5.67E+04	1.13E+05	1.61E-17	1.61E-17
4068044	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	4.28E+02	7.38E+02	1.13E-19	1.13E-19
4068045	$^{46}Ti\ (n,p)\ ^{46}Sc$	1.11E+03	2.00E+03	1.92E-18	1.92E-18
4068046	$^{93}Nb\ (n,n')\ ^{93m}Nb$	9.92E+04	3.84E+05	5.93E-17	5.93E-17
4068047	$^{59}Co\ (n,\gamma)\ ^{60}Co$	8.94E+05	1.54E+06	3.45E-14	3.45E-14
4068048	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	4.42E+05	7.62E+05	1.70E-14	1.70E-14

Note
(a) Measured activity decay corrected to November 30, 2016.

Table F-22 Measured Sensor Activities and Reaction Rates for Braidwood Unit 2 Surveillance Capsule U

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
901634	^{63}Cu (n,α) ^{60}Co	5.41E+04	4.30E+05	6.55E-17	6.06E-17	6.06E-17
901639	^{63}Cu (n,α) ^{60}Co	4.78E+04	3.80E+05	5.79E-17		
901644	^{63}Cu (n,α) ^{60}Co	4.81E+04	3.82E+05	5.83E-17		
901636	^{54}Fe (n,p) ^{54}Mn	1.31E+06	4.03E+06	6.40E-15	5.88E-15	5.88E-15
901641	^{54}Fe (n,p) ^{54}Mn	1.16E+06	3.57E+06	5.66E-15		
901646	^{54}Fe (n,p) ^{54}Mn	1.14E+06	3.51E+06	5.57E-15		
901635	^{58}Ni (n,p) ^{58}Co	4.99E+06	5.78E+07	8.28E-15	7.68E-15	7.68E-15
901640	^{58}Ni (n,p) ^{58}Co	4.46E+06	5.17E+07	7.40E-15		
901645	^{58}Ni (n,p) ^{58}Co	4.44E+06	5.14E+07	7.36E-15		
901632	^{59}Co (n,γ) ^{60}Co	1.08E+07	8.58E+07	5.60E-12	5.54E-12	5.54E-12
901637	^{59}Co (n,γ) ^{60}Co	1.07E+07	8.50E+07	5.54E-12		
901642	^{59}Co (n,γ) ^{60}Co	1.06E+07	8.42E+07	5.49E-12		
901633	^{59}Co(Cd) (n,γ) ^{60}Co	5.42E+06	4.30E+07	2.81E-12	2.86E-12	2.86E-12
901638	^{59}Co(Cd) (n,γ) ^{60}Co	5.57E+06	4.42E+07	2.89E-12		
901643	^{59}Co(Cd) (n,γ) ^{60}Co	5.56E+06	4.42E+07	2.88E-12		
901630	^{238}U (n,f) ^{137}Cs	1.54E+05	6.01E+06	3.95E-14	3.95E-14	3.32E-14
901631	^{237}Np (n,f) ^{137}Cs	1.39E+06	5.43E+07	3.46E-13	3.46E-13	3.43E-13

Note:

(a) Measured activity decay corrected to October 17, 1990.

Table F-23 Measured Sensor Activities and Reaction Rates for Braidwood Unit 2 Surveillance Capsule X

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
901681	^{63}Cu (n,α) ^{60}Co	1.35E+05	3.68E+05	5.61E-17	5.24E-17	5.24E-17
901687	^{63}Cu (n,α) ^{60}Co	1.23E+05	3.35E+05	5.11E-17		
901693	^{63}Cu (n,α) ^{60}Co	1.20E+05	3.27E+05	4.99E-17		
901679	^{54}Fe (n,p) ^{54}Mn	1.71E+06	3.23E+06	5.13E-15	4.76E-15	4.76E-15
901685	^{54}Fe (n,p) ^{54}Mn	1.53E+06	2.89E+06	4.59E-15		
901691	^{54}Fe (n,p) ^{54}Mn	1.52E+06	2.87E+06	4.56E-15		
901680	^{58}Ni (n,p) ^{58}Co	9.39E+06	4.92E+07	7.04E-15	6.58E-15	6.58E-15
901686	^{58}Ni (n,p) ^{58}Co	8.48E+06	4.44E+07	6.36E-15		
901692	^{58}Ni (n,p) ^{58}Co	8.43E+06	4.42E+07	6.32E-15		
901678	^{59}Co (n,γ) ^{60}Co	2.27E+07	6.18E+07	4.03E-12	4.11E-12	4.11E-12
901684	^{59}Co (n,γ) ^{60}Co	2.35E+07	6.40E+07	4.18E-12		
901690	^{59}Co (n,γ) ^{60}Co	2.32E+07	6.32E+07	4.12E-12		
901677	^{59}Co(Cd) (n,γ) ^{60}Co	1.20E+07	3.27E+07	2.13E-12	2.19E-12	2.19E-12
901683	^{59}Co(Cd) (n,γ) ^{60}Co	1.24E+07	3.38E+07	2.20E-12		
901689	^{59}Co(Cd) (n,γ) ^{60}Co	1.26E+07	3.43E+07	2.24E-12		
901676	^{238}U (n,f) ^{137}Cs	4.84E+05	5.38E+06	3.54E-14	3.54E-14	2.87E-14
901675	^{237}Np (n,f) ^{137}Cs	3.31E+06	3.68E+07	2.35E-13	2.35E-13	2.33E-13

Note:

(a) Measured activity decay corrected to August 30, 1994

Table F-24 Measured Sensor Activities and Reaction Rates for Braidwood Unit 2 Surveillance Capsule W

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
99090617	^{63}Cu (n,α) ^{60}Co	1.88E+05	3.35E+05	5.12E-17	4.73E-17	4.73E-17
99090622	^{63}Cu (n,α) ^{60}Co	1.68E+05	3.00E+05	4.57E-17		
99090627	^{63}Cu (n,α) ^{60}Co	1.65E+05	2.94E+05	4.49E-17		
99090619	^{54}Fe (n,p) ^{54}Mn	1.76E+06	3.19E+06	5.06E-15	4.68E-15	4.68E-15
99090624	^{54}Fe (n,p) ^{54}Mn	1.55E+06	2.81E+06	4.46E-15		
99090629	^{54}Fe (n,p) ^{54}Mn	1.57E+06	2.85E+06	4.52E-15		
99090618	^{58}Ni (n,p) ^{58}Co	7.41E+06	4.95E+07	7.09E-15	6.63E-15	6.63E-15
99090623	^{58}Ni (n,p) ^{58}Co	6.66E+06	4.45E+07	6.37E-15		
99090628	^{58}Ni (n,p) ^{58}Co	6.72E+06	4.49E+07	6.43E-15		
99090615	^{59}Co (n,γ) ^{60}Co	3.14E+07	5.60E+07	3.66E-12	3.68E-12	3.68E-12
99090620	^{59}Co (n,γ) ^{60}Co	3.18E+07	5.67E+07	3.70E-12		
99090625	^{59}Co (n,γ) ^{60}Co	3.15E+07	5.62E+07	3.67E-12		
99090616	^{59}Co(Cd) (n,γ) ^{60}Co	1.60E+07	2.86E+07	1.86E-12	1.93E-12	1.93E-12
99090621	^{59}Co(Cd) (n,γ) ^{60}Co	1.66E+07	2.96E+07	1.93E-12		
99090626	^{59}Co(Cd) (n,γ) ^{60}Co	1.70E+07	3.03E+07	1.98E-12		
99090613	^{238}U (n,f) ^{137}Cs	8.49E+05	4.94E+06	3.25E-14	3.25E-14	2.54E-14
99090614	^{237}Np (n,f) ^{137}Cs	6.60E+06	3.84E+07	2.45E-13	2.45E-13	2.43E-13

Note:

(a) Measured activity decay corrected to October 15, 1999.

Table F-25 Measured Sensor Activities and Reaction Rates for Braidwood Unit 2 Surveillance Capsule V

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
65273005	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.06E+05	2.87E+05	4.38E-17		
65273010	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	9.36E+04	2.54E+05	3.87E-17	4.03E-17	4.03E-17
65273015	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	9.24E+04	2.51E+05	3.82E-17		
65273007	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.78E+04	2.61E+06	4.14E-15		
65273012	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.56E+04	2.28E+06	3.62E-15	3.83E-15	3.83E-15
65273017	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.60E+04	2.34E+06	3.72E-15		
65273003	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.55E+07	4.20E+07	2.74E-12		
65273008	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.63E+07	4.42E+07	2.88E-12	2.82E-12	2.82E-12
65273013	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.60E+07	4.34E+07	2.83E-12		
65273004	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	8.73E+06	2.37E+07	1.54E-12		
65273009	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	8.82E+06	2.39E+07	1.56E-12	1.55E-12	1.55E-12
65273014	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	8.78E+06	2.38E+07	1.55E-12		
65273001	$^{238}U\ (n,f)\ ^{137}Cs$	1.34E+06	4.61E+06	3.02E-14	3.02E-14	2.20E-14
65273002	$^{237}Np\ (n,f)\ ^{137}Cs$	9.17E+06	3.15E+07	2.01E-13	2.01E-13	1.99E-13

Note:

(a) Measured activity decay corrected to October 29, 2015.

Table F-26 Reaction Rates for Braidwood Unit 2 EVND Capsule A; 0.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217001	^{54}Fe (n,p) ^{54}Mn	9.85E+03	1.73E+04	2.74E-17	2.74E-17
9217002	^{54}Fe (n,p) ^{54}Mn	9.85E+03	1.73E+04	2.74E-17	
9217003	^{58}Ni (n,p) ^{58}Co	1.26E+05	2.54E+05	3.64E-17	3.64E-17
9217004	^{63}Cu (n,α) ^{60}Co	4.28E+02	2.67E+03	4.07E-19	4.07E-19
9217005	^{46}Ti (n,p) ^{46}Sc	2.98E+03	5.46E+03	5.26E-18	5.26E-18
9217006	^{59}Co (n,γ) ^{60}Co	3.14E+05	1.96E+06	3.41E-14	3.41E-14
9217007	^{59}Co(Cd) (n,γ) ^{60}Co	1.73E+05	1.08E+06	1.88E-14	1.88E-14
9217095	^{93}Nb (n,n') ^{93m}Nb	4.74E+04	8.36E+05	1.29E-16	1.29E-16

Note:

(a) Measured activity decay corrected to December 17, 2009.

Table F-27 Reaction Rates for Braidwood Unit 2 EVND Capsule B; 14.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217008	^{54}Fe (n,p) ^{54}Mn	1.39E+04	2.44E+04	3.87E-17	3.83E-17
9217009	^{54}Fe (n,p) ^{54}Mn	1.36E+04	2.39E+04	3.79E-17	
9217010	^{58}Ni (n,p) ^{58}Co	1.64E+05	3.30E+05	4.73E-17	4.73E-17
9217011	^{63}Cu (n,α) ^{60}Co	5.44E+02	3.39E+03	5.17E-19	5.17E-19
9217012	^{46}Ti (n,p) ^{46}Sc	3.88E+03	7.11E+03	6.85E-18	6.85E-18
9217013	^{59}Co (n,γ) ^{60}Co	5.22E+05	3.25E+06	5.66E-14	5.66E-14
9217014	^{59}Co(Cd) (n,γ) ^{60}Co	2.67E+05	1.66E+06	2.90E-14	2.90E-14
8605096	^{93}Nb (n,n') ^{93m}Nb	6.16E+04	1.09E+06	1.68E-16	1.68E-16

Note:

(a) Measured activity decay corrected to December 17, 2009.

Table F-28 Reaction Rates for Braidwood Unit 2 EVND Capsule C; 29.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217015	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.42E+04	2.49E+04	3.95E-17	3.95E-17
9217016	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.42E+04	2.49E+04	3.95E-17	
9217017	$^{58}Ni\ (n,p)\ ^{58}Co$	1.75E+05	3.53E+05	5.05E-17	5.05E-17
9217018	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	5.41E+02	3.37E+03	5.14E-19	5.14E-19
9217019	$^{46}Ti\ (n,p)\ ^{46}Sc$	3.83E+03	7.02E+03	6.76E-18	6.76E-18
9217020	$^{59}Co\ (n,\gamma)\ ^{60}Co$	5.92E+05	3.69E+06	6.42E-14	6.42E-14
9217021	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	3.03E+05	1.89E+06	3.29E-14	3.29E-14
8605097	$^{93}Nb\ (n,n')\ ^{93m}Nb$	6.87E+04	1.21E+06	1.87E-16	1.87E-16

Note:

(a) Measured activity decay corrected to December 17, 2009

Table F-29 Reaction Rates for Braidwood Unit 2 EVND Capsule D; 44.5° at 182.88 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217022	$^{54}Fe\ (n,p)\ ^{54}Mn$	3.81E+03	6.69E+03	1.06E-17	1.10E-17
9217023	$^{54}Fe\ (n,p)\ ^{54}Mn$	4.09E+03	7.18E+03	1.14E-17	
9217024	$^{58}Ni\ (n,p)\ ^{58}Co$	5.87E+04	1.18E+05	1.69E-17	1.69E-17
9217025	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.35E+02	8.41E+02	1.28E-19	1.28E-19
9217026	$^{46}Ti\ (n,p)\ ^{46}Sc$	1.17E+03	2.14E+03	2.07E-18	2.07E-18
9217027	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.71E+05	1.07E+06	1.86E-14	1.86E-14
9217028	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	9.90E+04	6.17E+05	1.07E-14	1.07E-14
8605098	$^{93}Nb\ (n,n')\ ^{93m}Nb$	2.61E+04	4.60E+05	7.10E-17	7.10E-17

Note:

(a) Measured activity decay corrected to December 17, 2009.

Table F-30 Reaction Rates for Braidwood Unit 2 EVND Capsule E; 44.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217029	^{54}Fe (n,p) ^{54}Mn	1.04E+04	1.83E+04	2.90E-17	2.85E-17
9217030	^{54}Fe (n,p) ^{54}Mn	1.01E+04	1.77E+04	2.81E-17	
9217031	^{58}Ni (n,p) ^{58}Co	1.32E+05	2.66E+05	3.81E-17	3.81E-17
9217032	^{63}Cu (n,α) ^{60}Co	3.67E+02	2.29E+03	3.49E-19	3.49E-19
9217033	^{46}Ti (n,p) ^{46}Sc	2.62E+03	4.80E+03	4.63E-18	4.63E-18
9217034	^{59}Co (n,γ) ^{60}Co	3.66E+05	2.28E+06	3.97E-14	3.97E-14
9217035	^{59}Co(Cd) (n,γ) ^{60}Co	2.22E+05	1.38E+06	2.41E-14	2.41E-14
8605099	^{93}Nb (n,n') ^{93m}Nb	6.43E+04	1.13E+06	1.75E-16	1.75E-16

Note.

(a) Measured activity decay corrected to December 17, 2009.

Table F-31 Reaction Rates for Braidwood Unit 2 EVND Capsule F; 44.5° at -182.88 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
9217036	^{54}Fe (n,p) ^{54}Mn	4.15E+03	7.28E+03	1.16E-17	1.12E-17
9217037	^{54}Fe (n,p) ^{54}Mn	3.89E+03	6.83E+03	1.08E-17	
9217038	^{58}Ni (n,p) ^{58}Co	5.70E+04	1.15E+05	1.64E-17	1.64E-17
9217039	^{63}Cu (n,α) ^{60}Co	1.30E+02	8.10E+02	1.24E-19	1.24E-19
9217040	^{46}Ti (n,p) ^{46}Sc	1.07E+03	1.96E+03	1.89E-18	1.89E-18
9217041	^{59}Co (n,γ) ^{60}Co	2.52E+05	1.57E+06	2.73E-14	2.73E-14
9217042	^{59}Co(Cd) (n,γ) ^{60}Co	1.23E+05	7.66E+05	1.33E-14	1.33E-14
8605100	^{93}Nb (n,n') ^{93m}Nb	2.13E+04	3.76E+05	5.80E-17	5.80E-17

Note:

(a) Measured activity decay corrected to December 17, 2009.

Table F-32 Reaction Rates for Braidwood Unit 2 EVND Capsule G; 0.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288001	^{54}Fe (n,p) ^{54}Mn	1.51E+04	1.85E+04	2.94E-17	2.95E-17
4288002	^{54}Fe (n,p) ^{54}Mn	1.52E+04	1.87E+04	2.96E-17	
4288003	^{58}Ni (n,p) ^{58}Co	1.32E+05	2.75E+05	3.94E-17	3.94E-17
4288004	^{63}Cu (n,α) ^{60}Co	1.51E+03	2.59E+03	3.96E-19	3.96E-19
4288005	^{46}Ti (n,p) ^{46}Sc	2.94E+03	5.51E+03	5.31E-18	5.31E-18
4288006	^{59}Co (n,γ) ^{60}Co	1.15E+06	1.98E+06	3.44E-14	3.44E-14
4288007	^{59}Co(Cd) (n,γ) ^{60}Co	6.37E+05	1.09E+06	1.91E-14	1.91E-14
4288008	^{93}Nb (n,n') ^{93m}Nb	2.22E+05	8.54E+05	1.32E-16	1.32E-16

Note:
(a) Measured activity decay corrected to July 1, 2017

Table F-33 Reaction Rates for Braidwood Unit 2 EVND Capsule H; 14.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288009	^{54}Fe (n,p) ^{54}Mn	2.03E+04	2.50E+04	3.97E-17	3.96E-17
4288010	^{54}Fe (n,p) ^{54}Mn	2.02E+04	2.49E+04	3.95E-17	
4288011	^{58}Ni (n,p) ^{58}Co	1.80E+05	3.71E+05	5.31E-17	5.31E-17
4288012	^{63}Cu (n,α) ^{60}Co	1.94E+03	3.34E+03	5.10E-19	5.10E-19
4288013	^{46}Ti (n,p) ^{46}Sc	3.88E+03	7.19E+03	6.93E-18	6.93E-18
4288014	^{59}Co (n,γ) ^{60}Co	2.05E+06	3.53E+06	6.15E-14	6.15E-14
4288015	^{59}Co(Cd) (n,γ) ^{60}Co	9.82E+05	1.69E+06	2.95E-14	2.95E-14
4288016	^{93}Nb (n,n') ^{93m}Nb	2.99E+05	1.15E+06	1.78E-16	1.78E-16

Note.
(a) Measured activity decay corrected to July 1, 2017.

Table F-34 Reaction Rates for Braidwood Unit 2 EVND Capsule I; 29.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288017	^{54}Fe (n,p) ^{54}Mn	2.15E+04	2.62E+04	4.15E-17	4.15E-17
4288018	^{54}Fe (n,p) ^{54}Mn	2.15E+04	2.62E+04	4.15E-17	
4288019	^{58}Ni (n,p) ^{58}Co	1.79E+05	3.60E+05	5.15E-17	5.15E-17
4288020	^{63}Cu (n,α) ^{60}Co	1.92E+03	3.30E+03	5.04E-19	5.04E-19
4288021	^{46}Ti (n,p) ^{46}Sc	4.10E+03	7.41E+03	7.14E-18	7.14E-18
4288022	^{59}Co (n,γ) ^{60}Co	2.34E+06	4.03E+06	7.01E-14	7.01E-14
4288023	^{59}Co(Cd) (n,γ) ^{60}Co	1.12E+06	1.93E+06	3.36E-14	3.36E-14
4288024	^{93}Nb (n,n') ^{93m}Nb	3.38E+05	1.30E+06	2.01E-16	2.01E-16

Note

(a) Measured activity decay corrected to July 1, 2017.

Table F-35 Reaction Rates for Braidwood Unit 2 EVND Capsule J; 44.5° at 182.88 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288025	^{54}Fe (n,p) ^{54}Mn	5.96E+03	7.24E+03	1.15E-17	1.17E-17
4288026	^{54}Fe (n,p) ^{54}Mn	6.16E+03	7.48E+03	1.19E-17	
4288027	^{58}Ni (n,p) ^{58}Co	6.34E+04	1.28E+05	1.83E-17	1.83E-17
4288028	^{63}Cu (n,α) ^{60}Co	4.87E+02	8.37E+02	1.28E-19	1.28E-19
4288029	^{46}Ti (n,p) ^{46}Sc	1.29E+03	2.34E+03	2.26E-18	2.26E-18
4288030	^{59}Co (n,γ) ^{60}Co	6.15E+05	1.06E+06	1.84E-14	1.84E-14
4288031	^{59}Co(Cd) (n,γ) ^{60}Co	3.66E+05	6.29E+05	1.10E-14	1.10E-14
4288032	^{93}Nb (n,n') ^{93m}Nb	1.25E+05	4.81E+05	7.42E-17	7.42E-17

Note:

(a) Measured activity decay corrected to July 1, 2017.

Table F-36 Reaction Rates for Braidwood Unit 2 EVND Capsule K; 44.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288033	^{54}Fe (n,p) ^{54}Mn	1.62E+04	1.95E+04	3.10E-17	3.05E-17
4288034	^{54}Fe (n,p) ^{54}Mn	1.57E+04	1.89E+04	3.01E-17	
4288035	^{58}Ni (n,p) ^{58}Co	1.47E+05	2.95E+05	4.23E-17	4.23E-17
4288036	^{63}Cu (n,α) ^{60}Co	1.31E+03	2.25E+03	3.43E-19	3.43E-19
4288037	^{46}Ti (n,p) ^{46}Sc	2.78E+03	5.02E+03	4.84E-18	4.84E-18
4288038	^{59}Co (n,γ) ^{60}Co	1.34E+06	2.30E+06	4.00E-14	4.00E-14
4288039	^{59}Co(Cd) (n,γ) ^{60}Co	8.11E+05	1.39E+06	2.42E-14	2.42E-14
4288040	^{93}Nb (n,n') ^{93m}Nb	3.18E+05	1.22E+06	1.89E-16	1.89E-16

Note:

(a) Measured activity decay corrected to July 1, 2017.

Table F-37 Reaction Rates for Braidwood Unit 2 EVND Capsule L; 44.5° at -182.88 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
4288041	^{54}Fe (n,p) ^{54}Mn	6.67E+03	8.00E+03	1.27E-17	1.20E-17
4288042	^{54}Fe (n,p) ^{54}Mn	5.95E+03	7.13E+03	1.13E-17	
4288043	^{58}Ni (n,p) ^{58}Co	5.90E+04	1.17E+05	1.68E-17	1.68E-17
4288044	^{63}Cu (n,α) ^{60}Co	4.71E+02	8.07E+02	1.23E-19	1.23E-19
4288045	^{46}Ti (n,p) ^{46}Sc	1.17E+03	2.09E+03	2.02E-18	2.02E-18
4288046	^{59}Co (n,γ) ^{60}Co	1.98E+06	3.39E+06	5.91E-14	5.91E-14
4288047	^{59}Co(Cd) (n,γ) ^{60}Co	4.60E+05	7.88E+05	1.37E-14	1.37E-14
4288048	^{93}Nb (n,n') ^{93m}Nb	1.10E+05	4.23E+05	6.52E-17	6.52E-17

Note:

(a) Measured activity decay corrected to July 1, 2017.

Table F-38 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 1 In-Vessel Capsules

Reaction	M/C				Avg. M/C	% Unc. (1σ)
	Capsule					
	U	X	W	V		
^{63}Cu (n,α) ^{60}Co	1.15	1.19	1.17	1.10	1.15	3.4%
^{54}Fe (n,p) ^{54}Mn	0.98	0.98	1.08	0.96	1.00	5.4%
^{58}Ni (n,p) ^{58}Co	0.88	0.94	1.08	-	0.97	10.6%
^{238}U(Cd) (n,f) ^{137}Cs	1.03	1.08	1.26	1.12	1.12	8.8%
^{237}Np(Cd) (n,f) ^{137}Cs	1.03	0.86	1.05	0.97	0.98	8.7%
Linear Average of M/C Results					1.05	10.1

Table F-39 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 1 Ex-Vessel Midplane Capsules

Reaction	M/C								Avg. M/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
^{63}Cu (n,α) ^{60}Co	0.88	0.95	0.94	0.92	0.88	0.95	0.93	0.92	0.92	3.0%
^{46}Ti (n,p) ^{46}Sc	0.85	1.02	0.97	0.91	0.84	0.93	0.93	0.91	0.92	6.4%
^{54}Fe (n,p) ^{54}Mn	0.90	0.96	0.89	0.95	0.92	0.97	0.97	0.96	0.94	3.4%
^{58}Ni (n,p) ^{58}Co	0.80	0.85	0.88	0.86	0.81	0.88	0.89	0.84	0.85	3.9%
^{93}Nb (n,n') ^{93m}Nb	1.00	0.94	0.99	1.06	1.03	1.03	1.02	1.08	1.02	4.2%
Linear Average of M/C Results									0.93	7.1%

Table F-40 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 1 Ex-Vessel Off-Midplane Capsules

Reaction	M/C			
	Capsule			
	D	F	J	L
^{63}Cu (n,α) ^{60}Co	0.64	0.97	0.80	0.69
^{46}Ti (n,p) ^{46}Sc	0.77	1.10	0.94	0.84
^{54}Fe (n,p) ^{54}Mn	0.69	1.09	0.86	0.82
^{58}Ni (n,p) ^{58}Co	0.75	1.06	0.96	0.80
^{93}Nb (n,n') ^{93m}Nb	0.83	1.06	1.09	0.81

Table F-41 Comparison of Calculated and Best-Estimate Exposure Rates for Braidwood Unit 1 In-Vessel Capsules

Reaction	BE/C				Avg. BE/C	% Unc. (1σ)
	Capsule					
	U	X	W	V		
Fast Neutron Fluence rate (E > 1.0 MeV)	0.95	0.94	1.10	1.00	1.00	7.3
dpa/s	0.97	0.94	1.10	1.00	1.00	6.9

Table F-42 Comparison of Calculated and Best-Estimate Exposure Rates for Braidwood Unit 1 Ex-Vessel Midplane Capsules

Reaction	BE/C								Avg. BE/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.92	0.91	0.93	0.97	0.94	0.97	0.97	0.98	0.95	2.8%
dpa/s	0.98	0.94	0.97	1.02	1.00	1.01	1.00	1.03	0.99	2.9%

Table F-43 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 2 In-Vessel Capsules

Reaction	M/C				Avg. M/C	% Unc. (1σ)
	Capsule					
	U	X	W	V		
^{63}Cu (n,α) ^{60}Co	1.14	1.15	1.16	1.13	1.15	1.1%
^{54}Fe (n,p) ^{54}Mn	0.97	0.94	1.04	0.99	0.99	4.3%
^{58}Ni (n,p) ^{58}Co	0.90	0.93	1.05	-	0.96	8.3%
^{238}U(Cd) (n,f) ^{137}Cs	1.00	1.05	1.05	1.07	1.04	2.9%
^{237}Np(Cd) (n,f) ^{137}Cs	1.05	0.87	1.03	1.00	0.99	8.2%
Linear Average of M/C Results					1.03	8.1%

Table F-44 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 2 Ex-Vessel Midplane Capsules

Reaction	M/C								Avg. M/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	0.93	0.98	0.98	0.93	0.89	0.93	0.94	0.91	0.94	3.3%
$^{46}Ti\ (n,p)\ ^{46}Sc$	0.90	0.94	0.93	0.89	0.89	0.92	0.96	0.92	0.92	2.7%
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.88	0.95	0.96	0.92	0.94	0.95	0.99	0.97	0.95	3.5%
$^{58}Ni\ (n,p)\ ^{58}Co$	0.84	0.84	0.87	0.84	0.89	0.91	0.86	0.93	0.87	4.0%
$^{93}Nb\ (n,n')\ ^{93m}Nb$	1.05	1.01	1.03	1.08	1.05	1.04	1.07	1.15	1.06	4.0%
Linear Average of M/C Results									0.95	7.4%

Table F-45 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Braidwood Unit 2 Ex-Vessel Off-Midplane Capsules

Reaction	M/C			
	Capsule			
	D	F	J	L
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	0.76	0.74	0.78	0.75
$^{46}Ti\ (n,p)\ ^{46}Sc$	0.87	0.81	0.98	0.88
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.77	0.80	0.85	0.88
$^{58}Ni\ (n,p)\ ^{58}Co$	0.82	0.81	0.92	0.84
$^{93}Nb\ (n,n')\ ^{93m}Nb$	0.97	0.79	1.03	0.90

Table F-46 Comparison of Calculated and Best-Estimate Exposure Rates for Braidwood Unit 2 In-Vessel Capsules

Reaction	BE/C				Avg. BE/C	% Unc. (1σ)
	Capsule					
	U	X	W	V		
Fast Neutron Fluence (E > 1.0 MeV)	0.95	0.92	1.02	1.01	0.98	4.9%
dpa/s	0.97	0.93	1.02	1.00	0.98	4.0%

Table F-47 Comparison of Calculated and Best-Estimate Exposure Rates for Braidwood Unit 2 Ex-Vessel Midplane Capsules

Reaction	BE/C								Avg. BE/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.95	0.94	0.96	0.96	0.99	0.98	0.99	1.04	0.98	3.2%
dpa/s	1.00	0.97	0.98	1.01	1.01	1.00	1.02	1.07	1.01	3.0%

APPENDIX G APPLICABILITY OF BAW-2308 TO BYRON AND BRAIDWOOD NOZZLE-TO-SHELL WELDS

G.1 INTRODUCTION

BAW-2308 [6] was developed to establish alternative unirradiated reference temperatures based on the "Master Curve" method for reactor vessel (RV) welds produced using Linde 80 flux. The report makes use of fracture toughness testing and American Society of Mechanical Engineers (ASME) Code Case N-629 [41] to develop heat-specific and generic Linde 80 reference temperatures. The alternatives to RT_{NDT} (termed RTT_0) along with corresponding margin terms (σ_I) developed in [6] were approved by the NRC, with the latest approved version of the report being Revision 2-A [6].

The focus of [6] is on Babcock & Wilcox Owners Group (B&WOG) RV working group beltline Linde 80 welds, as these welds were the materials expected to see a benefit from using the Master Curve method at that time since they were the limiting materials. However, since the development of [6], areas beyond the traditional beltline region of the RV termed the "extended beltline" must now also be analyzed consistent with Regulatory Issue Summary (RIS) 2014-11 [14]. Since the extended beltline was not considered in the development of [6], materials exist in reactor vessel extended beltline regions to which [6] applies that were not identified when [6] was completed.

One group of materials often comprising a part of the extended beltline is the primary inlet and outlet nozzle-to-shell welds. At Byron Units 1 and 2, as well as Braidwood Units 1 and 2, the nozzle-to-shell welds were determined to be Linde 80 welds. This attachment demonstrates that the work completed in [6] is applicable to certain Byron and Braidwood nozzle-to-shell welds.

G.2 BYRON AND BRAIDWOOD NOZZLE-TO-SHELL MATERIALS

Tables G-1 and Table G-2 summarize the material properties for the Byron and Braidwood nozzle-to-shell welds, respectively. Each of the welds in these tables is a Linde 80 flux type and Mn-Mo-Ni filler type weld fabricated using single wire automatic submerged arc welding. The filler wire qualification tests for each of these welds are dated from 1973 to 1976. All of the information in Tables G-1 and G-2 come directly from original B&W weld certification records.

Table G-1 Byron Units 1 and 2 Nozzle-to-Shell Weld Material Properties

Material	Heat #	Flux Lot	Cu wt. %	Ni wt. %	Ultimate Strength (ksi)	Yield Strength (ksi)
Byron Unit 1[(a)]						
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-419	1P5412	8969	0.178	0.69	92.5	76
Byron Unit 2						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-559	41403	8061	0.15	0.59	81.25	64.5
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-545	442010	8060	0.22	0.63	81	61.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-559	41403	8061	0.15	0.59	81.25	64.5

Note:

(a) Additional nozzle-to-shell weld materials exist for Byron Unit 1; however, use of the material properties from [6] for these additional materials is not considered in this Appendix.

Table G-2 Braidwood Units 1 and 2 Nozzle-to-Shell Weld Material Properties

Material	Heat #	Flux Lot	Cu wt. %	Ni wt. %	Ultimate Strength (ksi)	Yield Strength (ksi)
Braidwood Unit 1						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-598	41403	0852	0.29	0.56	87	67.75
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-598	41403	0852	0.29	0.56	87	67.75
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-588	41403	8119	0.29	0.63	82.5	65.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-579	442010	8119	0.25	0.63	85	68
Braidwood Unit 2						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-654	41404	0261	0.18	0.52	83.5	66.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-654	41404	0261	0.18	0.52	83.5	66.25

G.3 COMPARISON OF WELD MATERIALS

Comparison of the identifiers for the Byron and Braidwood nozzle-to-shell welds provides evidence that these welds fall into the same class. The identification numbers (WF-) in Tables G-1 and G-2 show that the naming convention utilized for the Byron and Braidwood nozzle-to-shell welds is consistent with those utilized for Linde 80 welds in [6] (see [6], Revision 1 Table 3-2). Additionally, some of the heat numbers in Tables G-1 and G-2 are identical to Heat numbers reported in [6], Revision 1-A Table 3-2, while the majority of those not specifically listed in [6], Revision 1-A are extremely similar to a Heat number reported in [6], Revision 1-A. Furthermore, the four-digit flux lot naming convention is also consistent with Linde 80 weld lot numbers for beltline welds. These naming conventions, in combination with the fact that these welds were fabricated by the same supplier (B&W) in approximately the same time period (1960's to 1970's) as the beltline Linde 80 welds discussed in [6] ensures that the results of [6] apply to the Byron and Braidwood nozzle-to-shell welds.

For Byron Unit 2 and Braidwood Units 1 and 2, as indicated by bare wire versus groove weld chemistries reported in the CMTRs, these weld wires were Cu coated when used for the nozzle-to-shell welds. It is noted, however, that the beltline welds of the Byron and Braidwood units did not use Cu coated weld wires. For the Byron Unit 1 weld in Table G-1, bare wire chemistry values are not available; however based on the Cu value, it is believed this weld was also copper coated. The copper content of these welds fall within or just outside of the range of copper values listed in Table 1-2 of [6], Revision 1 Additionally, the tensile properties (ultimate strength and yield strength) of the Byron and Braidwood nozzle-to-shell

welds fall well within the same range as the Linde 80 materials shown in Figures 3-1 and 3-2 of [6], Revision 1.

Table 3-3 of [6], Revision 1 contains a comparison of T_{NDT} values. For the majority of the welds listed in Tables G-1 and G-2, full drop-weight test results are unavailable. Only one weld record contains full test drop-weight results, and the corresponding T_{NDT} value was measured to be -60°F. This value is consistent with the T_{NDT} values in Table 3-2 of [6], Revision 1-A. The remaining welds contain drop-weight results at only one temperature with a result of two "no break" specimens. Thus, T_{NDT} cannot be determined precisely for the majority of the welds listed in Tables G-1 and G-2. Since less focus was placed on the extended beltline welds during fabrication of the RV, it is likely that documentation of full drop-weight testing was considered unnecessary at the time of fabrication.

Another common characteristic of Linde 80 welds are having low initial upper-shelf energy, compared to other reactor vessel welds. The Byron and Braidwood nozzle-to-shell welds also exhibit this characteristic, as the upper-shelf energy values for the majority of the Byron and Braidwood nozzle-to-shell welds are less than 80 ft-lbs, and the maximum USE value amongst these welds is only 85 ft-lbs. The upper shelf energy (USE) of the Linde 80 weld materials meet the 10 CFR 50, Appendix G requirements, i.e > 50 ft-lbs at end of license, as analyzed in WCAP-18092-NP, Rev. 1 (for Unit 1), and WCAP-18107-NP, Rev. 0 (for Unit 2).

It is noted that the nozzle-to-shell weld thickness is typically greater than the beltline weld thickness; however, as described in [42] Question 4, a difference in weld thickness over a range of 8 to 12 inches would not be expected to lead to systemic differences in material properties. Additionally, a large portion of the data utilized in [6] is based on nozzle dropout (ND) welds. The ND welds considered in [6] have a comparable thickness to the Byron and Braidwood nozzle-to-shell weld thickness. Since the σ term evaluation completed in [6] "accounts for any variation between the measured toughness of the source material (RVSP *[reactor vessel surveillance program]* block, ND or MD-1 *[Midland Unit 1]* beltline weld) and the actual vessel weld", applicable uncertainty is already considered in the [6] results.

Given the above discussion, it is concluded that the use of material properties from [6] is appropriate for the Byron and Braidwood nozzle-to-shell welds listed in Tables G-1 and G-2. Since the specific heats relevant to these welds were not analyzed in [6], the generic "all heats" RT_{T0} and uncertainty values will be utilized. It is noted that the generic values contain an added 20°F conservatism compared to the heat-specific values, as described in [43].

ENCLOSURE 2

WCAP-18371-NP, Revision 0

Byron Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

WCAP-18371-NP
Revision 0

September 2019

Byron Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WCAP-18371-NP
Revision 0

Byron Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation

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RECORD OF REVISION

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the Byron Units 1 and 2 reactor vessels. The P-T limit curves were generated using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G. This P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Note that the 4th ISI interval ASME Section XI Code of Record for Byron is the 2007 Edition through the 2008 Addenda. The requirements of Appendix G of the 2008 Addenda of the Code are equivalent to those from the 1998 Edition through 2000 Addenda edition used herein. The heatup and cooldown P-T limit curves utilize the Adjusted Reference Temperature (ART) values for Byron Units 1 and 2 calculated using Regulatory Guide 1.99, Revision 2. The limiting ART values in material with a postulated axial flaw were those of the Byron Unit 1 Intermediate Shell Forging 5P-5933 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations. The limiting ART values in material with a postulated circumferential flaw were those of the Byron Unit 2 Intermediate to Lower Shell Weld at both 1/4T and 3/4T locations. The axially oriented flaw cases are limiting; therefore, only the axially oriented flaw curves are presented in this report.

The P-T limit curves were generated for 57 effective full-power years (EFPY) using a heatup rate of 100°F/hr, and cooldown rates of 0° (steady-state), -25°, -50°, and -100°F/hr and are applicable to both Byron Unit 1 and Unit 2. The curves were developed without the flange requirements of 10 CFR 50, Appendix G, as justified by WCAP-16143-P [5] and approved in [25]. Reference [43] revised the exemption to account for Revision 1 of WCAP-16143-P, which considers a 53-stud configuration. The curves were also developed without margins for instrumentation errors. The curves can be found in Figures 8-1 and 8-2.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 57 EFPY.

Appendix B contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles to adhere to the requirements of Regulatory Issue Summary (RIS) 2014-11 [14]. As discussed in Appendix B, the P-T limit curves generated based on cylindrical beltline material (Byron Unit 1 Intermediate Shell Forging 5P-5933) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Byron Units 1 and 2 at 57 EFPY.

Appendix C contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix C, all the other ferritic RCPB components meet the applicable requirements of Section III of the ASME Code.

Appendix D contains the credibility evaluation of the Byron Units 1 and 2 reactor vessels surveillance data per the requirements of Regulatory Guide 1.99, Revision 2. Byron Units 1 and 2 fluence values and ex-vessel neutron dosimetry (EVND), described in Section 2.0, were used to complete the evaluation.

Appendix E contains a pressurized thermal shock (PTS) evaluation for all the Byron Units 1 and 2 reactor vessel beltline and extended beltline materials. Per Appendix E, all beltline and extended beltline materials have projected RT_{PTS} values below the screening criteria set forth in 10 CFR 50.61 at 57 EFPY.

Appendix F provides the validation of the radiation transport calculation models based on neutron dosimetry measurement.

Appendix G demonstrates the applicability of BAW-2308 results to the Byron Units 1 and 2 nozzle-to-shell welds as utilized herein.

1 INTRODUCTION

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) of the beltline region material of the reactor vessel. The adjusted RT_{NDT} 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 (T_{NDT}) 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} ($RT_{NDT(U)}$). 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 U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + 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 heatup and cooldown P-T limit curves documented in this report were generated using the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [2]. Specifically, the K_{Ic} methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, Appendix G [3] was used. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors.

The purpose of this report is to present the calculations and the development of the Byron Units 1 and 2 heatup and cooldown P-T limit curves for 57 EFPY. This report documents the calculated ART values and the development of the P-T limit curves for normal operation. The calculated ART values for 57 EFPY are documented in Section 7 of this report. The fluence projections used in calculation of the ART values are provided in Section 2 of this report.

The P-T limit curves herein were generated without instrumentation errors and are applicable to both Byron Unit 1 and Unit 2. The reactor vessel flange requirements of 10 CFR 50, Appendix G [4] have not been incorporated in the P-T limit curves. As part of the P-T limit curve development, the initial RT_{NDT} values for some of the Units 1 and 2 inlet and outlet nozzle to upper shell forging welds were redefined to take advantage of the "Master Curve" method. The use of the Master Curve is a departure from the ASME Code, Section III Subsection NB-2300 method required by 10 CFR 50, Appendix G; therefore, it requires the submittal and NRC approval of a 10 CFR 50.12 exemption for its use. Additional details about this method are defined in Section 3, and a justification for its use is contained in Appendix G. Per [5] and [25], the flange requirements have been eliminated from Byron Units 1 and 2. As discussed in Appendix B, the P-T limit curves generated in Section 8 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Byron Units 1 and 2 at 57 EFPY. Discussion of the other reactor coolant pressure boundary (RCPB) ferritic components relative to P-T limits is contained in Appendix C.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

Two discrete ordinates (S_N) transport analyses were performed for the Byron Units 1 and 2 reactors, respectively, to determine the neutron radiation environment within the reactor pressure vessels. In these analyses, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. The dosimetry analysis documented in WCAP-18334-NP [7] and WCAP-18367-NP [8] showed that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [9], is met, based on comparison with the four in-vessel surveillance capsules and twelve Ex-Vessel Neutron Dosimetry (EVND) capsules tested to-date for each unit from Byron Unit 1 and Unit 2. More details on compliance with Regulatory Guide 1.190 are contained in Appendix F. These validated calculations form the basis for providing projections of the neutron exposure of the reactor pressure vessel through the end of license extension (EOLE).

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2].

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the Byron Units 1 and 2 reactor vessels, a series of fuel-cycle-specific forward transport calculations were performed using the following two-dimensional/one-dimensional synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) \times \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron fluence rate distribution, $\phi(r,\theta)$ is the transport solution in r, θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r, θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at Byron Units 1 and 2, respectively.

All of the transport calculations were carried out using a two-dimensional discrete ordinates code, DORT [10], and the BUGLE-96 cross-section library [11]. The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions were treated on a fuel-cycle-specific basis.

The r,θ models used for this analysis are depicted in Figure 2-1, Figure 2-2, and Figure 2-3. In each of these figures, a single octant is depicted showing the arrangement of neutron pads and surveillance

capsules, as applicable. In regard to these three geometries, the maximum exposure of the pressure vessel occurs in octants with the 12.5-degree neutron pad span where no surveillance capsules are present. The surveillance capsules are modeled in the octants where the neutron pads span 20.0 degrees and 22.5 degrees, whereas EVND is located in an octant with the 12.5 degrees neutron pad span.

In addition to the core, reactor internals, pressure vessel, and primary biological shield, the transport models developed for these octant geometries included explicit representations of the surveillance capsules, the pressure vessel cladding, and the insulation located external to the pressure vessel. The reactor vessel insulation is modeled as a mixture of stainless steel and air, with majority of the volume being air. The reactor vessel insulation can be seen in Figure 2-1 and Figure 2-2 as the brown donut shape band outside of the reactor vessel wall, which is the blue color donut shape band.

From a neutronic standpoint, the inclusion of the surveillance capsules and associated support structure in the analytical model is significant. Because the presence of the capsules and structure has a marked effect on the magnitude of the neutron flux, as well as on the relative neutron and gamma-ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are properly accounted for in the analysis. The ex-core detector wells located in the concrete biological shielding of Byron Unit 1 and Unit 2 were also incorporated into the r-θ models, because experience has shown that these voids can significantly affect EVND dosimetry results. The detector wells do not affect the in-vessel surveillance capsule results.

In contrast to the relatively massive stainless steel and carbon steel structures associated with the in-vessel surveillance capsules, the small aluminum capsules used in the EVND program were designed to minimize perturbations in the neutron flux, and, thus, to provide essentially free-field data at the measurement locations. Therefore, specific modeling of these small capsules was not required.

The r, θ analytical models of the reactor geometry shown in Figure 2-1, Figure 2-2, and Figure 2-3 employed nominal design dimensions for the various structural components. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The r, θ geometric mesh description of the reactor model shown in Figure 2-1 consisted of 257 radial by 131 azimuthal intervals, whereas the reactor models shown in Figure 2-2 and Figure 2-3 consisted of 255 radial by 143 azimuthal intervals. Differences in the number of radial and azimuthal intervals between neutron pad configurations can be attributed to extra mesh refinement in the region of the surveillance capsules. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r, θ calculations was set at a value of 0.001.

A section view of the r, z model of the Byron Units 1 and 2 reactors is shown in Figure 2-4. The model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately six feet below the active fuel to approximately five feet above the active fuel. As in the case of the r, θ model, nominal design dimensions and full-power coolant densities were employed in the calculations. In this case, the homogenous core

region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The volume fractions utilized for the fuel region, the bypass region, the downcomer region, and the reactor pressure vessel (RPV) insulation region were consistent with the equivalent regions in the r, θ model.

The r, z geometric mesh description of the reactor model shown in Figure 2-4 consisted of 153 radial by 216 axial intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration convergence criterion utilized in the r, z calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 153 radial mesh intervals included in the r, z model. Thus, radial synthesis factors could easily be determined on a mesh-wise basis throughout the entire geometry.

The core power distributions used in the plant-specific transport analysis for the Byron Unit 1 and Unit 2 reactors were taken from nuclear design documentation. The data extracted included fuel assembly-specific initial enrichments, beginning-of-cycle burnups, and end-of-cycle burnups. Appropriate axial power distributions were also obtained.

For each fuel cycle of operation, the fuel assembly-specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope-dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the spatial mesh arrays used in the discrete ordinates calculations.

In Table 2-1, axial and azimuthal locations of the Byron Unit 1 and Unit 2 pressure vessel materials are provided. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the midplane of the active fuel stack.

Cycle-specific calculations were performed for Cycles 1 through 21 for Byron Unit 1 and for Cycles 1 through 20 for Unit 2, with core thermal powers given in Table 2-2 and Table 2-3, respectively. Note that fluence data for the most recently completed Unit 1 and Unit 2 cycles were not included because this information was not available when the fluence projections supporting the P-T limits herein were calculated.

Neutron fluence rate and fluence are given in Table 2-4, Table 2-5, and Table 2-8 for Byron Unit 1 and in Table 2-10, Table 2-11, and Table 2-14 for Byron Unit 2. Similarly, iron atom displacement rate and iron atom displacements are provided in Table 2-6, Table 2-7, and Table 2-9 for Byron Unit 1, and Table 2-12, Table 2-13, and Table 2-15 for Byron Unit 2. The data presented represent the maximum neutron exposure experienced by RPV materials that will constitute inputs to the reactor vessel integrity analysis. The reported data consider both the inner and outer radius of the RPV base metal, and account for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for

each operating cycle of the Byron Unit 1 and Unit 2 reactors. Note that, for any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

In Table 2-4 through Table 2-15, calculated exposure values are projected to 32, 48, 54, 57, and 60 EFPY. Projections were based on the average of Cycles 19, 20, and 21 spatial power distributions and reactor operating conditions for Unit 1 and the average of Cycles 18, 19, and 20 spatial power distributions and reactor operating conditions for Unit 2 with a rated core power of 3658 MWt for both units, accounting for calorimetric uncertainty. The projected results will remain valid as long as future plant operation is consistent with these assumptions.

Results of the discrete ordinates transport analyses pertinent to the Byron Unit 1 surveillance capsule evaluations are provided in Table 2-16 through Table 2-18. In Table 2-16, the calculated fast neutron fluence rate and fluence (E > 1.0 MeV) are provided at the geometric center of capsules and at core midplane, as a function of irradiation time for the Byron Unit 1 reactor. Similar data presented in terms of iron atom displacement rate (dpa/s) and integrated iron atom displacements (dpa) are given in Table 2-17.

In Table 2-18, lead factors associated with surveillance capsules are provided as a function of operating time for the Byron Unit 1 reactor. The lead factor is defined as the ratio of the neutron fluence (E > 1.0 MeV) at the geometric center of the surveillance capsule to the maximum neutron fluence (E > 1.0 MeV) at the pressure vessel clad/base metal interface.

All surveillance capsules have been removed from Byron Unit 1, so fluence data at the surveillance positions beyond Cycle 15 is unnecessary (because there are no capsules receiving any fluence). However, if any capsules were to be re-inserted, it would be necessary to know the fast fluence rate at the surveillance capsule holder positions. To allow determination of potential fast fluence accumulation, projected fast fluence rate (E > 1.0 MeV) at each surveillance capsule location is provided in Table 2-19. Projections of future operation are based on an average of Cycles 19, 20, and 21. The additional fast fluence accumulated for any re-inserted capsule can be determined by multiplying the fast fluence rate value in Table 2-19 for the appropriate capsule position times the irradiation duration in effective full-power seconds (EFPS).

Note that neutron fluence and iron atom displacement results are calculated by multiplying response functions which vary for each energy group with the neutron fluence intensities for that energy group. These response functions are based on the BUGLE-96 library which provides the neutron cross-section data sets.

Results of the discrete ordinates transport analyses pertinent to the Byron Unit 2 surveillance capsule evaluations are provided in Table 2-20 through Table 2-22. In Table 2-20, the calculated fast neutron fluence rate and fluence (E > 1.0 MeV) are provided at the geometric center of capsules and at core midplane, as a function of irradiation time for the Byron Unit 2 reactor. Similar data presented in terms of iron atom displacement rate (dpa/s) and integrated iron atom displacements (dpa) are given in Table 2-21.

In Table 2-22, lead factors associated with surveillance capsules are provided as a function of operating time for the Byron Unit 2 reactor. The lead factor is defined as the ratio of the neutron fluence

(E > 1.0 MeV) at the geometric center of the surveillance capsule to the maximum neutron fluence (E > 1.0 MeV) at the pressure vessel clad/base metal interface.

All surveillance capsules have been removed from Byron Unit 2, so fluence data at the surveillance positions beyond Cycle 15 is unnecessary (because there are no capsules receiving any fluence). However, if any capsules were to be re-inserted, it would be necessary to know the fast fluence rate at the surveillance capsule holder positions. To allow determination of potential fast fluence accumulation, projected fast fluence rate (E > 1.0 MeV) at each surveillance capsule location is provided in Table 2-23. Projections of future operation are based on an average of Cycles 18, 19, and 20. The additional fast fluence accumulated for any re-inserted capsule can be determined by multiplying the fast fluence rate value in Table 2-23 for the appropriate capsule position times the irradiation duration in EFPS.

2.3 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Byron Unit 1 and Unit 2 pressure vessels is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology used in the Byron Unit 1 and Unit 2 reactor pressure vessels neutron exposure evaluations was carried out in the following four stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator (NUREG/CR-6454, "Pool Critical Assembly Pressure Vessel Facility Benchmark" [12]) at the Oak Ridge National Laboratory (ORNL).

2. Comparison of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment (NUREG/CR-6453, "H.B. Robinson-2 Pressure Vessel Benchmark" [13]).

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments.

4. Comparison of the calculations with all available dosimetry results from measurement programs carried out at the Byron Units 1 and 2 reactors.

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross sections. This phase, however, did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational and geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations as well as to a lack of knowledge relative to various plant-specific parameters. The overall calculational uncertainty applicable to the Byron Units 1 and 2 analyses were established from the results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Byron Units 1 and 2 measurements) was used solely to demonstrate the adequacy of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used to bias the final results in any way.

Table 2-24 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in WCAP-14040-A. The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in WCAP-18334-NP and WCAP-18367-NP [7 and 8] support these uncertainty assessments for Byron Unit 1 and Unit 2.

Table 2-1 Byron Unit 1 and Unit 2 RPV Material Locations

Material	Axial Location(a) [cm]	Azimuthal Location(b) [degrees]
Outlet Nozzle Forging to Vessel Shell Welds – Lowest Extent(c)(d)		
Nozzle 1	268.22 to 272.03	22
Nozzle 2	268.22 to 272.03	158
Nozzle 3	268.22 to 272.03	202
Nozzle 4	268.22 to 272.03	338
Inlet Nozzle Forging to Vessel Shell Welds – Lowest Extent(c)(d)		
Nozzle 1	264.09 to 267.90	67
Nozzle 2	264.09 to 267.90	113
Nozzle 3	264.09 to 267.90	247
Nozzle 4	264.09 to 267.90	293
Nozzle Shell Forging – Lowest Extent	188.24	0 to 360
Nozzle Shell to Intermediate Shell Circumferential Weld(d)	186.34 to 188.24	0 to 360
Intermediate Shell Forging	-59.25 to 186.34	0 to 360
Intermediate Shell to Lower Shell Circumferential Weld(d)	-63.06 to -59.25	0 to 360
Lower Shell Forging	-307.69 to -63.06	0 to 360
Lower Shell to Lower Vessel Head Circumferential Weld(d)	-310.50 to -307.69	0 to 360

Notes:

(a) Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.

(b) Azimuthal locations are indexed to θ = 0.0 as shown on reactor vessel general assembly drawings.

(c) No credit is taken for the azimuthal location of the nozzle welds – the azimuthal angle providing the maximum exposure is reported.

(d) Values given are for the axial range of the weld with consideration for the weld thickness. Unit 1 results are reported at the weld centerline locations which are determined as the midpoint of the axial ranges provided.

Table 2-2 Reactor Core Power Level – Byron Unit 1

Cycle	Core Power [MWt]
1	3411.0
2	3411.0
3	3411.0
4	3411.0
5	3411.0
6	3411.0
7	3411.0
8	3411.0
9	3411.0
10	3411.0
11	3518.4
12	3586.6
13	3586.6
14	3586.6
15	3586.6
16	3586.6
17	3586.6
18	3586.6
19	3589.8[a]
20	3658.0[b]
21	3658.0[b]

Notes.

(a) With a mid-cycle uprate from 3586.6 MWt to 3645 MWt on February 10, 2014

(b) The uprate to 3645 MWt is modeled as 3658 MWt to account for calorimetric uncertainty

Table 2-3 Reactor Core Power Level – Byron Unit 2

Cycle	Core Power [MWt]
1	3411.0
2	3411.0
3	3411.0
4	3411.0
5	3411.0
6	3411.0
7	3411.0
8	3411.0
9	3411.0
10	3518.4
11	3586.6
12	3586.6
13	3586.6
14	3586.6
15	3586.6
16	3586.6
17	3586.6
18	3619.8[(a)]
19	3658.0[(b)]
20	3658.0[(b)]

Notes:

(a) With a mid-cycle uprate from 3586 6 MWt to 3645 MWt on February 1, 2014.

(b) The uprate to 3645 MWt is modeled as 3658 MWt to account for calorimetric uncertainty.

Table 2-4 Calculated Maximum Fast Neutron Fluence Rate (E > 1.0 MeV) at the Byron Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	1.35E+10	2.16E+10	2.45E+10	2.73E+10	2.73E+10	45°
2	1.04	2.21	1.25E+10	1.78E+10	1.68E+10	1.72E+10	1.94E+10	21°
3	1.06	3.27	1.05E+10	1.68E+10	1.70E+10	1.66E+10	1.95E+10	22°
4	1.27	4.54	1.05E+10	1.65E+10	1.84E+10	1.97E+10	1.97E+10	45°
5	1.13	5.67	9.85E+09	1.52E+10	1.72E+10	1.86E+10	1.86E+10	45°
6	1.28	6.95	1.00E+10	1.46E+10	1.74E+10	2.06E+10	2.06E+10	45°
7	1.14	8.09	8.84E+09	1.38E+10	1.58E+10	1.71E+10	1.71E+10	45°
8	1.19	9.27	9.27E+09	1.39E+10	1.53E+10	1.48E+10	1.62E+10	22°
9	1.03	10.30	9.06E+09	1.41E+10	1.43E+10	1.19E+10	1.62E+10	22°
10	1.39	11.68	9.18E+09	1.34E+10	1.63E+10	1.77E+10	1.77E+10	45°
11	1.41	13.10	8.79E+09	1.34E+10	1.54E+10	1.44E+10	1.60E+10	23°
12	1.49	14.59	9.31E+09	1.45E+10	1.57E+10	1.39E+10	1.69E+10	22°
13	1.37	15.96	9.09E+09	1.34E+10	1.43E+10	1.43E+10	1.50E+10	22°
14	1.46	17.41	9.57E+09	1.46E+10	1.57E+10	1.42E+10	1.69E+10	22°
15	1.40	18.81	9.85E+09	1.44E+10	1.56E+10	1.43E+10	1.66E+10	22°
16	1.41	20.22	1.00E+10	1.52E+10	1.62E+10	1.50E+10	1.74E+10	22°
17	1.42	21.64	1.07E+10	1.56E+10	1.67E+10	1.57E+10	1.79E+10	22°
18	1.33	22.97	1.04E+10	1.56E+10	1.65E+10	1.49E+10	1.79E+10	22°
19	1.41	24.39	9.97E+09	1.52E+10	1.66E+10	1.59E+10	1.76E+10	22°
20	1.43	25.82	1.05E+10	1.64E+10	1.78E+10	1.64E+10	1.93E+10	22°
21	1.38	27.20	1.01E+10	1.56E+10	1.68E+10	1.51E+10	1.82E+10	22°

Table 2-5 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Byron Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	5.00E+17	8.04E+17	9.10E+17	1.02E+18	1.02E+18	45°
2	1.04	2.21	8.95E+17	1.36E+18	1.44E+18	1.56E+18	1.57E+18	22°
3	1.06	3.27	1.25E+18	1.93E+18	2.01E+18	2.11E+18	2.22E+18	22°
4	1.27	4.54	1.67E+18	2.59E+18	2.75E+18	2.90E+18	3.01E+18	22°
5	1.13	5.67	2.02E+18	3.13E+18	3.35E+18	3.55E+18	3.65E+18	22°
6	1.28	6.95	2.42E+18	3.72E+18	4.06E+18	4.39E+18	4.39E+18	45°
7	1.14	8.09	2.74E+18	4.21E+18	4.62E+18	5.00E+18	5.00E+18	45°
8	1.19	9.27	3.08E+18	4.73E+18	5.19E+18	5.54E+18	5.54E+18	45°
9	1.03	10.30	3.37E+18	5.18E+18	5.65E+18	5.93E+18	6.05E+18	22°
10	1.39	11.68	3.73E+18	5.70E+18	6.28E+18	6.62E+18	6.67E+18	22°
11	1.41	13.10	4.10E+18	6.26E+18	6.93E+18	7.22E+18	7.34E+18	22°
12	1.49	14.59	4.53E+18	6.94E+18	7.66E+18	7.87E+18	8.13E+18	22°
13	1.37	15.96	4.92E+18	7.51E+18	8.28E+18	8.48E+18	8.78E+18	22°
14	1.46	17.41	5.35E+18	8.16E+18	8.97E+18	9.11E+18	9.53E+18	22°
15	1.40	18.81	5.78E+18	8.80E+18	9.66E+18	9.74E+18	1.03E+19	22°
16	1.41	20.22	6.23E+18	9.47E+18	1.04E+19	1.04E+19	1.10E+19	22°
17	1.42	21.64	6.70E+18	1.02E+19	1.11E+19	1.11E+19	1.18E+19	22°
18	1.33	22.97	7.14E+18	1.08E+19	1.18E+19	1.17E+19	1.26E+19	22°
19	1.41	24.39	7.58E+18	1.15E+19	1.25E+19	1.24E+19	1.34E+19	22°
20	1.43	25.82	8.02E+18	1.22E+19	1.33E+19	1.31E+19	1.42E+19	22°
21[(a)]	1.38	27.20	8.46E+18	1.29E+19	1.40E+19	1.38E+19	1.50E+19	22°
		32.00	9.97E+18	1.52E+19	1.65E+19	1.61E+19	1.77E+19	22°
		36.00	1.12E+19	1.71E+19	1.87E+19	1.80E+19	1.99E+19	22°
		40.00	1.25E+19	1.90E+19	2.08E+19	2.00E+19	2.22E+19	22°
		44.00	1.37E+19	2.10E+19	2.29E+19	2.19E+19	2.44E+19	22°
		48.00	1.50E+19	2.29E+19	2.50E+19	2.39E+19	2.67E+19	22°
		52.00	1.62E+19	2.48E+19	2.71E+19	2.58E+19	2.90E+19	22°
		54.00	1.69E+19	2.58E+19	2.82E+19	2.68E+19	3.01E+19	22°
		57.00	1.78E+19	2.73E+19	2.97E+19	2.82E+19	3.18E+19	22°
		60.00	1.88E+19	2.87E+19	3.13E+19	2.97E+19	3.35E+19	22°

Note:

(a) Values beyond EOC 21 are projected based on an average of the spatial core power distributions and associated plant operating characteristics of Cycle 19 through 21.

Table 2-6 Calculated Maximum Iron Atom Displacement Rate at the Byron Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	2.09E-11	3.33E-11	3.78E-11	4.33E-11	4.33E-11	45°
2	1.04	2.21	1.94E-11	2.74E-11	2.60E-11	2.72E-11	2.97E-11	21°
3	1.06	3.27	1.64E-11	2.59E-11	2.63E-11	2.62E-11	2.98E-11	22°
4	1.27	4.54	1.63E-11	2.55E-11	2.85E-11	3.12E-11	3.12E-11	45°
5	1.13	5.67	1.53E-11	2.34E-11	2.66E-11	2.95E-11	2.95E-11	45°
6	1.28	6.95	1.56E-11	2.26E-11	2.70E-11	3.26E-11	3.26E-11	45°
7	1.14	8.09	1.38E-11	2.13E-11	2.44E-11	2.70E-11	2.70E-11	45°
8	1.19	9.27	1.44E-11	2.14E-11	2.37E-11	2.35E-11	2.48E-11	22°
9	1.03	10.30	1.41E-11	2.17E-11	2.21E-11	1.88E-11	2.48E-11	22°
10	1.39	11.68	1.43E-11	2.06E-11	2.51E-11	2.80E-11	2.80E-11	45°
11	1.41	13.10	1.37E-11	2.07E-11	2.38E-11	2.28E-11	2.46E-11	26°
12	1.49	14.59	1.45E-11	2.23E-11	2.42E-11	2.20E-11	2.60E-11	22°
13	1.37	15.96	1.41E-11	2.06E-11	2.22E-11	2.27E-11	2.31E-11	22°
14	1.46	17.41	1.49E-11	2.25E-11	2.42E-11	2.26E-11	2.59E-11	22°
15	1.40	18.81	1.53E-11	2.22E-11	2.41E-11	2.26E-11	2.55E-11	22°
16	1.41	20.22	1.56E-11	2.34E-11	2.50E-11	2.38E-11	2.67E-11	22°
17	1.42	21.64	1.66E-11	2.40E-11	2.58E-11	2.49E-11	2.75E-11	22°
18	1.33	22.97	1.62E-11	2.40E-11	2.54E-11	2.37E-11	2.75E-11	22°
19	1.41	24.39	1.55E-11	2.34E-11	2.57E-11	2.52E-11	2.70E-11	22°
20	1.43	25.82	1.64E-11	2.52E-11	2.75E-11	2.59E-11	2.96E-11	22°
21	1.38	27.20	1.57E-11	2.40E-11	2.60E-11	2.39E-11	2.80E-11	22°

Table 2-7 Calculated Maximum Iron Atom Displacements at the Byron Unit 1 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacements (dpa)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.18	1.18	7.78E-04	1.24E-03	1.41E-03	1.61E-03	1.61E-03	45°
2	1.04	2.21	1.39E-03	2.10E-03	2.22E-03	2.46E-03	2.46E-03	45°
3	1.06	3.27	1.94E-03	2.96E-03	3.10E-03	3.34E-03	3.41E-03	22°
4	1.27	4.54	2.59E-03	3.98E-03	4.25E-03	4.59E-03	4.61E-03	22°
5	1.13	5.67	3.13E-03	4.81E-03	5.18E-03	5.62E-03	5.62E-03	45°
6	1.28	6.95	3.76E-03	5.72E-03	6.26E-03	6.94E-03	6.94E-03	45°
7	1.14	8.09	4.26E-03	6.49E-03	7.14E-03	7.91E-03	7.91E-03	45°
8	1.19	9.27	4.79E-03	7.28E-03	8.02E-03	8.77E-03	8.77E-03	45°
9	1.03	10.30	5.25E-03	7.98E-03	8.73E-03	9.38E-03	9.38E-03	45°
10	1.39	11.68	5.80E-03	8.78E-03	9.71E-03	1.05E-02	1.05E-02	45°
11	1.41	13.10	6.37E-03	9.64E-03	1.07E-02	1.14E-02	1.14E-02	45°
12	1.49	14.59	7.05E-03	1.07E-02	1.18E-02	1.25E-02	1.25E-02	22°
13	1.37	15.96	7.66E-03	1.16E-02	1.28E-02	1.34E-02	1.35E-02	22°
14	1.46	17.41	8.32E-03	1.26E-02	1.39E-02	1.44E-02	1.46E-02	22°
15	1.40	18.81	8.99E-03	1.35E-02	1.49E-02	1.54E-02	1.57E-02	22°
16	1.41	20.22	9.69E-03	1.46E-02	1.60E-02	1.65E-02	1.69E-02	22°
17	1.42	21.64	1.04E-02	1.56E-02	1.72E-02	1.76E-02	1.81E-02	22°
18	1.33	22.97	1.11E-02	1.66E-02	1.82E-02	1.86E-02	1.93E-02	22°
19	1.41	24.39	1.18E-02	1.77E-02	1.94E-02	1.97E-02	2.05E-02	22°
20	1.43	25.82	1.25E-02	1.87E-02	2.05E-02	2.08E-02	2.17E-02	22°
21[(a)]	1.38	27.20	1.32E-02	1.98E-02	2.17E-02	2.18E-02	2.29E-02	22°
		32.00	1.55E-02	2.34E-02	2.56E-02	2.55E-02	2.71E-02	22°
		36.00	1.75E-02	2.63E-02	2.88E-02	2.86E-02	3.06E-02	22°
		40.00	1.94E-02	2.93E-02	3.21E-02	3.17E-02	3.40E-02	22°
		44.00	2.14E-02	3.23E-02	3.53E-02	3.47E-02	3.75E-02	22°
		48.00	2.33E-02	3.53E-02	3.86E-02	3.78E-02	4.10E-02	22°
		52.00	2.53E-02	3.83E-02	4.19E-02	4.09E-02	4.45E-02	22°
		54.00	2.63E-02	3.98E-02	4.35E-02	4.25E-02	4.62E-02	22°
		57.00	2.77E-02	4.20E-02	4.59E-02	4.48E-02	4.88E-02	22°
		60.00	2.92E-02	4.43E-02	4.84E-02	4.71E-02	5.15E-02	22°

Note:

(a) Values beyond EOC 21 are projected based on an average of the spatial core power distributions and associated plant operating characteristics of Cycle 19 through 21.

Table 2-8 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Byron Unit 1 Pressure Vessel Welds and Shells

Material	Fast Neutron Fluence (n/cm²)			
	25.82 EFPY	**27.20 EFPY**	**32 EFPY**	**36 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	4.25E+16	4.50E+16	5.40E+16	6.15E+16
Inlet Nozzle Forging to Vessel Shell Welds	5.61E+16	5.94E+16	7.13E+16	8.12E+16
Nozzle Shell Forging[(a)]	4.95E+18	5.23E+18	6.24E+18	7.09E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	4.95E+18	5.23E+18	6.24E+18	7.09E+18
Intermediate Shell Forging	1.41E+19	1.49E+19	1.76E+19	1.99E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.37E+19	1.45E+19	1.71E+19	1.93E+19
Lower Shell Forging	1.42E+19	1.50E+19	1.77E+19	1.99E+19
Lower Shell to Lower Vessel Head Circumferential Weld	6.38E+15	6.74E+15	7.98E+15	9.01E+15

Material	Fast Neutron Fluence (n/cm²)			
	48 EFPY	**54 EFPY**	**57 EFPY**	**60 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	8.40E+16	9.52E+16	1.01E+17	1.06E+17
Inlet Nozzle Forging to Vessel Shell Welds	1.11E+17	1.26E+17	1.33E+17	1.41E+17
Nozzle Shell Forging[(a)]	9.62E+18	1.09E+19	1.15E+19	1.21E+19
Nozzle Shell to Intermediate Shell Circumferential Weld	9.62E+18	1.09E+19	1.15E+19	1.21E+19
Intermediate Shell Forging	2.67E+19	3.01E+19	3.19E+19	3.36E+19
Intermediate Shell to Lower Shell Circumferential Weld	2.58E+19	2.91E+19	3.07E+19	3.24E+19
Lower Shell Forging	2.67E+19	3.01E+19	3.18E+19	3.35E+19
Lower Shell to Lower Vessel Head Circumferential Weld	1.21E+16	1.37E+16	1.44E+16	1.52E+16

Note

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld.

Table 2-9 Calculated Maximum Iron Atom Displacements at the Byron Unit 1 Pressure Vessel Welds and Shells

Material	Displacements per atom (dpa)			
	25.82 EFPY	27.20 EFPY	32 EFPY	36 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.01E-04	1.07E-04	1.27E-04	1.44E-04
Inlet Nozzle Forging to Vessel Shell Welds	1.12E-04	1.19E-04	1.41E-04	1.60E-04
Nozzle Shell Forging[(a)]	7.58E-03	8.01E-03	9.56E-03	1.08E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	7.58E-03	8.01E-03	9.56E-03	1.08E-02
Intermediate Shell Forging	2.16E-02	2.28E-02	2.70E-02	3.05E-02
Intermediate Shell to Lower Shell Circumferential Weld	2.11E-02	2.23E-02	2.63E-02	2.97E-02
Lower Shell Forging	2.17E-02	2.29E-02	2.71E-02	3.06E-02
Lower Shell to Lower Vessel Head Circumferential Weld	3.96E-05	4.18E-05	4.94E-05	5.58E-05

Material	Displacements per atom (dpa)			
	48 EFPY	54 EFPY	57 EFPY	60 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.95E-04	2.20E-04	2.33E-04	2.45E-04
Inlet Nozzle Forging to Vessel Shell Welds	2.16E-04	2.44E-04	2.58E-04	2.73E-04
Nozzle Shell Forging[(a)]	1.47E-02	1.67E-02	1.76E-02	1.86E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	1.47E-02	1.67E-02	1.76E-02	1.86E-02
Intermediate Shell Forging	4.10E-02	4.62E-02	4.88E-02	5.15E-02
Intermediate Shell to Lower Shell Circumferential Weld	3.97E-02	4.48E-02	4.73E-02	4.98E-02
Lower Shell Forging	4.09E-02	4.61E-02	4.87E-02	5.13E-02
Lower Shell to Lower Vessel Head Circumferential Weld	7.48E-05	8.44E-05	8.91E-05	9.39E-05

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld.

Table 2-10 Calculated Maximum Fast Neutron Fluence Rate (E > 1.0 MeV) at the Byron Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.19	1.19	1.33E+10	2.14E+10	2.42E+10	2.69E+10	2.69E+10	45°
2	1.16	2.35	1.05E+10	1.57E+10	1.60E+10	1.67E+10	1.75E+10	21°
3	1.13	3.48	1.21E+10	1.61E+10	1.62E+10	1.63E+10	1.76E+10	21°
4	1.19	4.67	1.02E+10	1.61E+10	1.86E+10	2.06E+10	2.06E+10	45°
5	1.23	5.90	1.04E+10	1.60E+10	1.83E+10	2.00E+10	2.00E+10	45°
6	1.32	7.22	1.01E+10	1.54E+10	1.75E+10	1.91E+10	1.91E+10	45°
7	1.41	8.63	9.46E+09	1.49E+10	1.66E+10	1.70E+10	1.77E+10	22°
8	1.41	10.04	8.68E+09	1.39E+10	1.54E+10	1.43E+10	1.64E+10	22°
9	1.39	11.43	8.64E+09	1.32E+10	1.55E+10	1.66E+10	1.66E+10	45°
10	1.40	12.82	8.91E+09	1.24E+10	1.25E+10	1.23E+10	1.36E+10	21°
11	1.46	14.28	8.79E+09	1.33E+10	1.42E+10	1.27E+10	1.54E+10	22°
12	1.45	15.74	8.68E+09	1.25E+10	1.33E+10	1.32E+10	1.41E+10	22°
13	1.45	17.19	8.82E+09	1.36E+10	1.51E+10	1.41E+10	1.60E+10	22°
14	1.39	18.58	9.63E+09	1.40E+10	1.56E+10	1.48E+10	1.62E+10	23°
15	1.47	20.05	9.59E+09	1.54E+10	1.70E+10	1.56E+10	1.83E+10	22°
16	1.34	21.40	1.00E+10	1.47E+10	1.56E+10	1.46E+10	1.68E+10	22°
17	1.43	22.83	9.12E+09	1.46E+10	1.60E+10	1.48E+10	1.73E+10	22°
18	1.40	24.23	1.03E+10	1.49E+10	1.57E+10	1.50E+10	1.69E+10	22°
19	1.46	25.69	9.91E+09	1.46E+10	1.62E+10	1.56E+10	1.70E+10	23°
20	1.36	27.05	9.51E+09	1.37E+10	1.44E+10	1.43E+10	1.53E+10	22°

Table 2-11 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Byron Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.19	1.19	5.00E+17	8.05E+17	9.10E+17	1.01E+18	1.01E+18	45°
2	1.16	2.35	8.82E+17	1.38E+18	1.49E+18	1.62E+18	1.62E+18	45°
3	1.13	3.48	1.30E+18	1.94E+18	2.05E+18	2.18E+18	2.22E+18	22°
4	1.19	4.67	1.69E+18	2.54E+18	2.75E+18	2.96E+18	2.96E+18	45°
5	1.23	5.90	2.09E+18	3.16E+18	3.46E+18	3.72E+18	3.72E+18	45°
6	1.32	7.22	2.51E+18	3.79E+18	4.18E+18	4.52E+18	4.52E+18	45°
7	1.41	8.63	2.93E+18	4.46E+18	4.92E+18	5.28E+18	5.28E+18	45°
8	1.41	10.04	3.32E+18	5.07E+18	5.61E+18	5.91E+18	5.94E+18	22°
9	1.39	11.43	3.69E+18	5.65E+18	6.29E+18	6.64E+18	6.64E+18	45°
10	1.40	12.82	4.09E+18	6.20E+18	6.84E+18	7.18E+18	7.22E+18	22°
11	1.46	14.28	4.49E+18	6.81E+18	7.49E+18	7.76E+18	7.92E+18	22°
12	1.45	15.74	4.87E+18	7.35E+18	8.07E+18	8.34E+18	8.54E+18	22°
13	1.45	17.19	5.27E+18	7.97E+18	8.76E+18	8.98E+18	9.27E+18	22°
14	1.39	18.58	5.69E+18	8.59E+18	9.44E+18	9.63E+18	9.98E+18	22°
15	1.47	20.05	6.11E+18	9.26E+18	1.02E+19	1.03E+19	1.08E+19	22°
16	1.34	21.40	6.53E+18	9.88E+18	1.08E+19	1.09E+19	1.15E+19	22°
17	1.43	22.83	6.95E+18	1.05E+19	1.16E+19	1.16E+19	1.23E+19	22°
18	1.40	24.23	7.39E+18	1.12E+19	1.22E+19	1.22E+19	1.30E+19	22°
19	1.46	25.69	7.82E+18	1.18E+19	1.29E+19	1.29E+19	1.37E+19	22°
20[(a)]	1.36	27.05	8.20E+18	1.24E+19	1.35E+19	1.35E+19	1.43E+19	22°
		32.00	9.67E+18	1.45E+19	1.58E+19	1.57E+19	1.68E+19	22°
		36.00	1.09E+19	1.63E+19	1.77E+19	1.75E+19	1.88E+19	22°
		40.00	1.22E+19	1.81E+19	1.97E+19	1.94E+19	2.09E+19	22°
		48.00	1.46E+19	2.17E+19	2.35E+19	2.32E+19	2.50E+19	22°
		54.00	1.65E+19	2.44E+19	2.64E+19	2.60E+19	2.81E+19	22°
		57.00	1.74E+19	2.58E+19	2.79E+19	2.74E+19	2.96E+19	22°
		60.00	1.84E+19	2.71E+19	2.94E+19	2.88E+19	3.12E+19	22°

Note:

(a) Values beyond EOC 20 are projected based on an average of the spatial core power distributions and associated plant operating characteristics of Cycle 18 through 20.

Table 2-12 Calculated Maximum Iron Atom Displacement Rate at the Byron Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.19	1.19	2.07E-11	3.29E-11	3.73E-11	4.26E-11	4.26E-11	45°
2	1.16	2.35	1.64E-11	2.42E-11	2.47E-11	2.64E-11	2.69E-11	21°
3	1.13	3.48	1.87E-11	2.47E-11	2.50E-11	2.57E-11	2.70E-11	21°
4	1.19	4.67	1.59E-11	2.47E-11	2.88E-11	3.26E-11	3.26E-11	45°
5	1.23	5.90	1.62E-11	2.47E-11	2.83E-11	3.17E-11	3.17E-11	45°
6	1.32	7.22	1.57E-11	2.37E-11	2.70E-11	3.03E-11	3.03E-11	45°
7	1.41	8.63	1.47E-11	2.29E-11	2.56E-11	2.69E-11	2.71E-11	22°
8	1.41	10.04	1.35E-11	2.13E-11	2.38E-11	2.27E-11	2.51E-11	23°
9	1.39	11.43	1.35E-11	2.04E-11	2.40E-11	2.63E-11	2.63E-11	45°
10	1.40	12.82	1.39E-11	1.91E-11	1.93E-11	1.94E-11	2.08E-11	21°
11	1.46	14.28	1.37E-11	2.06E-11	2.20E-11	2.02E-11	2.37E-11	22°
12	1.45	15.74	1.35E-11	1.92E-11	2.06E-11	2.09E-11	2.16E-11	22°
13	1.45	17.19	1.37E-11	2.10E-11	2.33E-11	2.23E-11	2.45E-11	23°
14	1.39	18.58	1.50E-11	2.15E-11	2.41E-11	2.35E-11	2.49E-11	23°
15	1.47	20.05	1.49E-11	2.37E-11	2.62E-11	2.47E-11	2.81E-11	22°
16	1.34	21.40	1.56E-11	2.26E-11	2.40E-11	2.30E-11	2.57E-11	22°
17	1.43	22.83	1.42E-11	2.25E-11	2.46E-11	2.34E-11	2.65E-11	22°
18	1.40	24.23	1.60E-11	2.29E-11	2.43E-11	2.37E-11	2.59E-11	22°
19	1.46	25.69	1.54E-11	2.24E-11	2.50E-11	2.47E-11	2.61E-11	23°
20	1.36	27.05	1.48E-11	2.11E-11	2.23E-11	2.26E-11	2.34E-11	22°

Table 2-13 Calculated Maximum Iron Atom Displacements at the Byron Unit 2 Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacements (dpa)					Maximum Location
			0°	15°	30°	45°	Maximum	
1	1.19	1.19	7.77E-04	1.24E-03	1.40E-03	1.60E-03	1.60E-03	45°
2	1.16	2.35	1.37E-03	2.12E-03	2.30E-03	2.56E-03	2.56E-03	45°
3	1.13	3.48	2.02E-03	2.98E-03	3.17E-03	3.46E-03	3.46E-03	45°
4	1.19	4.67	2.62E-03	3.91E-03	4.25E-03	4.68E-03	4.68E-03	45°
5	1.23	5.90	3.25E-03	4.86E-03	5.34E-03	5.90E-03	5.90E-03	45°
6	1.32	7.22	3.90E-03	5.84E-03	6.46E-03	7.15E-03	7.15E-03	45°
7	1.41	8.63	4.56E-03	6.86E-03	7.60E-03	8.35E-03	8.35E-03	45°
8	1.41	10.04	5.16E-03	7.81E-03	8.66E-03	9.36E-03	9.36E-03	45°
9	1.39	11.43	5.75E-03	8.70E-03	9.71E-03	1.05E-02	1.05E-02	45°
10	1.40	12.82	6.36E-03	9.54E-03	1.06E-02	1.14E-02	1.14E-02	45°
11	1.46	14.28	6.98E-03	1.05E-02	1.16E-02	1.23E-02	1.23E-02	45°
12	1.45	15.74	7.57E-03	1.13E-02	1.25E-02	1.32E-02	1.32E-02	45°
13	1.45	17.19	8.20E-03	1.23E-02	1.35E-02	1.42E-02	1.42E-02	45°
14	1.39	18.58	8.85E-03	1.32E-02	1.46E-02	1.52E-02	1.53E-02	22°
15	1.47	20.05	9.50E-03	1.43E-02	1.57E-02	1.63E-02	1.65E-02	22°
16	1.34	21.40	1.02E-02	1.52E-02	1.67E-02	1.73E-02	1.76E-02	22°
17	1.43	22.83	1.08E-02	1.62E-02	1.79E-02	1.84E-02	1.88E-02	22°
18	1.40	24.23	1.15E-02	1.72E-02	1.89E-02	1.94E-02	1.99E-02	22°
19	1.46	25.69	1.22E-02	1.82E-02	2.00E-02	2.04E-02	2.11E-02	22°
20[(a)]	1.36	27.05	1.28E-02	1.90E-02	2.09E-02	2.13E-02	2.20E-02	22°
		32.00	1.50E-02	2.23E-02	2.44E-02	2.48E-02	2.57E-02	22°
		36.00	1.70E-02	2.51E-02	2.74E-02	2.78E-02	2.89E-02	22°
		40.00	1.89E-02	2.79E-02	3.04E-02	3.08E-02	3.20E-02	22°
		48.00	2.28E-02	3.34E-02	3.64E-02	3.67E-02	3.83E-02	22°
		54.00	2.57E-02	3.76E-02	4.09E-02	4.12E-02	4.31E-02	22°
		57.00	2.71E-02	3.97E-02	4.31E-02	4.34E-02	4.54E-02	22°
		60.00	2.86E-02	4.18E-02	4.54E-02	4.56E-02	4.78E-02	22°

Note:

(a) Values beyond EOC 20 are projected based on an average of the spatial core power distributions and associated plant operating characteristics of Cycle 18 through 20

Table 2-14 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Byron Unit 2 Pressure Vessel Welds and Shells

Material	Fast Neutron Fluence (n/cm^2)			
	27.05 EFPY	**32 EFPY**	**36 EFPY**	**40 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	4.24E+16	5.14E+16	5.87E+16	6.60E+16
Inlet Nozzle Forging to Vessel Shell Welds	5.65E+16	6.85E+16	7.81E+16	8.78E+16
Nozzle Shell Forging[a]	4.50E+18	5.42E+18	6.16E+18	6.91E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	4.71E+18	5.66E+18	6.44E+18	7.21E+18
Intermediate Shell Forging	1.43E+19	1.68E+19	1.88E+19	2.09E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.39E+19	1.63E+19	1.82E+19	2.01E+19
Lower Shell Forging	1.43E+19	1.68E+19	1.87E+19	2.07E+19
Lower Shell to Lower Vessel Head Circumferential Weld	6.57E+15	7.72E+15	8.66E+15	9.59E+15

Material	Fast Neutron Fluence (n/cm^2)			
	48 EFPY	**54 EFPY**	**57 EFPY**	**60 EFPY**
Outlet Nozzle Forging to Vessel Shell Welds	8.05E+16	9.14E+16	9.68E+16	1.02E+17
Inlet Nozzle Forging to Vessel Shell Welds	1.07E+17	1.22E+17	1.29E+17	1.36E+17
Nozzle Shell Forging[a]	8.40E+18	9.51E+18	1.01E+19	1.06E+19
Nozzle Shell to Intermediate Shell Circumferential Weld	8.76E+18	9.92E+18	1.05E+19	1.11E+19
Intermediate Shell Forging	2.50E+19	2.81E+19	2.96E+19	3.12E+19
Intermediate Shell to Lower Shell Circumferential Weld	2.39E+19	2.68E+19	2.82E+19	2.96E+19
Lower Shell Forging	2.46E+19	2.75E+19	2.90E+19	3.05E+19
Lower Shell to Lower Vessel Head Circumferential Weld	1.15E+16	1.29E+16	1.36E+16	1.42E+16

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld.

Table 2-15 Calculated Maximum Iron Atom Displacements at the Byron Unit 2 Pressure Vessel Welds and Shells

Material	Displacements (dpa)			
	27.05 EFPY	32 EFPY	36 EFPY	40 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.02E-04	1.22E-04	1.38E-04	1.54E-04
Inlet Nozzle Forging to Vessel Shell Welds	1.13E-04	1.36E-04	1.54E-04	1.72E-04
Nozzle Shell Forging[(a)]	6.90E-03	8.31E-03	9.45E-03	1.06E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	7.22E-03	8.69E-03	9.88E-03	1.11E-02
Intermediate Shell Forging	2.19E-02	2.57E-02	2.89E-02	3.20E-02
Intermediate Shell to Lower Shell Circumferential Weld	2.14E-02	2.51E-02	2.80E-02	3.09E-02
Lower Shell Forging	2.20E-02	2.57E-02	2.87E-02	3.17E-02
Lower Shell to Lower Vessel Head Circumferential Weld	4.10E-05	4.80E-05	5.38E-05	5.96E-05

Material	Displacements (dpa)			
	48 EFPY	54 EFPY	57 EFPY	60 EFPY
Outlet Nozzle Forging to Vessel Shell Welds	1.86E-04	2.11E-04	2.23E-04	2.35E-04
Inlet Nozzle Forging to Vessel Shell Welds	2.08E-04	2.35E-04	2.48E-04	2.61E-04
Nozzle Shell Forging[(a)]	1.29E-02	1.46E-02	1.54E-02	1.63E-02
Nozzle Shell to Intermediate Shell Circumferential Weld	1.34E-02	1.52E-02	1.61E-02	1.70E-02
Intermediate Shell Forging	3.83E-02	4.31E-02	4.54E-02	4.78E-02
Intermediate Shell to Lower Shell Circumferential Weld	3.68E-02	4.12E-02	4.34E-02	4.56E-02
Lower Shell Forging	3.77E-02	4.22E-02	4.45E-02	4.67E-02
Lower Shell to Lower Vessel Head Circumferential Weld	7.11E-05	7.98E-05	8.41E-05	8.84E-05

Note:

(a) The maximum exposure on the nozzle shell forging is taken to be equal to the maximum exposure on the nozzle shell to intermediate shell circumferential weld

Table 2-16 Calculated Fast Neutron Fluence Rate and Fluence (E > 1.0 MeV) at Byron Unit 1 Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)			Fluence (n/cm^2)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	1.01E+11	1.10E+11	1.09E+11	3.77E+18	4.09E+18	4.04E+18
2	1.04	2.21	6.53E+10	6.90E+10	6.79E+10	5.90E+18	6.35E+18	6.26E+18
3	1.06	3.27	7.30E+10	7.56E+10	7.45E+10	8.35E+18	8.88E+18	8.75E+18
4	1.27	4.54	7.70E+10	8.31E+10	8.19E+10	1.14E+19	1.22E+19	1.20E+19
5	1.13	5.67	6.96E+10	7.48E+10	7.38E+10	1.39E+19	1.49E+19	1.47E+19
6	1.28	6.95	6.88E+10	---	7.63E+10	1.67E+19	----	1.77E+19
7	1.14	8.09	6.38E+10	---	6.81E+10	1.90E+19	----	2.02E+19
8	1.19	9.27	6.24E+10	---	6.55E+10	2.13E+19	---	2.26E+19
9	1.03	10.30	5.80E+10	---	5.83E+10	2.32E+19	---	2.45E+19
10	1.39	11.68	6.08E+10	---	6.59E+10	2.58E+19	----	2.74E+19
11	1.41	13.10	6.33E+10	---	6.61E+10	2.87E+19	---	3.03E+19
12	1.49	14.59	6.27E+10	---	6.44E+10	3.16E+19	---	3.34E+19
13	1.37	15.96	5.62E+10	---	---	3.40E+19	---	---
14	1.46	17.41	6.31E+10	---	---	3.69E+19	----	---
15	1.40	18.81	6.25E+10	---	---	3.97E+19	----	---

Note. Dual 29° applies to surveillance Capsules V and Y. Dual 31 5° applies to surveillance Capsules U and X and Single 31 5° applies to surveillance Capsules W and Z

Table 2-17 Calculated Iron Atom Displacement Rate and Iron Atom Displacements at Byron Unit 1 Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)			Displacements (dpa)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	2.00E-10	2.17E-10	2.14E-10	7.45E-03	8.08E-03	7.96E-03
2	1.04	2.21	1.28E-10	1.35E-10	1.33E-10	1.16E-02	1.25E-02	1.23E-02
3	1.06	3.27	1.43E-10	1.48E-10	1.45E-10	1.64E-02	1.74E-02	1.72E-02
4	1.27	4.54	1.51E-10	1.63E-10	1.60E-10	2.25E-02	2.40E-02	2.36E-02
5	1.13	5.67	1.36E-10	1.47E-10	1.44E-10	2.73E-02	2.92E-02	2.87E-02
6	1.28	6.95	1.35E-10	---	1.49E-10	3.28E-02	---	3.47E-02
7	1.14	8.09	1.25E-10	---	1.33E-10	3.72E-02	---	3.95E-02
8	1.19	9.27	1.22E-10	---	1.28E-10	4.18E-02	---	4.43E-02
9	1.03	10.30	1.13E-10	---	1.13E-10	4.55E-02	---	4.80E-02
10	1.39	11.68	1.19E-10	---	1.29E-10	5.07E-02	---	5.36E-02
11	1.41	13.10	1.23E-10	---	1.29E-10	5.62E-02	---	5.93E-02
12	1.49	14.59	1.22E-10	---	1.25E-10	6.19E-02	---	6.52E-02
13	1.37	15.96	1.10E-10	---	---	6.67E-02	---	---
14	1.46	17.41	1.23E-10	---	---	7.23E-02	---	---
15	1.40	18.81	1.22E-10	---	---	7.77E-02	---	---

Note: Dual 29° applies to surveillance Capsules V and Y Dual 31.5° applies to surveillance Capsules U and X and Single 31.5° applies to surveillance Capsules W and Z.

Table 2-18 Calculated Byron Unit 1 Surveillance Capsule Lead Factors

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Lead Factor		
			Dual 29°	Dual 31.5°	Single 31.5°
1	1.18	1.18	3.71	4.03(a)	3.97
2	1.04	2.21	3.75	4.04	3.98
3	1.06	3.27	3.75	3.99	3.93
4	1.27	4.54	3.80	4.06	4.00
5	1.13	5.67	3.81	4.08(b)	4.02
6	1.28	6.95	3.80	---	4.04
7	1.14	8.09	3.80	---	4.04
8	1.19	9.27	3.84	---	4.08(c)
9	1.03	10.30	3.83	---	4.05
10	1.39	11.68	3.87	---	4.11
11	1.41	13.10	3.91	---	4.13
12	1.49	14.59	3.89(d)	---	4.11(e)
13	1.37	15.96	3.88	---	---
14	1.46	17.41	3.88	---	---
15(g)	1.40	18.81	3.87(f)	---	---

Notes:

(a) Applicable to Surveillance Capsule U (withdrawn at EOC 1).
(b) Applicable to Surveillance Capsule X (withdrawn at EOC 5)
(c) Applicable to Surveillance Capsule W (withdrawn at EOC 8).
(d) Applicable to Surveillance Capsule V (withdrawn at EOC 12).
(e) Applicable to Surveillance Capsule Z (withdrawn at EOC 12).
(f) Applicable to Surveillance Capsule Y (withdrawn at EOC 15).
(g) Lead factors were not calculated beyond Cycle 15, as all surveillance capsules have been removed.

Table 2-19 Projected Fast Neutron Fluence Rate (E > 1.0 MeV) at Byron Unit 1 Surveillance Capsule Positions (Future Operation)

Capsule Position	Fluence Rate (n/cm^2-s)
Dual 29°	6.80E+10
Dual 31.5°	7.15E+10
Single 31.5°	7.04E+10

Table 2-20 Calculated Fast Neutron Fluence Rate and Fluence (E > 1.0 MeV) at Byron Unit 2 Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)			Fluence (n/cm^2)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.19	1.19	9.95E+10	1.08E+11	1.06E+11	3.74E+18	4.06E+18	4.00E+18
2	1.16	2.35	6.51E+10	6.99E+10	6.89E+10	6.13E+18	6.62E+18	6.53E+18
3	1.13	3.48	6.43E+10	6.87E+10	6.77E+10	8.42E+18	9.07E+18	8.94E+18
4	1.19	4.67	7.70E+10	8.39E+10	8.28E+10	1.13E+19	1.22E+19	1.21E+19
5	1.23	5.90	7.46E+10	8.10E+10	7.99E+10	1.42E+19	1.54E+19	1.51E+19
6	1.32	7.22	7.11E+10	7.72E+10	7.61E+10	1.72E+19	1.86E+19	1.83E+19
7	1.41	8.63	6.84E+10	7.29E+10	7.18E+10	2.02E+19	2.18E+19	2.15E+19
8	1.41	10.04	6.19E+10	---	6.42E+10	2.30E+19	---	2.44E+19
9	1.39	11.43	6.28E+10	---	6.80E+10	2.57E+19	---	2.73E+19
10	1.40	12.82	5.12E+10	---	5.40E+10	2.80E+19	---	2.97E+19
11	1.46	14.28	5.89E+10	---	6.05E+10	3.07E+19	---	3.25E+19
12	1.45	15.74	5.37E+10	---	---	3.31E+19	---	---
13	1.45	17.19	6.24E+10	---	---	3.60E+19	---	---
14	1.39	18.58	6.26E+10	---	---	3.88E+19	---	---
15	1.47	20.05	6.88E+10	---	---	4.19E+19	---	---

Note: Dual 29° applies to surveillance Capsules V and Y Dual 31.5° applies to surveillance Capsules U and X and Single 31.5° applies to surveillance Capsules W and Z

Table 2-21 Calculated Iron Atom Displacement Rate and Iron Atom Displacements at Byron Unit 2 Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Displacement Rate (dpa/s)			Displacements (dpa)		
			Dual 29°	Dual 31.5°	Single 31.5°	Dual 29°	Dual 31.5°	Single 31.5°
1	1.19	1.19	1.97E-10	2.13E-10	2.10E-10	7.40E-03	8.02E-03	7.90E-03
2	1.16	2.35	1.27E-10	1.36E-10	1.34E-10	1.21E-02	1.30E-02	1.28E-02
3	1.13	3.48	1.26E-10	1.34E-10	1.32E-10	1.65E-02	1.78E-02	1.75E-02
4	1.19	4.67	1.51E-10	1.65E-10	1.62E-10	2.22E-02	2.40E-02	2.36E-02
5	1.23	5.90	1.46E-10	1.59E-10	1.56E-10	2.79E-02	3.01E-02	2.97E-02
6	1.32	7.22	1.39E-10	1.51E-10	1.49E-10	3.37E-02	3.64E-02	3.59E-02
7	1.41	8.63	1.34E-10	1.43E-10	1.40E-10	3.96E-02	4.28E-02	4.21E-02
8	1.41	10.04	1.21E-10	---	1.25E-10	4.50E-02	---	4.77E-02
9	1.39	11.43	1.23E-10	---	1.33E-10	5.04E-02	---	5.35E-02
10	1.40	12.82	9.99E-11	---	1.05E-10	5.48E-02	---	5.81E-02
11	1.46	14.28	1.15E-10	---	1.18E-10	6.01E-02	---	6.35E-02
12	1.45	15.74	1.05E-10	---	---	6.49E-02	---	---
13	1.45	17.19	1.22E-10	---	---	7.05E-02	---	---
14	1.39	18.58	1.22E-10	---	---	7.58E-02	---	---
15	1.47	20.05	1.34E-10	---	---	8.21E-02	---	---

Note Dual 29° applies to surveillance Capsules V and Y. Dual 31.5° applies to surveillance Capsules U and X and Single 31.5° applies to surveillance Capsules W and Z

Table 2-22 Calculated Byron Unit 2 Surveillance Capsule Lead Factors

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Lead Factor		
			Dual 29°	Dual 31.5°	Single 31.5°
1	1.19	1.19	3.71	4.02[a]	3.96
2	1.16	2.35	3.79	4.10	4.04
3	1.13	3.48	3.80	4.09	4.04
4	1.19	4.67	3.83	4.14	4.08[b]
5	1.23	5.90	3.81	4.12	4.07
6	1.32	7.22	3.80	4.11	4.05
7	1.41	8.63	3.83	4.13[c]	4.08
8	1.41	10.04	3.86	----	4.10
9	1.39	11.43	3.87	----	4.12
10	1.40	12.82	3.87	----	4.12
11	1.46	14.28	3.87[d]	----	4.10[e]
12	1.45	15.74	3.88	----	----
13	1.45	17.19	3.89	----	----
14	1.39	18.58	3.88	----	----
15[g]	1.47	20.05	3.89[f]	----	----

Notes

(a) Applicable to Surveillance Capsule U (withdrawn at EOC 1).
(b) Applicable to Surveillance Capsule W (withdrawn at EOC 4)
(c) Applicable to Surveillance Capsule X (withdrawn at EOC 7)
(d) Applicable to Surveillance Capsule V (withdrawn at EOC 11).
(e) Applicable to Surveillance Capsule Z (withdrawn at EOC 11).
(f) Applicable to Surveillance Capsule Y (withdrawn at EOC 15).
(g) Lead factors were not calculated beyond Cycle 15, as all surveillance capsules have been removed.

Table 2-23 Projected Fast Neutron Fluence Rate (E > 1.0 MeV) at Byron Unit 2 Surveillance Capsule Positions (Future Operation)

Capsule Position	Fluence Rate (n/cm^2-s)
Dual 29°	6.18E+10
Dual 31.5°	6.60E+10
Single 31.5°	6.51E+10

Table 2-24 Calculational Uncertainties

Description	Uncertainty	
	Capsule	**Vessel Inner Radius**
PCA Comparisons	3%	3%
H. B. Robinson Comparisons	3%	3%
Analytical Sensitivity Studies	10%	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%	5%
Net Calculational Uncertainty	12%	13%

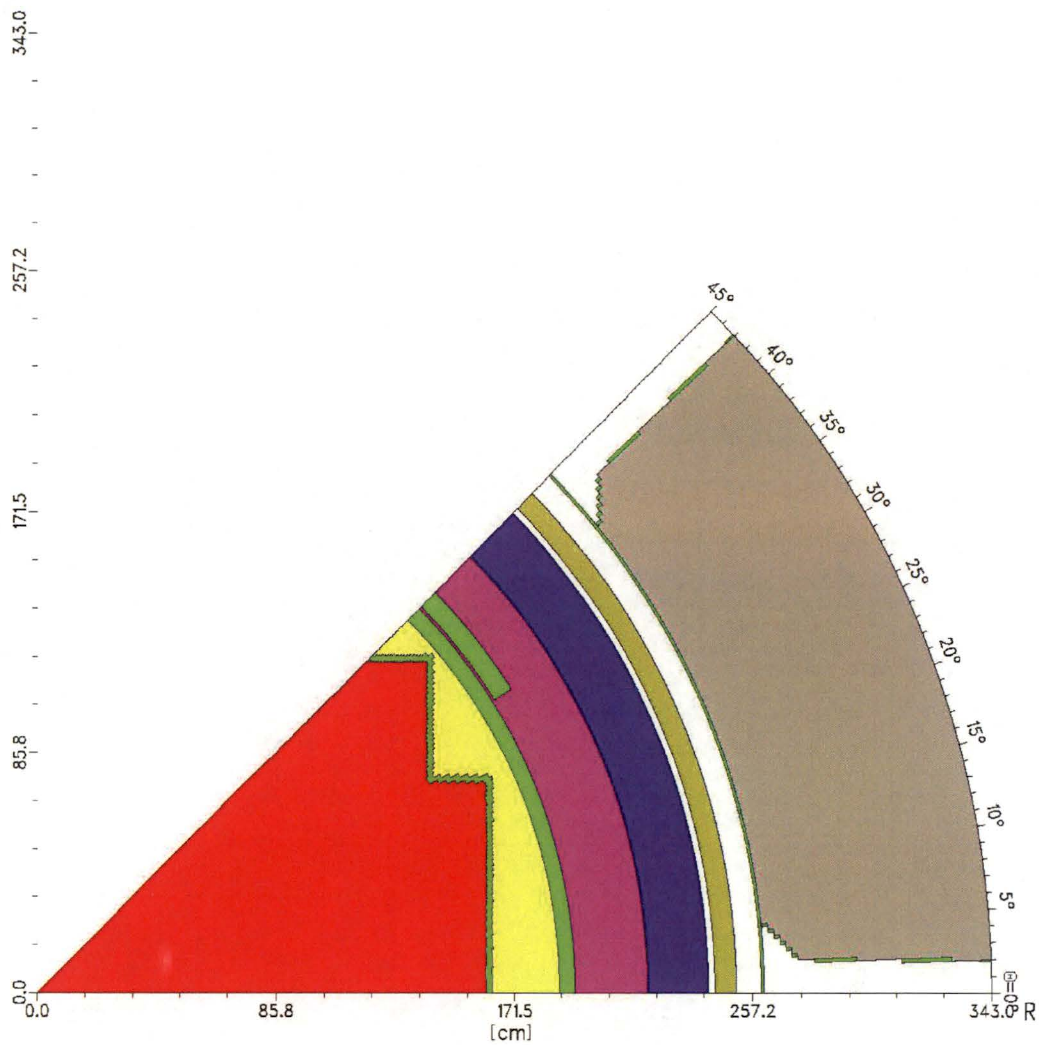

Figure 2-1 Byron Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 12.5-Degree Neutron Pad Configuration

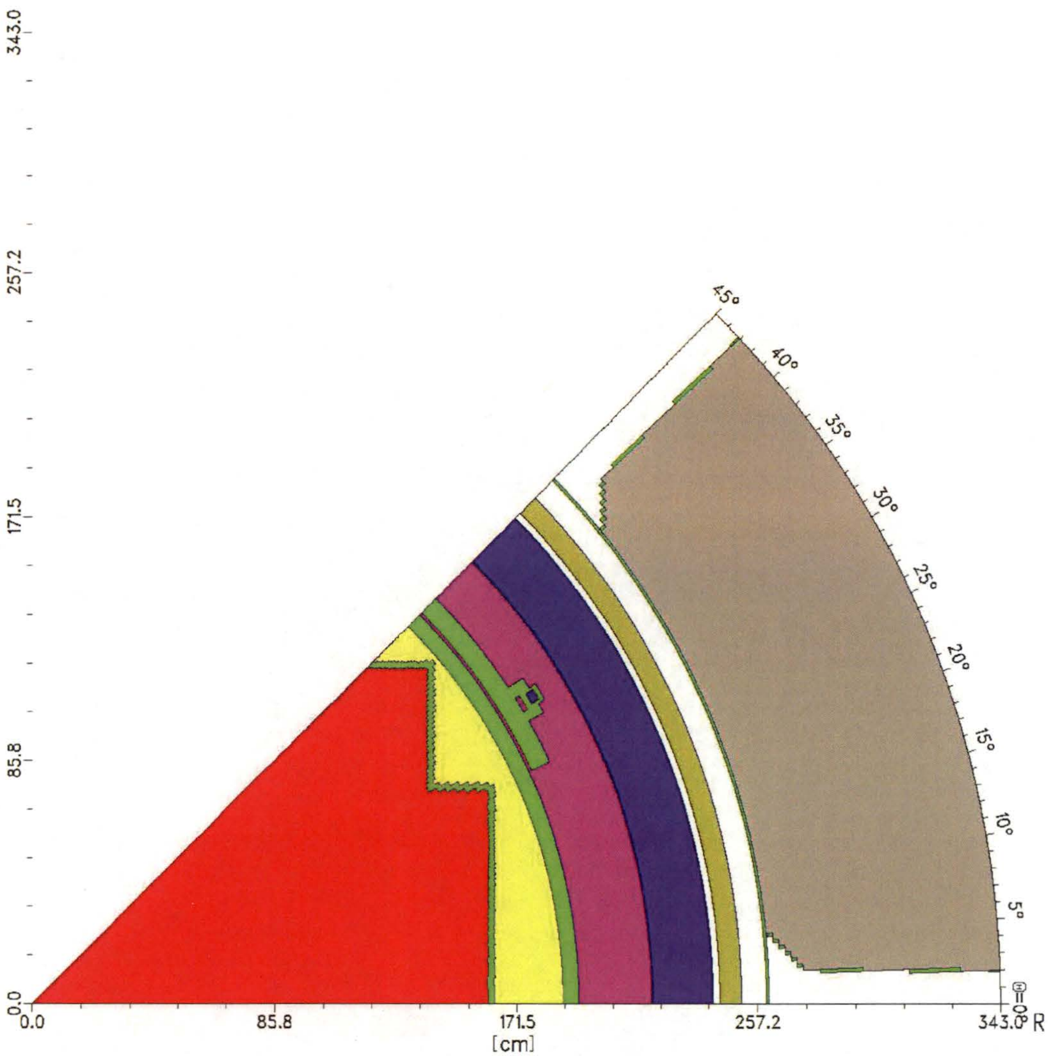

Figure 2-2 Byron Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 20.0-Degree Neutron Pad Configuration

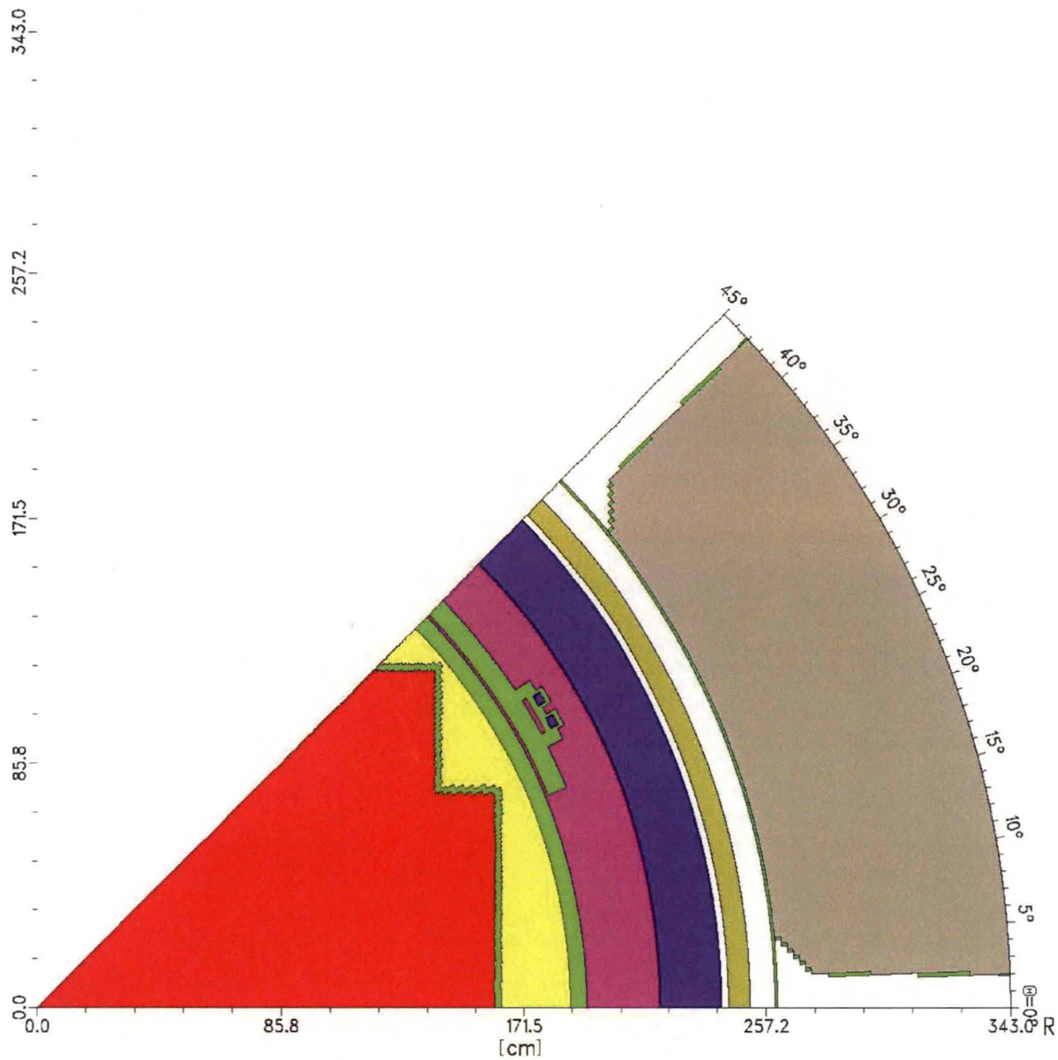

Figure 2-3 Byron Units 1 and 2 r, θ Reactor Geometry at the Core Midplane 22.5-Degree Neutron Pad Configuration

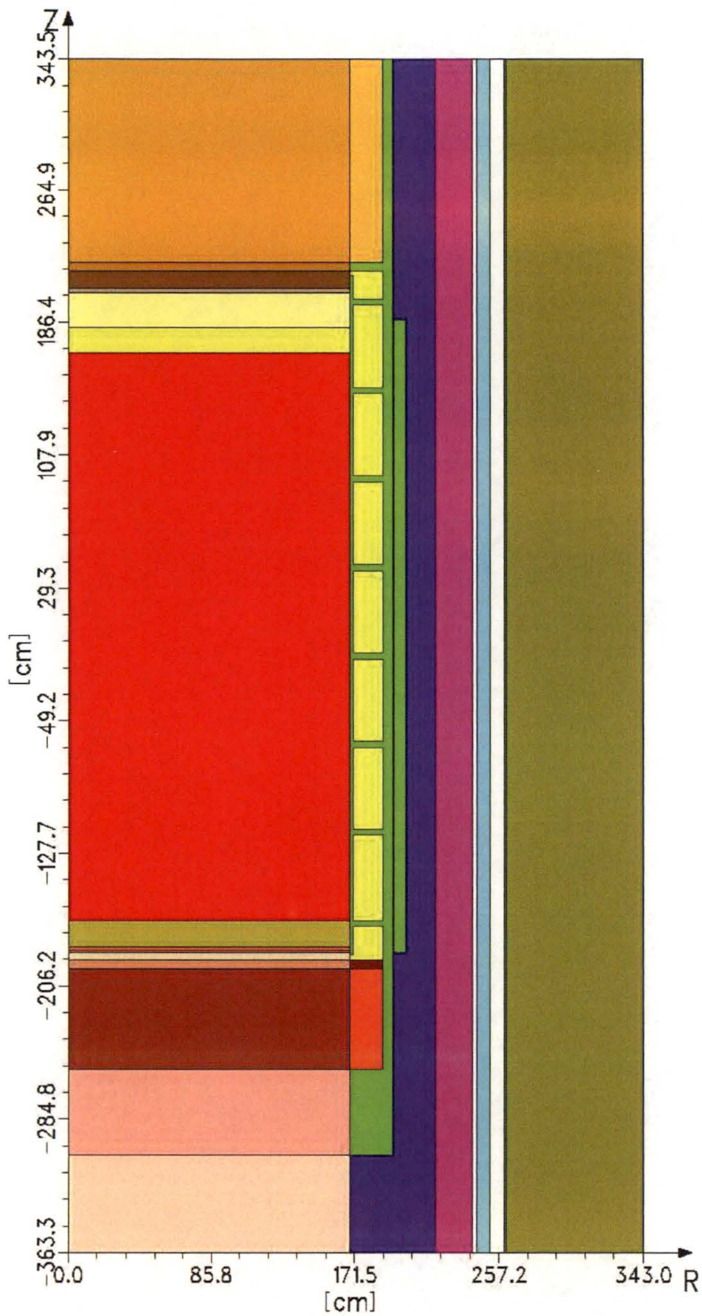

Figure 2-4 Byron Units 1 and 2 r, z Reactor Geometry

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [4]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *"the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage "*

The Byron Units 1 and 2 beltline materials traditionally included the Nozzle Shell Forgings, Intermediate Shell Forgings, Lower Shell Forgings, Nozzle to Intermediate Shell Circumferential Welds, and Intermediate to Lower Shell Circumferential Welds; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [14], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. As seen from Table 2-8 and Table 2-14 of this report, the fluence for both units' inlet nozzle to nozzle shell welds and the Unit 1 outlet nozzle to nozzle shell welds are greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 57 EFPY. Thus, these materials are a part of the extended beltline. The Byron Unit 2 outlet nozzle to nozzle shell welds have a projected fluence of less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 57 EFPY. Although the Unit 2 outlet nozzle to nozzle shell forging welds have projected fluence values less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period, these materials are evaluated with the extended beltline materials herein. Since the fluence values for both the inlet and outlet nozzle to nozzle shell forging welds are conservatively applied to the inlet and outlet nozzle forgings, both the inlet and outlet nozzle forgings are also considered a part of the extended beltline. Per NRC RIS 2014-11, the nozzle forging materials must be evaluated for their potential effect on P-T limit curves regardless of exposure. See Appendix B for more details.

A summary of the best-estimate copper (Cu) and nickel (Ni) contents, in units of weight percent (wt. %), as well as the initial RT_{NDT} values for the reactor vessel beltline and extended beltline materials are provided in Table 3-1 and Table 3-2 for Byron Units 1 and 2, respectively. Table 3-3 provides the initial RT_{NDT} values for the reactor vessel closure head and vessel flange materials for Byron Units 1 and 2. These values are taken from References [44] and [45].

"Master Curve" Fracture Toughness Properties

As part of this P-T limit curve development effort, the initial RT_{NDT} for some of the Units 1 and 2 inlet/outlet nozzle to shell forging welds were redefined in order to take advantage of the "Master Curve" method ($RT_{NDT} = T_0 + 35°F$) in BAW-2308 Revision 1-A Safety Evaluation (SE) and Revision 2-A SE [6]. When using these Master Curve-generated initial RT_{NDT} values, the chemistry factor (CF) and σ_Δ terms will be adjusted to a minimum of 167°F and 28°F, respectively; however, if the material-specific CF value is greater than 167°F, the material-specific value will be used. The values of 167°F and 28°F comply with the "Conditions and Limitation" placed on the use of "Master Curve" fracture toughness properties in Section 5.0 of Revision 1-A SE of [6].

The use of the Master Curve is a departure from the ASME Code, Section III, Subsection NB-2300 method required by 10 CFR 50, Appendix G; therefore, it requires the submittal and NRC approval of a 10 CFR 50.12 exemption for its use in P-T limit curve development. All of the technical requirements outlined in the SEs for [6] have been met and justification for the use of the Master Curve method in [6] at Byron Units 1 and 2 is provided in Appendix G.

Table 3-1 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Byron Unit 1 Reactor Vessel Materials(a)

Reactor Vessel Material	Heat Number	Wt. % Cu(b)	Wt. % Ni(b)	$RT_{NDT(u)}$ (c) (°F)
Reactor Vessel Beltline Materials				
Nozzle Shell Forging	123J218	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
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501	442011 (Linde 80 flux type, Lot # 8086)	0.03	0.67	10
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336	442002 (Linde 80 flux type, Lot # 8873)	0.04	0.63	-30
Surveillance Weld Material – Byron Unit 1	442002 (Linde 80 flux type, Lot # 8873	0.02	0.69	---
Reactor Vessel Extended Beltline Materials				
Inlet Nozzle 03-001	1V4684-3V1320(e)	0.12	0.82	-10
Inlet Nozzle 03-002	1V4684-3V1320	0.12	0.82	-20
Inlet Nozzle 04-001	1V4695	0.13	0.79	-20
Inlet Nozzle 04-002	1V4695	0.12	0.78	-20
Outlet Nozzle 01-001	1V4656	0.11	0.84	0
Outlet Nozzle 01-002	1V4656	0.11	0.84	-20
Outlet Nozzle 02-001	2V2557	0.11	0.85	-20
Outlet Nozzle 02-002	2V2557	0.11	0.84	-10
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337	442002 (Linde 80 flux type, Lot # 8873)(f)	0.15	0.56	-10
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419	1P5412 (Linde 80 flux type, Lot # 8969)	0.178	0.69	-48.6(d)
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406	504 (Linde 80 flux type, Lot # 8968)	0.054	0.80	10

Notes:

(a) All data taken from Section 3 of [16], unless otherwise noted

(b) The chemical compositions are based on Byron Unit 1 CMTR data

(c) The initial RT_{NDT} values are based on measured data for all beltline and extended beltline materials Initial RT_{NDT} values for the forging materials are based on Charpy test data for specimens oriented in the "weak" direction per Section B 1.1 of NUREG-0800 Branch Technical Position 5-3 [24].

(d) Generic value taken from Table 9 of [6] Revision 2-A with an associated σ_I of 18.0°F. See Appendix G for more details Use of these values must also meet certain conditions when used in safety evaluations per [6].

(e) Reference [16] identifies Inlet Nozzle 03-001 as Heat # 1V4684-3V<u>*31*</u>20; however, a review of the CMTR confirmed that the heat number is 1V4684-3V<u>*13*</u>20.

(f) Note that the surveillance weld material is not representative of this weld material even though the two welds share the same material heat number, flux type, and lot number The reactor vessel beltline and surveillance welds have low weight-percent copper values due to restrictions on copper content in the beltline region, whereas the nozzle circ. weld seams do not have the low weight percent copper restrictions The embrittlement behavior for these two welds would not be the same. Therefore, surveillance weld data will not be applied to this weld material.

Table 3-2 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Byron Unit 2 Reactor Vessel Materials[(a)]

Reactor Vessel Material	Heat Number	Wt. % Cu[(b)]	Wt. % Ni[(b)]	$RT_{NDT(u)}$[(c)] (°F)
Reactor Vessel Beltline Materials				
Nozzle Shell Forging	4P-6107	0.05	0.74	10
Intermediate Shell Forging	[49D329/49C297]-1-1	0.01	0.70	-20
Lower Shell Forging	[49D330/49C298]-1-1	0.06	0.73	-20
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562	442011 (Linde 80 flux type, Lot # 8061)	0.03	0.67	40
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447	442002 (Linde 80 flux type, Lot # 8064)	0.04	0.63	10
Surveillance Weld Material – Byron Unit 2	442002 (Linde 80 flux type, Lot # 8064)	0.02	0.71	---
Reactor Vessel Extended Beltline Materials				
Inlet Nozzle 01-001	51-2979	0.07	0.86	-10
Inlet Nozzle 01-002	51-2979	0.07	0.86	-20
Inlet Nozzle 02-001	42-5105	0.07	0.84	0
Inlet Nozzle 02-002	42-5105	0.07	0.84	0
Outlet Nozzle 01-001	11-5052	0.09	0.85	-10
Outlet Nozzle 01-002	11-5052	0.08	0.81	-10
Outlet Nozzle 02-001	4-2953	0.09	0.78	-20
Outlet Nozzle 02-002	4-2956	0.09	0.81	-10
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403 (Linde 80 flux type, Lot # 8061)	0.15	0.59	-48.6[(d)]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-545	442010 (Linde 80 flux type, Lot # 8060)	0.22	0.63	-48.6[(d)]
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403 (Linde 80 flux type, Lot # 8061)	0.15	0.59	-48.6[(d)]

Notes:

(a) All data taken from Section 3 of [16], unless otherwise noted.

(b) The chemical compositions are based on Byron Unit 2 CMTR data.

(c) The initial RT_{NDT} values are based on measured data for all beltline and extended beltline materials. Initial RT_{NDT} values for the forging materials are based on Charpy test data for specimens oriented in the "weak" direction per Section B.1.1 of NUREG-0800 Branch Technical Position 5-3 [24].

(d) Generic value taken from Table 9 of [6] Revision 2-A with an associated σ_I of 18.0°F. See Appendix G for more details. Use of these values must also meet certain conditions when used in safety evaluations per [6].

Table 3-3 Initial RT_{NDT} Values for the Byron Units 1 and 2 Reactor Vessel Closure Head and Vessel Flange Materials[(a)]

Reactor Vessel Material	Unit 1 Initial RT_{NDT} (°F)	Unit 2 Initial RT_{NDT} (°F)
Closure Head Flange	60	0
Vessel Flange	10	30

Note

(a) Data taken from [44] and [45].

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [1], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister plant' data. For Byron Unit 1 and Unit 2, sister plant data exists from the Braidwood Unit 1 and Unit 2 surveillance programs (weld Heat #442011). The sister plant data is available in WCAP-18370-NP [46].

The surveillance capsule forging material for Byron Units 1 and 2 is from Intermediate Shell Forging 5P-5933 and Lower Shell Forging [49D330/49C298]-1-1, respectively. Per Appendix D, the forging surveillance data are deemed non-credible for both Byron Unit 1 and Unit 2; therefore, a full margin term will be utilized in the ART calculations contained in Section 7 and the pressurized thermal shock (PTS) calculations contained in Appendix E for the Byron Unit 1 Intermediate Shell Forging and for the Byron Unit 2 Lower Shell Forging.

The surveillance capsule weld material for Byron Units 1 and 2 is Heat # 442002, which is applicable to the intermediate to lower shell circumferential weld at each unit. Per Appendix D, the surveillance weld data are deemed credible; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7 and the PTS calculations contained in Appendix E for Heat # 442002.

Table 4-1 and Table 4-2 summarize the Byron Units 1 and 2 surveillance data for the forging and weld material that will be used in the calculation of the Position 2.1 chemistry factor values for the relevant materials.

Table 4-1 Byron Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.409	28.7
	X	1.49	18.3
	W	2.26	49.5
	Y	3.97	27.8
Intermediate Shell Forging 5P-5933 (Axial)	U	0.409	18.6
	X	1.49	54.6
	W	2.26	29.5
	Y	3.97	11.7
Surveillance Weld Material (Heat # 442002)	U	0.409	5.2
	X	1.49	40.1
	W	2.26	50.6
	Y	3.97	76.7

Notes:

(a) Fluence values are from Section 2.

(b) Measured ΔRT_{NDT} values are from [17].

Table 4-2 Byron Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)
Lower Shell Forging [49D330/49C298]-1-1 (Tangential)	U	0.406	0.0[c]
	W	1.21	2.5
	X	2.18	14.9
	Y	4.19	44.5
Lower Shell Forging [49D330/49C298]-1-1 (Axial)	U	0.406	20.4
	W	1.21	32.1
	X	2.18	39.5
	Y	4.19	68.6
Surveillance Weld Material (Heat # 442002)	U	0.406	8.7
	W	1.21	28.8
	X	2.18	54.2
	Y	4.19	58.7

Notes:

(a) Fluence values are from Section 2

(b) Measured ΔRT_{NDT} values are from [18].

(c) Per [18], this ΔRT_{NDT} value was calculated to be negative. The actual ΔRT_{NDT} is -4.8°F. Physically, this should not occur; therefore, a conservative value of 0°F is shown in this table.

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using Regulatory Guide 1.99, Revision 2 [1], Positions 1.1 and 2.1. Position 1.1 CFs for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of [1]. The best-estimate copper and nickel weight percent values for the Byron Units 1 and 2 reactor vessel materials are provided in Table 3-1 and Table 3-2 of this report.

The Position 2.1 CFs are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in [1]. The Byron Units 1 and 2 surveillance data was summarized in Section 4 of this report and will be utilized in the Position 2.1 CF calculations in this Section. Table 5-1 through Table 5-3 calculate the Byron Units 1 and 2 Position 2.1 CFs.

Sister-plant data from Braidwood Unit 1 and Unit 2 is summarized in WCAP-18370-NP [46] and is also used in Position 2.1 CF calculations in Section 5 of that report. Since the Position 1.1 CF is identical for the Byron Unit 1 and Unit 2 Heat # 442011 vessel welds and the Braidwood Unit 1 and Unit 2 Heat # 442011 surveillance welds, no chemistry adjustment is needed to use the sister plant data. Additionally, since each of the Byron and Braidwood units operate at similar temperatures, no temperature adjustment is necessary. Thus, the Position 2.1 CF for Heat # 442011 from WCAP-18370-NP [46] also applies to the Byron Unit 1 and Unit 2 vessel welds made of Heat # 442011 herein.

Position 1.1 and Position 2.1 CFs are summarized in Table 5-4 and Table 5-5 for Byron Units 1 and 2, respectively. Adjustment of the ΔRT_{NDT} values due to chemistry differences between the surveillance and vessel material per [1] was required for Heat # 442002. The Position 1.1 CF for the intermediate shell to lower shell forging circumferential weld seams is 54°F for both Byron units, while the surveillance program weld Position 1.1 CF is 27°F for both Byron Units. Therefore, the chemistry adjustment factor would be equal to 54 / 27 = 2. However, no temperature adjustments are needed for Heat # 442002 since Byron Units 1 and 2 have similar operating temperatures.

Table 5-1 Byron Unit 1 Reactor Vessel Forging Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence(a) (x 10^{19} n/cm², E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF²
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.409	0.752	28.7	21.58	0.57
	X	1.49	1.110	18.3	20.32	1.23
	W	2.26	1.221	49.5	60.42	1.49
	Y	3.97	1.355	27.8	37.66	1.83
Intermediate Shell Forging 5P-5933 (Axial)	U	0.409	0.752	18.6	13.99	0.57
	X	1.49	1.110	54.6	60.63	1.23
	W	2.26	1.221	29.5	36.01	1.49
	Y	3.97	1.355	11.7	15.85	1.83
				SUM:	266.46	10.25
	$CF_{5P-5933} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (266.46) - (10.25) =$ **26.0°F**					

Notes:

(a) The calculated fluence values are from Section 2

(b) FF = fluence factor = $f^{(0\ 28 - 0\ 10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [17].

Table 5-2 Byron Unit 2 Reactor Vessel Forging Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence(a) (x 10^{19} n/cm², E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF²
Lower Shell Forging [49D330/49C298]-1-1 (Tangential)	U	0.406	0.750	0 0(4)	0.00	0 56
	W	1.21	1.053	2.5	2.63	1.11
	X	2.18	1.211	14.9	18.05	1.47
	Y	4.19	1.366	44.5	60.79	1.87
Lower Shell Forging [49D330/49C298]-1-1 (Axial)	U	0.406	0.750	20.4	15.30	0.56
	W	1.21	1.053	32.1	33.81	1.11
	X	2.18	1.211	39.5	47.85	1.47
	Y	4.19	1.366	68.6	93.72	1.87
				SUM:	272.16	10.01
	$CF_{[49D330/49C298]-1-1} = \Sigma(FF * \Delta RT_{NDT}) - \Sigma(FF^2) = (272.16) - (10.01) =$ **27.2°F**					

Notes:

(a) The calculated fluence values are from Section 2

(b) FF = fluence factor = $f^{(0\ 28 - 0\ 10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [18].

(d) A negative ΔRT_{NDT} value was calculated The actual ΔRT_{NDT} value is -4 8°. Physically, this should not occur, therefore, a conservative value of zero was used in this calculation

Table 5-3 Byron Units 1 and 2 Reactor Vessel Welds Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Byron Unit 1 Surveillance Weld Material (Heat # 442002)	U	0.409	0.752	10.4	7.82	0.57
	X	1.49	1.110	80.2	89.06	1.23
	W	2.26	1.221	101.2	123.54	1.49
	Y	3.97	1.355	153.4	207.79	1.83
Byron Unit 2 Surveillance Weld Material (Heat # 442002)	U	0.406	0.750	17.4	13.05	0.56
	W	1.21	1.053	57.6	60.66	1.11
	X	2.18	1.211	108.4	131.32	1.47
	Y	4.19	1.366	117.4	160.39	1.87
				SUM:	793.63	10.13
	$CF_{\text{Weld Heat \# 442002}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (793.63) \div (10.13) = \mathbf{78.3°F}$					

Notes:

(a) The calculated fluence values are from Section 2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [17] and [18]. The surveillance weld ΔRT_{NDT} values have been multiplied by 2.0 based on the ratio procedure of Reg. Guide 1.99, Rev 2 However, no temperature adjustments were necessary because the Byron Units 1 and 2 surveillance capsules were irradiated at essentially the same temperature.

Table 5-4 Summary of Byron Unit 1 Position 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material	Heat Number	Chemistry Factor (°F)	
		Position 1.1[a]	Position 2.1[b]
Reactor Vessel Beltline Materials			
Nozzle Shell Forging	123J218	31	
Intermediate Shell Forging	5P-5933	26	26.0
Lower Shell Forging	5P-5951	26	
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501	442011	41	31.2[c]
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336	442002	54	78.3
Surveillance Weld Material – Byron Unit 1	442002	27	
Reactor Vessel Extended Beltline Materials			
Inlet Nozzle 03-001	1V4684-3V1320	86	
Inlet Nozzle 03-002	1V4684-3V1320	86	
Inlet Nozzle 04-001	1V4695	95.8	
Inlet Nozzle 04-002	1V4695	85.7	
Outlet Nozzle 01-001	1V4656	77	
Outlet Nozzle 01-002	1V4656	77	
Outlet Nozzle 02-001	2V2557	77	
Outlet Nozzle 02-002	2V2557	77	
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337	442002	139.2	Note (e)
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419	1P5412	168.3[d]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406	504	73.6	

Notes:

(a) Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-1 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [1]

(b) Position 2.1 chemistry factors were taken from Table 5-1 and Table 5-3 of this report, unless otherwise noted. As discussed in Appendix D, the surveillance forging data is deemed non-credible. Also as discussed in Appendix D, the surveillance weld data is deemed credible.

(c) This CF value is calculated using weld surveillance data from the Braidwood Units 1 and 2 surveillance program and is taken from [46]. The data was deemed credible per [46].

(d) This CF value, calculated using Regulatory Guide 1 99, Revision 2, satisfies the condition stipulated in [6] that the CF be no less than 167°F when the initial RT_{NDT} values from BAW-2308 [6] are used

(e) Note that the surveillance weld data will not be applied to this weld material even though the two welds share the same material heat number. Table 3-1 footnote f describes that the surveillance weld material is not representative of this weld material.

Table 5-5 Summary of Byron Unit 2 Position 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material	Heat Number	Chemistry Factor (°F)	
		Position 1.1[(a)]	Position 2.1[(b)]
Reactor Vessel Beltline Materials			
Nozzle Shell Forging	4P-6107	31	
Intermediate Shell Forging	[49D329/49C297]-1-1	20	
Lower Shell Forging	[49D330/49C298]-1-1	37	27.2
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562	442011	41	31.2[(c)]
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447	442002	54	78.3
Surveillance Weld Material – Byron Unit 2	442002	27	
Reactor Vessel Extended Beltline Materials			
Inlet Nozzle 01-001	51-2979	44	
Inlet Nozzle 01-002	51-2979	44	
Inlet Nozzle 02-001	42-5105	44	
Inlet Nozzle 02-002	42-5105	44	
Outlet Nozzle 01-001	11-5052	58	
Outlet Nozzle 01-002	11-5052	51	
Outlet Nozzle 02-001	4-2953	58	
Outlet Nozzle 02-002	4-2956	58	
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403	167.0[(d)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-545	442010	172.0[(e)]	
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403	167.0[(d)]	

Notes:

(a) Position 1 1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-2 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [1]

(b) Position 2.1 chemistry factors were taken from Table 5-2 and Table 5-3 of this report, unless otherwise noted. As discussed in Appendix D, the surveillance forging data is deemed non-credible. Also as discussed in Appendix D, the surveillance weld data is deemed credible.

(c) This CF value is calculated using weld surveillance data from the Braidwood Units 1 and 2 surveillance program and is taken from [46]. The data was deemed credible per [46]

(d) Minimum CF required by [6] as a condition for using values from [6]. The actual calculated CF using Regulatory Guide 1.99, Revision 2 is 144 3°F

(e) This CF value, calculated using Regulatory Guide 1.99, Revision 2, satisfies the condition stipulated in [6] that the CF be no less than 167°F when the initial RT_{NDT} values from BAW-2308 [6] are used.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) 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 the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code [3]. The K_{Ic} curve is given by the following equation:

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

where,

K_{Ic} (ksi√in.) = 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 Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

6.2 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:

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

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

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

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:

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

Axial Flaw Methodology

For plates, forgings, and longitudinal welds, M_m for an inside axial 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 $\sqrt{t} > 3.464$

and, M_m for an outside axial 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 $\sqrt{t} > 3.464$

Circumferential Flaw Methodology

Similarly, for circumferential welds, M_m for an inside or an outside circumferential surface flaw is given by:

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

Where:

p = internal pressure (ksi), R_i = vessel inner radius (in), and t = vessel wall thickness (in.).

For an axial flaw and circumferential flaw of the same ART value, the axial flaw will always produce more limiting P-T limit curves for a given reactor vessel.

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

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m (4)

The maximum K_I produced by radial thermal gradient for the postulated axial or circumferential inside surface defect of G-2120 is:

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \tag{5}$$

where CR is the cooldown rate in °F/hr., or for a postulated axial or circumferential outside surface defect

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \tag{6}$$

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 ASME Code, Section XI, Appendix G, Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Code, 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 Figure 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 1/4T axial or circumferential inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{7}$$

or similarly, K_{It} during heatup for a 1/4T outside axial or circumferential surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{8}$$

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:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{9}$$

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

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2] Section 2.6 (Equations 2.6.2-4 and 2.6.3-1). Finally, the reactor vessel metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A, Revision 4 (Equation 2.6.1-1). This equation is solved utilizing values for thermal diffusivity of 0.518 ft^2/hr at 70°F and 0.379 ft^2/hr at 550°F and a constant convective heat-transfer coefficient value of 7000 Btu/hr-ft^2-°F.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw (the postulated flaw has a depth of 1/4 of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, Paragraph G-2120), the appropriate value for RT_{NDT}, and the reference fracture toughness curve (Equation 1). 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 1/4T flaw of Appendix G to Section XI of 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 vessel wall because the thermal gradients, which increase with increasing cooldown rates, produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Since an inside surface flaw has higher tensile stress than an outside flaw and is subject to more neutron embrittlement than an outside surface flaw in the beltline region, postulation of outside flaw for cooldown conditions is unnecessary. Allowable P-T curves are generated for steady-state (zero-rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

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 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) across the vessel wall 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.

These procedures are needed because there is no direct control on temperature at the 1/4T location and therefore, allowable pressures could be lower 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 P-T 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 inside 1/4T flaw during heatup is lower than the K_{Ic} for the flaw 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 P-T 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 must 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 third portion of the heatup analysis concerns the calculation of the P-T 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 P-T curves for 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 least 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 always be based on analysis of the most critical criterion.

6.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [4] addresses the metal temperature of the closure head flange and vessel flange regions. However, per the technical basis in [5], the flange requirement has been eliminated for Byron Units 1 and 2. An exemption to 10 CFR Part 50, Appendix G was approved for the Byron and Braidwood Units per [25]. This exemption was revised to consider a 53-stud configuration per [43].

6.4 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [4]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [2], the minimum boltup temperature should be 60°F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. Since the limiting unirradiated RT_{NDT} of this region is 60°F per Table 3-3, the minimum boltup temperature for the Byron Units 1 and 2 reactor vessels is 60°F. This limit is shown in Figures 8-1 and 8-2.

7 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (10)$$

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 [15]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided 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 should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \qquad (11)$$

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.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \qquad (12)$$

where x inches (reactor vessel cylindrical shell 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 11 to calculate the ΔRT_{NDT} at the specific depth.

The projected reactor vessel neutron fluence was updated for this analysis and documented in Section 2 of this report. The evaluation methods used in Section 2 are consistent with the methods presented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2].

Table 7-1 and Table 7-2 contain the surface fluence values at 57 EFPY for Byron Units 1 and 2, respectively. These values are used for the development of the P-T limit curves contained in this report. Table 7-1 and Table 7-2 also contain the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2 [1]. The values in this table are used to calculate the 57 EFPY ART values for the Byron Units 1 and 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value. When a generic value is used, the σ_I is obtained from the set of data used to establish the mean. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings when surveillance data is not used or is non-credible and 8.5°F (half the value) for plates or forgings when credible surveillance data is used For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used or is non-credible, and is 14°F (half the value) when credible surveillance capsule data is used. Per [1], the value for σ_Δ need not exceed 0.5 times the mean value of

ΔRT_{NDT}. When using initial RT_{NDT} values based on [6], such as those used for the Byron nozzle-to-shell welds, the values of σ_I and σ_Δ are stipulated by [6].

Contained in Tables 7-3 through 7-8 are the 57 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Byron Units 1 and 2 heatup and cooldown curves. For the materials listed in Table 7-3 through Table 7-8 circumferential flaws are considered in the circumferential weld materials and axial flaws are considered in all other materials. The limiting ART values for Byron Units 1 and 2 are summarized in Table 7-9.

The outlet nozzle forgings and welds for Byron Unit 2 have projected fluence values that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY at the lowest extent of the nozzle weld per Table 2-14. However, per Table 2-8 and Table 2-14, the Units 1 and 2 inlet nozzle forgings and Unit 1 outlet nozzle forgings and welds do have projected fluence values that exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY. Consistent with NRC RIS 2014-11 [14], neutron radiation embrittlement need not be considered herein for the Unit 2 outlet nozzle materials. However, ART calculations for all the inlet and outlet nozzle forging materials are conservatively evaluated in this section with consideration of embrittlement effects and without consideration for attenuation of the fluence through the vessel thickness.

Table 7-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the Byron Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Region	Surface Fluence, f[a] (x10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	1/4T f (x10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[b]	3/4T f (x10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[b]
Reactor Vessel Beltline Materials						
Nozzle Shell Forging	1.15	1.039	0.691	0.896	0.249	0.623
Intermediate Shell Forging	3.19	1.305	1.92	1.178	0.691	0.896
Lower Shell Forging	3.18	1.304	1.91	1.177	0.689	0.895
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501	1.15	1.039	0.691	0.896	0.249	0.623
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336	3.07	1.296	1.84	1.168	0.665	0.886
Reactor Vessel Extended Beltline Materials[c]						
Inlet Nozzles	0.0133	0.133	Note (d)			
Outlet Nozzles	0.0101	0.110				
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.0133	0.133				
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.0101	0.110				

Notes:

(a) 57 EFPY surface fluence values are documented in Table 2-8

(b) FF = fluence factor = $f^{(0.28 - 0.10 \cdot \log(f))}$.

(c) The nozzle forging fluence values are conservatively set equal to the fluence at the respective nozzle welds

(d) The inlet and outlet nozzle forgings and welds are conservatively evaluated without consideration for the attenuation of the fluence through the vessel thickness.

Table 7-2 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the Byron Unit 2 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Region	Surface Fluence, f [(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Surface FF[(b)]	1/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	3/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]
Reactor Vessel Beltline Materials						
Nozzle Shell Forging	1.01	1.003	0.607	0.860	0.219	0.591
Intermediate Shell Forging	2.96	1.288	1.78	1.158	0.641	0.875
Lower Shell Forging	2.90	1.283	1.74	1.153	0.628	0.870
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562	1.05	1.014	0.631	0.871	0.227	0.600
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447	2.82	1.276	1.69	1.145	0.611	0.862
Reactor Vessel Extended Beltline Materials[(c)]						
Inlet Nozzles	0.0129	0.130	Note (e)			
Outlet Nozzles	0.00968[(d)]	0.107				
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.0129	0.130				
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams	0.00968[(d)]	0.107				

Notes:

(a) 57 EFPY surface fluence values are documented in Table 2-14.

(b) FF = fluence factor = $f^{(0.28 - 0.10 \cdot \log(f))}$.

(c) The nozzle forging fluence values are conservatively set equal to the fluence at the respective nozzle welds.

(d) The outlet nozzle materials do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 57 EFPY; therefore, per NRC RIS 2014-11 [14], neutron irradiation embrittlement need not be considered for the nozzle materials herein. However, the results are included for information

(e) The inlet and outlet nozzle forgings and welds are conservatively evaluated without consideration for the attenuation of the fluence through the vessel thickness.

Table 7-3 Adjusted Reference Temperature Evaluation for the Byron Unit 1 Reactor Vessel Beltline Materials through 57 EFPY at the 1/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	1/4T Fluence[b] ($x\ 10^{19}$ n/cm², E > 1.0 MeV)	1/4T FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[d] (°F)	Margin (°F)	ART[e] (°F)
Nozzle Shell Forging 123J218	1.1	31	0.691	0.896	30	27.8	0	13.9	27.8	85.6
Intermediate Shell Forging 5P-5933	1.1	26	1.92	1.178	40	30.6	0	15.3	30.6	101.2
Intermediate Shell Forging Using Non-credible Byron Unit 1 Surveillance Data	2.1	26.0	1.92	1.178	40	30.6	0	15.3	30.6	101.2
Lower Shell Forging 5P-5951	1.1	26	1.91	1.177	10	30.6	0	15.3	30.6	71.2
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501 (Heat # 442011, Linde 80 flux type, Lot # 8086)	1.1	41	0.691	0.896	10	36.7	0	18.4	36.7	83.5
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501 Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	0.691	0.896	10	28.0	0	14.0	28.0	66.0
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336 (Heat # 442002, Linde 80 flux type, Lot # 8873)	1.1	54	1.84	1.168	-30	63.1	0	28.0	56.0	89.1
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336 Using Credible Byron Units 1 and 2 Surveillance Data	2.1	78.3	1.84	1.168	-30	91.4	0	14.0	28.0	89.4

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1

(c) Values are taken from Table 3-1.

(d) Per Appendix D, the intermediate shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of Regulatory Guide 1 99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and Position 2 1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2 1 with credible surveillance data. The weld metal σ_Δ = 28°F for the Position 1.1 data and Position 2 1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-4 Adjusted Reference Temperature Evaluation for the Byron Unit 1 Reactor Vessel Beltline Materials through 57 EFPY at the 3/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART[(e)] (°F)
Nozzle Shell Forging 123J218	1.1	31	0.249	0.623	30	19.3	0	9.7	19.3	68.6
Intermediate Shell Forging 5P-5933	1.1	26	0.691	0.896	40	23.3	0	11.7	23.3	86.6
Intermediate Shell Forging Using Non-credible Byron Unit 1 Surveillance Data	2.1	26.0	0.691	0.896	40	23.3	0	11.7	23.3	86.6
Lower Shell Forging 5P-5951	1.1	26	0.689	0.895	10	23.3	0	11.6	23.3	56.6
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501 (Heat # 442011, Linde 80 flux type, Lot # 8086)	1.1	41	0.249	0.623	10	25.5	0	12.8	25.5	61.1
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501 Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	0.249	0.623	10	19.4	0	9.7	19.4	48.9
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336 (Heat # 442002, Linde 80 flux type, Lot # 8873)	1.1	54	0.665	0.886	-30	47.8	0	23.9	47.8	65.6
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336 Using Credible Byron Units 1 and 2 Surveillance Data	2.1	78.3	0.665	0.886	-30	69.3	0	14.0	28.0	67.3

Notes:

(a) Values are taken from Table 5-4

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1

(d) Per Appendix D, the intermediate shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8 5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for the Position 1 1 data and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2 1 with credible surveillance data. However, σ_Δ need not exceed 0 5*ΔRT_{NDT} per regulatory guidance in [1]

(e) ART values are calculated in accordance with [1].

Table 7-5 Adjusted Reference Temperature Evaluation for the Byron Unit 1 Reactor Vessel Extended Beltline Materials through 57 EFPY at the Surface Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART[g] (°F)
Inlet Nozzle 03-001 (Heat # 1V4684-3V1320)	1.1	86	0.0133	0.133	-10	11.4	0	5.7	11.4	12.8
Inlet Nozzle 03-002 (Heat # 1V4684-3V1320)	1.1	86	0.0133	0.133	-20	11.4	0	5.7	11.4	2.8
Inlet Nozzle 04-001 (Heat # 1V4695)	1.1	95.8	0.0133	0.133	-20	12.7	0	6.4	12.7	5.4
Inlet Nozzle 04-002 (Heat # 1V4695)	1.1	85.7	0.0133	0.133	-20	11.4	0	5.7	11.4	2.7
Outlet Nozzle 01-001 (Heat # 1V4656)	1.1	77	0.0101	0.110	0	8.5	0	4.3	8.5	17.0
Outlet Nozzle 01-002 (Heat # 1V4656)	1.1	77	0.0101	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-001 (Heat # 2V2557)	1.1	77	0.0101	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-002 (Heat # 2V2557)	1.1	77	0.0101	0.110	-10	8.5	0	4.3	8.5	7.0
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337 (Heat # 442002, Linde 80 flux type, Lot # 8873)	1.1	139.2	0.0133	0.133	-10	18.5	0	9.2	18.5	26.9
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419 (Heat # 1P5412, Linde 80 flux type, Lot # 8969)	1.1	168.3	0.0101	0.110	-48.6	18.6	18.0[d]	28.0[f]	66.6	36.6
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406 Heat # 504, Linde 80 flux type, Lot # 8968)	1.1	73.6	0.0101	0.110	10	8.1	0	4.1	8.1	26.2

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(U)}$ value.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1, and the weld metal σ_Δ = 28°F for Position 1.1. However, σ_Δ need not exceed 0 5*ΔRT_{NDT} per regulatory guidance in [1].

(f) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(g) ART values are calculated in accordance with [1].

Table 7-6 Adjusted Reference Temperature Evaluation for the Byron Unit 2 Reactor Vessel Beltline Materials through 57 EFPY at the 1/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART[(e)] (°F)
Nozzle Shell Forging 4P-6107	1.1	31	0.607	0.860	10	26.7	0	13.3	26.7	63.3
Intermediate Shell Forging [49D329/49C297]-1-1	1.1	20	1.78	1.158	-20	23.2	0	11.6	23.2	26.3
Lower Shell Forging [49D330/49C298]-1-1	1.1	37	1.74	1.153	-20	42.6	0	17.0	34.0	56.6
Lower Shell Forging Using Non-credible Byron Unit 2 Surveillance Data	2.1	27.2	1.74	1.153	-20	31.3	0	15.7	31.3	42.7
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562 (Heat # 442011, Linde 80 flux type, Lot # 8061)	1.1	41	0.631	0.871	40	35.7	0	17.9	35.7	111.4
Nozzle to Intermediate Shell Forging Circ. Weld Seam WF-562 Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	0.631	0.871	40	27.2	0	13.6	27.2	94.3
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447 (Heat # 442002, Linde 80 flux type, Lot # 8064)	1.1	54	1.69	1.145	10	61.8	0	28.0	56.0	127.8
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 Using Credible Byron Units 1 and 2 Surveillance Data	2.1	78.3	1.69	1.145	10	89.7	0	14.0	28.0	127.7

Notes:

(a) Values are taken from Table 5-5

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1 1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance date The weld metal σ_Δ = 28°F for the Position 1.1 and Position 2 1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0 5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-7 Adjusted Reference Temperature Evaluation for the Byron Unit 2 Reactor Vessel Beltline Materials through 57 EFPY at the 3/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	3/4T Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[d] (°F)	Margin (°F)	ART[e] (°F)
Nozzle Shell Forging 4P-6107	1.1	31	0.219	0.591	10	18.3	0	9.2	18.3	46.6
Intermediate Shell Forging [49D329/49C297]-1-1	1.1	20	0.641	0.875	-20	17.5	0	8.8	17.5	15.0
Lower Shell Forging [49D330/49C298]-1-1	1.1	37	0.628	0.870	-20	32.2	0	16.1	32.2	44.4
Lower Shell Forging Using Non-credible Byron Unit 2 Surveillance Data	2.1	27.2	0.628	0.870	-20	23.7	0	11.8	23.7	27.3
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562 (Heat # 442011, Linde 80 flux type, Lot # 8061)	1.1	41	0.227	0.600	40	24.6	0	12.3	24.6	89.2
Nozzle to Intermediate Shell Forging Circ. Weld Seam WF-562 Using Credible Braidwood Units 1 and 2 Surveillance Data	2.1	31.2	0.227	0.600	40	18.7	0	9.4	18.7	77.5
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447 (Heat # 442002, Linde 80 flux type, Lot # 8064)	1.1	54	0.611	0.862	10	46.5	0	23.3	46.5	103.1
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 Using Credible Byron Units 1 and 2 Surveillance Data	2.1	78.3	0.611	0.862	10	67.5	0	14.0	28.0	105.5

Notes.

(a) Values are taken from Table 5-5.

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of regulatory Guide 1 99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance date. The weld metal σ_Δ = 28°F for the Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-8 Adjusted Reference Temperature Evaluation for the Byron Unit 2 Reactor Vessel Extended Beltline Materials through 57 EFPY at the Surface Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm², E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART[h] (°F)
Inlet Nozzle 01-001 (Heat # 51-2979)	1.1	44	0.0129	0.130	-10	5.7	0	2.9	5.7	1.4
Inlet Nozzle 01-002 (Heat # 51-2979)	1.1	44	0.0129	0.130	-20	5.7	0	2.9	5.7	-8.6
Inlet Nozzle 02-001 (Heat # 42-5105)	1.1	44	0.0129	0.130	0	5.7	0	2.9	5.7	11.4
Inlet Nozzle 02-002 (Heat # 42-5105)	1.1	44	0.0129	0.130	0	5.7	0	2.9	5.7	11.4
Outlet Nozzle 01-001 (Heat # 11-5052)	1.1	58	0.00968	0.107	-10	6.2	0	3.1	6.2	2.4
Outlet Nozzle 01-002 (Heat # 11-5052)	1.1	51	0.00968	0.107	-10	5.5	0	2.7	5.5	0.9
Outlet Nozzle 02-001 (Heat # 4-2953)	1.1	58	0.00968	0.107	-20	6.2	0	3.1	6.2	-7.6
Outlet Nozzle 02-002 (Heat # 4-2956)	1.1	58	0.00968	0.107	-10	6.2	0	3.1	6.2	2.4
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559 (Heat # 41403, Linde 80 flux type, Lot # 8061)	1.1	167.0[f]	0.0129	0.130	-48.6	21.7	18.0[d]	28.0[g]	66.6	39.7
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-545 (Heat # 442010, Linde 80 flux type, Lot # 8060)	1.1	172.0	0.00968	0.107	-48.6	18.4	18.0[d]	28.0[g]	66.6	36.4
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-559 (Heat # 41403, Linde 80 flux type, Lot # 8061)	1.1	167.0[f]	0.00968	0.107	-48.6	17.9	18.0[d]	28.0[g]	66.6	35.9

Notes:

(a) Values are taken from Table 5-5.

(b) Values are taken from Table 7-2.

(c) Values are taken from Table 3-2.

(d) Table 9 of [6] Revision 2-A identifies σ_I = 18 0°F associated with the use of the generic $RT_{NDT(U)}$ value.

(e) Per the guidance of regulatory Guide 1 99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1, and the weld metal σ_Δ = 28°F for Position 1.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(f) Value is required minimum per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(g) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(h) ART values are calculated in accordance with [1].

Table 7-9 Limiting ART Values for Byron Units 1 and 2 at 57 EFPY[a]

	Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)	Limiting Material
"Axial Flaw" Method	101.2	86.6	Byron Unit 1 Intermediate Shell Forging 5P-5933
"Circumferential Flaw" Method	127.7[b]	105.5[b]	Byron Unit 2 Intermediate to Lower Shell Forging Circumferential Weld Seam (Heat # 442002) using Credible Byron Units 1 and 2 Surveillance Data

Notes:

(a) Values are the limiting values from Tables 7-3 through 7-8. For the materials listed in Table 7-3 through Table 7-8 circumferential flaws are considered in the circumferential weld materials and axial flaws are considered in all other materials.

(b) For the Intermediate to Lower Shell Weld materials, the ART values calculated using Position 2.1 CFs are used instead of the Position 1.1 results because credible surveillance data is available.

8 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 cylindrical beltline region using the methods discussed in Sections 6 and 7 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [2].

The highest ART values for Byron Units 1 and 2 correspond to the Byron Unit 2 Intermediate to Lower Shell Forging Circumferential Weld (Heat # 442002). However, since this material is a "Circumferential Flaw" material, the applied membrane (pressure) stress and resulting stress intensity factor at the postulated flaw location are much lower than for the most limiting "Axial Flaw" material. Consequently, this material does not produce the most limiting P-T limit curves. The most limiting P-T limit curves for Byron Units 1 and 2 are produced by using the "Axial Flaw" methodology and the limiting "Axial Flaw" material ART values. Thus, the limiting ART values for Byron Units 1 and 2 used in the generation of the P-T limit curves are based on the Byron Unit 1 Intermediate Shell Forging from Table 7-9. For P-T limit curve development, the limiting ART values are conservatively rounded up as shown below in Table 8-1.

Table 8-1 ART Values to be used in P-T Limit Curves Development for Byron Units 1 and 2 at 57 EFPY[(a)]

Limiting Material	Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)
Byron Unit 1 Intermediate Shell Forging 5P-5933	102	87

Note:

(a) Values correspond to the limiting "Axial Flaw" Method ART values in Table 7-9 rounded up to the nearest whole number

Figure 8-1 presents the limiting heatup curves without margins for possible instrumentation errors using a heatup rate of 100°F/hr applicable for 57 EFPY, without the flange requirements and using the "Axial Flaw" methodology. Figure 8-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0°, -25°, -50°, and -100°F/hr applicable for 57 EFPY, without the flange requirements and using the "Axial Flaw" methodology. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G. As discussed in Section 6, the use of the "Axial Flaw" methodology and the limiting "Axial Flaw" ART values produce the most limiting P-T limit curves for Byron Units 1 and 2. The exclusion of the 10 CFR 50, Appendix G flange requirements is justified by WCAP-16143-P and approved by the NRC in [25]. The NRC originally approved the exemption from the Appendix G flange requirements in [25] based on Revision 0 of WCAP-16143-P. Reference [43] revised the exemption to account for Revision 1 of WCAP-16143-P, which considers a 53-stud configuration.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 8-1 and 8-2. 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 P-T combinations are to the right of the criticality limit line shown in Figure 8-1 (heatup curve only). 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 the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

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

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 6-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\,(T - RT_{NDT})]}$ [see page 6-1 Equation (1)],

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies P-T limits for core operation in order to provide additional margin during actual power production. The P-T limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding P-T curve for heatup and cooldown calculated as described in Section 6 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the Byron Units 1 and 2 reactor vessels at 57 EFPY is 162°F; this temperature value is calculated based on Equation (13). The vertical line drawn from these points on the P-T curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 8-1 and 8-2 define all of the above limits for ensuring prevention of non-ductile failure for the Byron Units 1 and 2 reactor vessels for 57 EFPY without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 8-1 and 8-2 are presented in Tables 8-2, 8-3, and 8-4. Vacuum refill limits for the Reactor Coolant System (RCS) are displayed on Figures 8-1 and 8-2 by showing a minimum pressure of 0 psia. This approach is consistent with the current Byron Units 1 and 2 Pressure and Temperature Limit Reports (PTLRs) [26 and 27].

As discussed in Appendix B, the P-T limits developed for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for Byron Units 1 and 2 at 57 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Byron Unit 1 Intermediate Shell Forging 5P-5933 using Regulatory Guide 1.99 Position 2.1 non-credible surveillance data

LIMITING ART VALUES AT 57 EFPY: 1/4T, 102°F (Axial Flaw)
3/4T, 87°F (Axial Flaw)

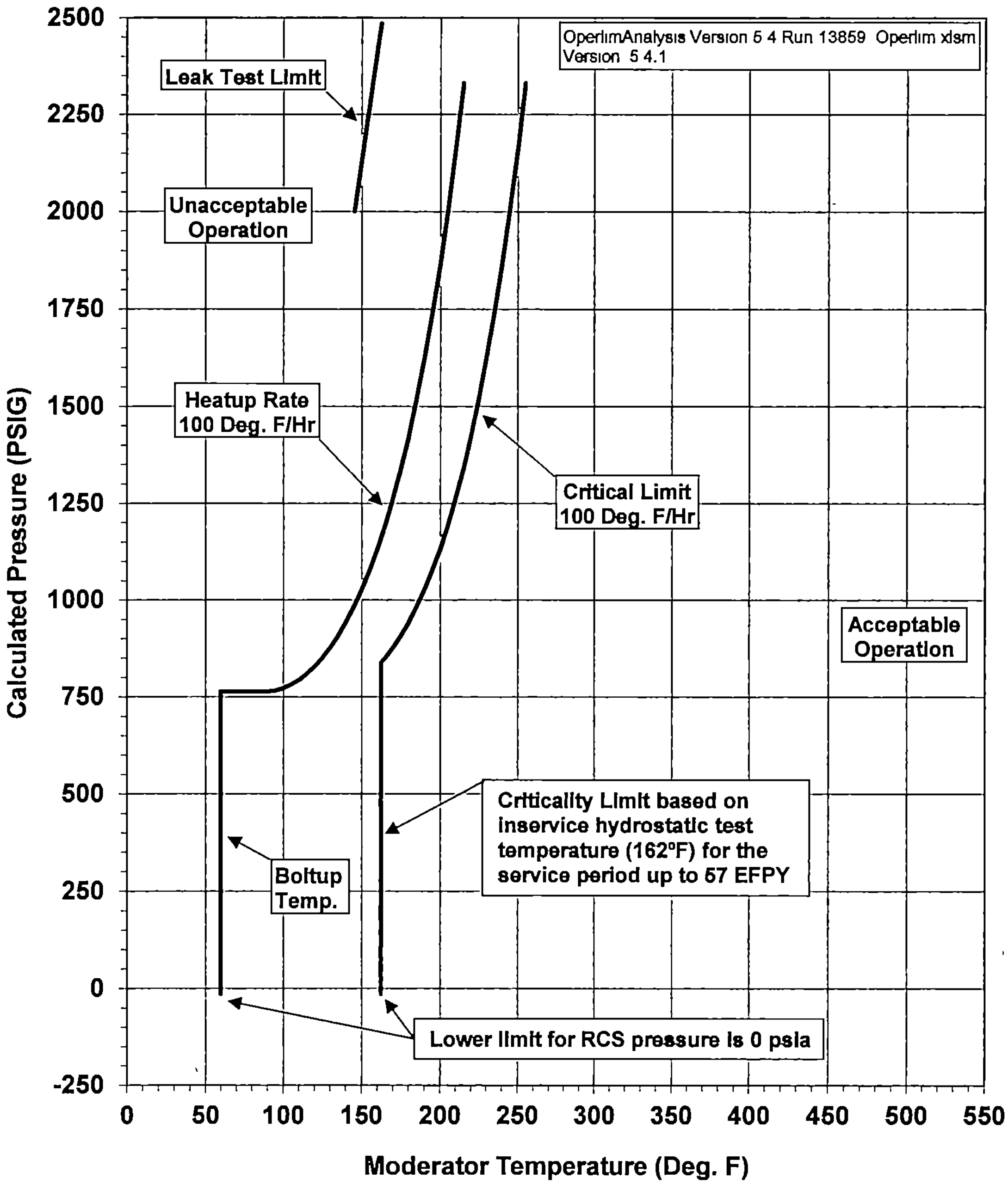

Figure 8-1 Byron Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for 57 EFPY (<u>without</u> Flange Requirements and <u>without</u> Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Byron Unit 1 Intermediate Shell Forging 5P-5933 using Regulatory Guide 1.99 Position 2.1 non-credible surveillance data

LIMITING ART VALUES AT 57 EFPY: 1/4T, 102°F (Axial Flaw)
3/4T, 87°F (Axial Flaw)

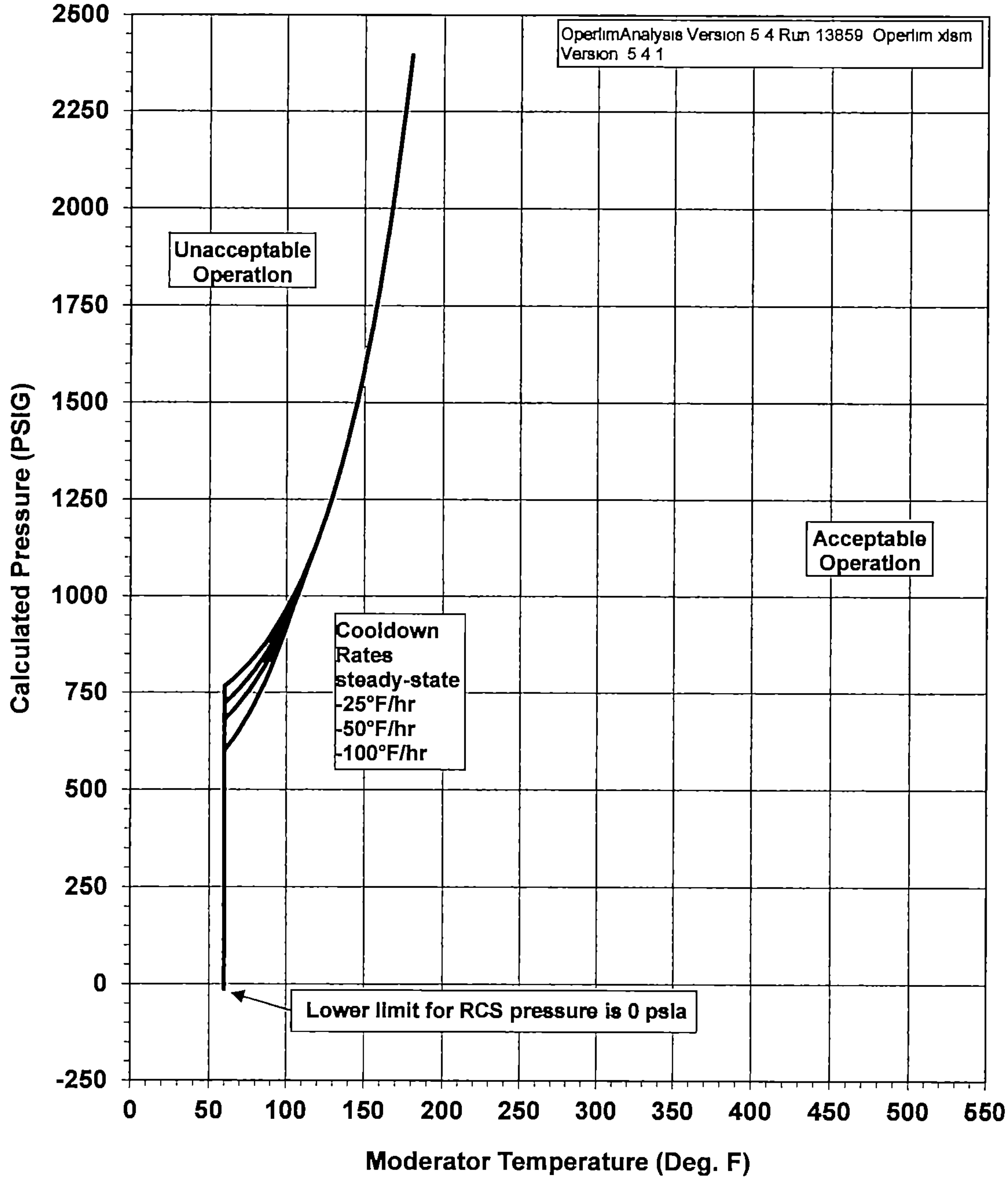

Figure 8-2 Byron Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, and 100°F/hr) Applicable for 57 EFPY (without Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-2 Byron Units 1 and 2 57 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/o Flange Requirements, and w/o Margins for Instrumentation Errors)

100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)
60	Note (a)	162	Note (a)
60	764	162	839
65	764	165	851
70	764	170	877
75	764	175	908
80	764	180	943
85	764	185	982
90	764	190	1027
95	766	195	1076
100	772	200	1132
105	781	205	1194
110	793	210	1263
115	809	215	1339
120	828	220	1424
125	851	225	1518
130	877	230	1622
135	908	235	1737
140	943	240	1864
145	982	245	2004
150	1027	250	2159
155	1076	255	2330
160	1132	-	-
165	1194	-	-
170	1263	-	-
175	1339	-	-
180	1424	-	-
185	1518	-	-
190	1622	-	-
195	1737	-	-
200	1864	-	-
205	2004	-	-
210	2159	-	-
215	2330	-	-

Note
(a) The minimum acceptable pressure is 0 psia.

Table 8-3 Byron Units 1 and 2 57 EFPY Leak Test Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/o Flange Requirements, and w/o Margins for Instrumentation Errors)

Leak Test Limits	
T (°F)	P (psig)
145	2000
162	2485

Table 8-4 Byron Units 1 and 2 57 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/o Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		-25°F/hr Cooldown		-50°F/hr Cooldown		-100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	Note (a)	60	Note (a)	60	Note (a)	60	Note (a)
60	766	60	723	60	681	60	601
65	783	65	742	65	702	65	628
70	802	70	763	70	726	70	658
75	823	75	786	75	752	75	691
80	846	80	812	80	780	80	727
85	871	85	840	85	812	85	768
90	900	90	872	90	848	90	814
95	931	95	907	95	887	95	864
100	965	100	945	100	930	100	920
105	1003	105	988	105	979	105	979
110	1045	110	1035	110	1032	110	1032
115	1092	115	1088	115	1088	115	1088
120	1143	120	1143	120	1143	120	1143
125	1200	125	1200	125	1200	125	1200
130	1263	130	1263	130	1263	130	1263
135	1332	135	1332	135	1332	135	1332
140	1409	140	1409	140	1409	140	1409
145	1494	145	1494	145	1494	145	1494
150	1587	150	1587	150	1587	150	1587
155	1691	155	1691	155	1691	155	1691
160	1805	160	1805	160	1805	160	1805
165	1932	165	1932	165	1932	165	1932
170	2071	170	2071	170	2071	170	2071
175	2226	175	2226	175	2226	175	2226
180	2396	180	2396	180	2396	180	2396

Note:
(a) The minimum acceptable pressure is 0 psia.

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APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) and vessel temperatures for the maximum heatup and cooldown rates at 57 EFPY for Byron Units 1 and 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 88.75 inches
- 3/4T Radius = 93.00 inches

Table A-1 K_{It} and Vessel Temperature Values for Byron Units 1 and 2 at 57 EFPY 100°F/hr Heatup Curves (w/o Flange Requirements and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.008	-0.994	55.046	0.477
65	58.618	-2.441	55.313	1.442
70	61.705	-3.684	56.014	2.421
75	65.007	-4.860	57.197	3.337
80	68.582	-5.872	58.805	4.153
85	72.269	-6.794	60.798	4.881
90	76.140	-7.591	63.135	5.523
95	80.110	-8.318	65.772	6.097
100	84.212	-8.952	68.677	6.604
105	88.400	-9.527	71.814	7.056
110	92.684	-10.031	75.155	7.459
115	97.039	-10.491	78.674	7.820
120	101.464	-10.896	82.350	8.144
125	105.948	-11.267	86.163	8.435
130	110.483	-11.595	90.098	8.697
135	115.065	-11.897	94.140	8.934
140	119.685	-12.166	98.275	9.147
145	124.343	-12.415	102.494	9.342
150	129.029	-12.637	106.785	9.518
155	133.746	-12.845	111.140	9.680
160	138.484	-13.032	115.552	9.828
165	143.247	-13.207	120.013	9.964
170	148.025	-13.366	124.518	10.089
175	152.822	-13.516	129.062	10.206
180	157.632	-13.653	133.640	10.314
185	162.457	-13.784	138.248	10.415
190	167.291	-13.905	142.883	10.509
195	172.137	-14.020	147.541	10.598
200	176.990	-14.128	152.219	10.682
205	181.852	-14.232	156.916	10.762
210	186.720	-14.329	161.629	10.838
215	191.595	-14.424	166.355	10.910

Table A-2 K_{It} and Vessel Temperature Values for Byron Units 1 and 2 at 57 EFPY 100°F/hr Cooldown Curves (w/o Flange Requirements and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
215	241.311	16.555
210	236.227	16.489
205	231.144	16.422
200	226.059	16.356
195	220.975	16.289
190	215.891	16.223
185	210.806	16.155
180	205.721	16.089
175	200.636	16.021
170	195.551	15.955
165	190.466	15.887
160	185.381	15.820
155	180.295	15.753
150	175.210	15.686
145	170.125	15 619
140	165.040	15.552
135	159.955	15.485
130	154.870	15.419
125	149.785	15.352
120	144.700	15.285
115	139.615	15.219
110	134.530	15.152
105	129.445	15.086
100	124.361	15.020
95	119.276	14.954
90	114.192	14.888
85	109.108	14.822
80	104.024	14.757
75	98.940	14.691
70	93.856	14.626
65	88.772	14.560
60	83.690	14.495

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES

As described in NRC Regulatory Issue Summary (RIS) 2014-11 [14], reactor vessel non-beltline materials may define pressure-temperature (P-T) limit curves that are more limiting than those calculated for the reactor vessel cylindrical shell beltline materials. Reactor vessel nozzles, penetrations, and other discontinuities have complex geometries that can exhibit significantly higher stresses than those for the reactor vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperatures (RT_{NDT}) for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries.

The methodology contained in WCAP-14040-A, Revision 4 [2] was used in the main body of this report to develop P-T limit curves for the limiting Byron Units 1 and 2 cylindrical shell beltline material; however, Reference [2] does not consider ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. Due to the geometric discontinuity, the inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in Appendix B herein. P-T limit curves are determined for the reactor vessel nozzle corner region for Byron Units 1 and 2 and compared to the P-T limit curves for the reactor vessel traditional beltline region in order to determine if the nozzles can be more limiting than the reactor vessel beltline as the plant ages and the vessel accumulates more neutron fluence. The increase in neutron fluence as the plant ages causes a concern for embrittlement of the reactor vessel above the beltline region. Therefore, the P-T limit curves are developed for the nozzle inside corner region since the geometric discontinuity results in high stresses due to internal pressure and the cooldown transient. The cooldown transient is analyzed as it results in tensile stresses at the inside surface of the nozzle corner.

A flaw is postulated at the inside surface of the reactor vessel nozzle corner, and stress intensity factors are determined based on the rounded curvature of the nozzle geometry. The allowable pressure is then calculated based on the fracture toughness of the nozzle material and the stress intensity factors for the postulated flaw.

B.1 CALCULATION OF ADJUSTED REFERENCE TEMPERATURES

The fracture toughness (K_{Ic}) used for the inlet and outlet nozzle material is defined in Appendix G of the Section XI ASME Code, as discussed in Section 6 of this report. The K_{Ic} fracture toughness curve is dependent on the Adjusted Reference Temperature (ART) value for irradiated materials. The ART values for the inlet and outlet nozzle materials are determined using the methodology contained in Regulatory Guide 1.99, Revision 2 [1], which is described in Section 7 of this report, as well as weight percent (wt. %) copper (Cu) values, wt. % nickel (Ni) values, initial RT_{NDT} values, and projected neutron fluence as inputs. The initial material properties for each of the reactor vessel inlet and outlet nozzle forging materials are documented in Table 3-1 and Table 3-2. A summary of the limiting inlet and outlet nozzle ART values used in the nozzle P-T limit curves analysis for Byron Units 1 and 2 is presented in Table B-1.

Nozzle Material Properties

The Byron Units 1 and 2 nozzle initial material properties are provided in Table 3-1 and Table 3-2, respectively. Cu and Ni weight percent (wt. %) values were obtained from the Byron Units 1 and 2 CMTRs for each of the Byron Units 1 and 2 reactor vessel inlet and outlet nozzles.

ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3) [15] was used to determine all initial RT_{NDT} values for the inlet and outlet nozzles. All initial RT_{NDT} values are consistent with those reported in WCAP-17606-NP [16]. The weak-direction Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was identified in CMTRs for Byron Units 1 and 2, and these weak-direction results were utilized in determining initial RT_{NDT} values.

Nozzle Calculated Neutron Fluence Values

The maximum fast neutron (E > 1.0 MeV) exposure of the Byron Units 1 and 2 reactor vessel materials is discussed in Section 2 of this report. The fluence values used in the inlet and outlet nozzle ART calculations were calculated at the lowest extent of the nozzles (i.e., the nozzle to nozzle shell weld locations) and are therefore taken at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle, for conservatism.

Per Table 2-8 and Table 2-14, the inlet nozzles at both units are determined to receive a projected maximum fluence of 1.33×10^{17} n/cm^2 (E > 1.0 MeV) and the outlet nozzles at both units are determined to receive a projected maximum fluence of 1.01×10^{17} n/cm^2 (E > 1.0 MeV) at the lowest extent of the nozzles at 57 EFPY. In Section 7, embrittlement of the inlet and outlet nozzle materials is considered for comparison with the beltline ART values. Conservatively, the fluence values utilized for the nozzle evaluations do not consider attenuation through the reactor vessel thickness.

The limiting nozzle ART values used for determination of the nozzle P-T limit curves are summarized in Table B-1.

Table B-1 Summary of the Limiting ART Values for the Byron Units 1 and 2 Inlet and Outlet Nozzle Materials

EFPY	Nozzle Description	Limiting ART Value (°F)
57	Byron Unit 1 Inlet Nozzles	12.8[(a)]
	Byron Unit 1 Outlet Nozzles	17.0[(b)]
	Byron Unit 2 Inlet Nozzles	11.4[(c)]
	Byron Unit 2 Outlet Nozzles	2.4[(d)]

Notes:

(a) Limiting value of all Byron Unit 1 Inlet Nozzles from Table 7-5. Limiting value corresponds to Byron Unit 1 Inlet Nozzle 03-001.

(b) Limiting value of all Byron Unit 1 Outlet Nozzles from Table 7-5. Limiting value corresponds to Byron Unit 1 Outlet Nozzle 01-001.

(c) Limiting value of all Byron Unit 2 Inlet Nozzles from Table 7-8. Limiting value corresponds to Byron Unit 2 Inlet Nozzles 02-001 and 02-002.

(d) Limiting value of all Byron Unit 2 Outlet Nozzle from Table 7-8. Limiting value corresponds to Byron Unit 2 Outlet Nozzles 01-001 and 02-002. This value conservatively considers embrittlement, even though the fluence is less than the threshold of 10^{17} n/cm^2 (E > 1.0 MeV) from RIS 2014-11.

B.2 NOZZLE COOLDOWN PRESSURE-TEMPERATURE LIMITS

Allowable pressures are determined for a given temperature based on the fracture toughness of the limiting nozzle material along with the appropriate pressure and thermal stress intensity factors. The Byron Units 1 and 2 nozzle fracture toughness used to determine the P-T limits is calculated using the limiting inlet and outlet nozzle ART values from Table B-1. The stress intensity factor correlations used for the nozzle corners are provided in ORNL study, ORNL/TM-2010/246 [19], and are consistent with ASME PVP2011-57015 [20]. The methodology includes postulating an inside surface nozzle corner flaw and calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

For the inlet and outlet nozzles, a 3-inch flaw was used for the generation of the nozzle P-T limit curves per guidance from Article G-2120 of the ASME Section XI Code [3]. In lieu of using a 1/4T circular corner flaw depth for the limiting inlet and outlet nozzles, Article G-2120 states that for sections greater than 12 inches thick, the postulated flaw depth based on the 12-inch section may be used. Thus, a 1/4T flaw for a 12-inch section is used (1/4 * 12" = 3") for the generation of the nozzle P-T limit curves for the inlet and outlet nozzles.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the following stress intensity factor expression

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

where,

K_I = stress intensity factor for a circular corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with a 3-inch flaw

The Byron Units 1 and 2 reactor vessel inlet nozzle P-T limit curves are shown in Figure B-1, while the outlet nozzle P-T limit curves are shown in Figure B-2. The nozzle P-T limit curves are based on the stress intensity factor expression discussed previously; also shown in these figures are the traditional

beltline cooldown P-T limit curves from Figure 8-2. The nozzle P-T limit curves are provided for a cooldown rate of 100°F/hr, along with a steady-state curve.

An outside surface flaw in the nozzle was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is also not provided since it would be less limiting than the cooldown nozzle P-T limit curve in Figures B-1 and B-2 for an inside surface flaw. Additionally, the cooldown transient is more limiting than the heatup transient since it results in tensile stresses (rather than compressive stresses) at the inside surface of the nozzle corner.

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical shell beltline curves. Therefore, the P-T limits provided in Section 8 for 57 EFPY remain limiting for the beltline and non-beltline reactor vessel components.

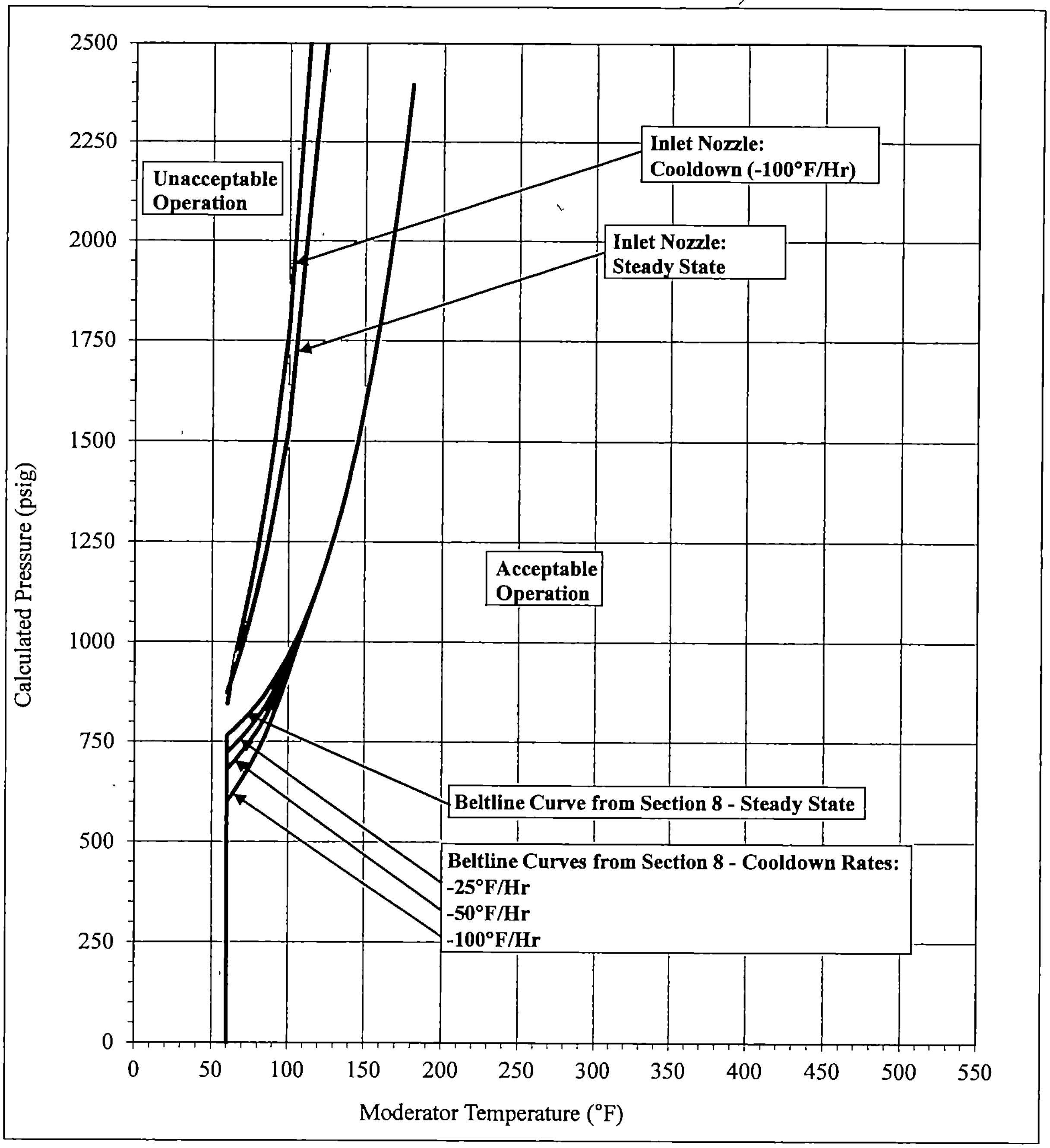

Figure B-1 Comparison of Byron Units 1 and 2 57 EFPY Beltline P-T Limits to 57 EFPY Limiting Inlet Nozzle P-T Limits, without Margins for Instrumentation Error

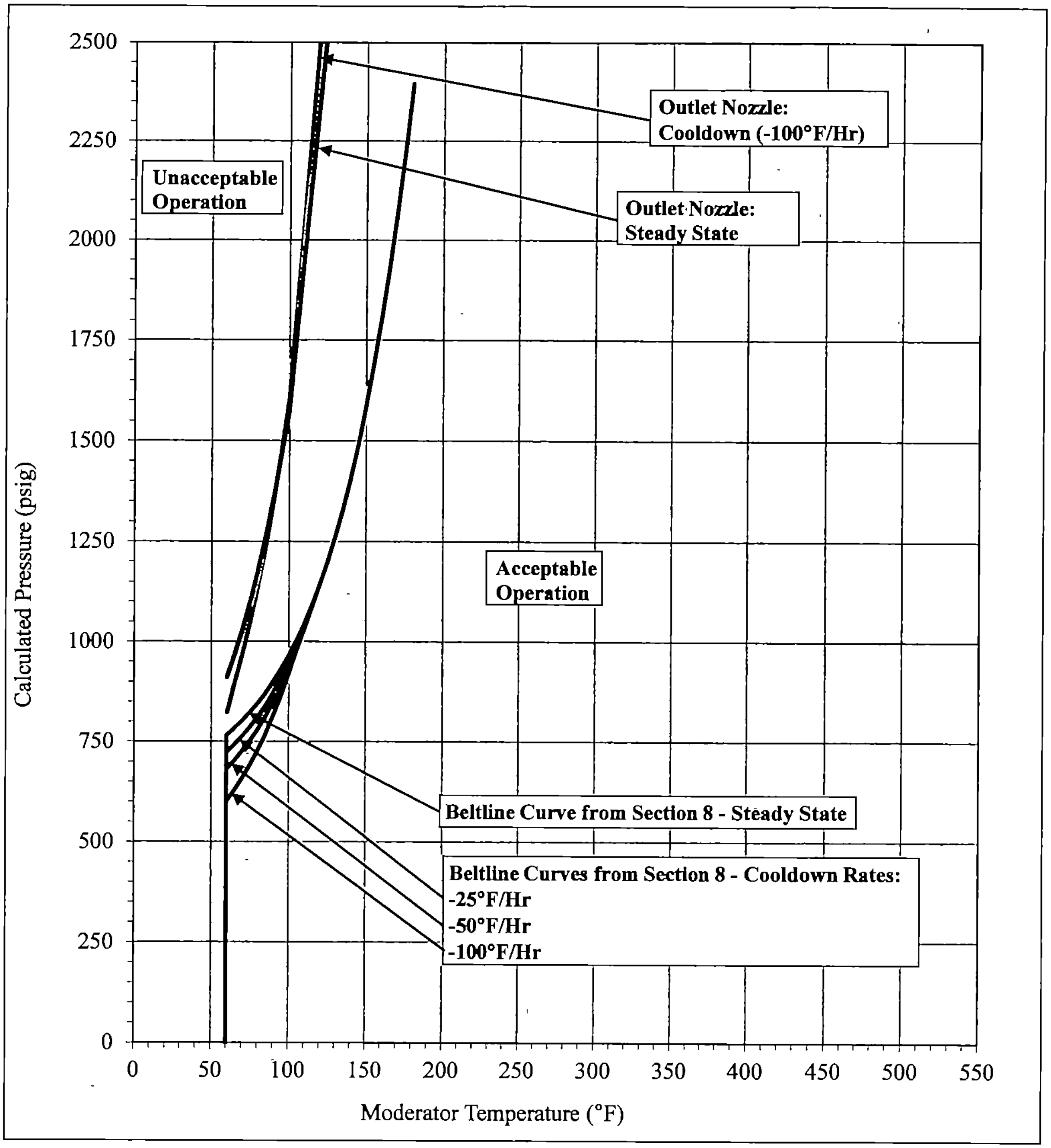

Figure B-2 Comparison of Byron Units 1 and 2 57 EFPY Beltline P-T Limits to 57 EFPY Limiting Outlet Nozzle P-T Limits, without Margins for Instrumentation Error

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G [4] requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature (LST) requirement for all RCPB components, which is specified in NB-2332(b) and NB-3211 of the ASME Code, Section III [15], is the relevant requirement that would affect the P-T limits. This requirement is applicable to ferritic materials outside of the RV with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps, and valves [15].

The Byron Units 1 and 2 reactor coolant systems do not have ferritic materials in the Class 1 piping, pumps, and valves (fabricated instead with stainless steel). Therefore, the LST requirements of the ASME Code, Section III, NB-2332(b) and NB-3211 [15] for these components do not need to be considered.

RIS 2014-11 [14] also addresses other ferritic components of the reactor coolant system relative to P-T limit, and states the following:

> *As specified in Sections I and IV.A of 10 CFR Part 50, Appendix G, ferritic RCPB components outside of the reactor vessel must meet the applicable requirements of ASME Code, Section III, "Rules for Construction of Nuclear Facility Components."*

The other ferritic RCPB components that are not part of the RV beltline or extended beltline for Byron Units 1 and 2 consist of the RV closure head, steam generators, and pressurizer. The Byron Units 1 and 2 primary system components are analyzed to the following ASME Code Section III Editions and met all applicable requirements at the time of construction. Therefore, no further consideration of these components is necessary:

- Reactor Vessel Closure Heads for Byron Unit 1 and Unit 2 – ASME Code Section III, 1971 Edition through Summer 1973 Addenda. These components were previously analyzed in WCAP-16143-P [5].
- Replacement Steam Generator for Byron Unit 1 – ASME Code Section III, 1986 Edition
- Steam Generator for Byron Unit 2 – ASME Code Section III, 1971 Edition through Summer 1972 and Winter 1974 Addenda
- Pressurizers for Byron Unit 1 and Unit 2 – ASME Code Section III, 1971 Edition through Summer 1973 Addenda

APPENDIX D BYRON UNITS 1 AND 2 SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

Regulatory Guide 1.99, Revision 2 [1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position 2.1 of [1], describes the method for calculating the adjusted reference temperature of reactor vessel beltline materials using surveillance capsule data. The methods of Position 2.1 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been four surveillance capsules removed and tested from each of the Byron Units 1 and 2 reactor vessels. To use the surveillance data, the data must be shown to be credible. In accordance with [1], the credibility of the surveillance data will be judged based on five criteria.

The purpose of this evaluation is to apply the credibility requirements of [1], to the Byron Units 1 and 2 reactor vessel surveillance data, including fluence values updated in Section 2, to determine if the surveillance data is credible.

It is noted that Byron Units 1 and 2 also utilize surveillance data from Braidwood Units 1 and 2 (Heat # 442011). This surveillance data was deemed credible in [46]. Since the Braidwood credibility evaluation does not depend on any Byron-specific information, the data is also credible for use at Byron Units 1 and 2 and need not be re-analyzed herein.

D.1 BYRON UNIT 1 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [4], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Byron Unit 1 reactor vessel beltline region consists of the following materials:

1. Nozzle Shell Forging 123J218
2. Intermediate Shell Forging 5P-5933
3. Lower Shell Forging 5P-5951
4. Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld Seam WF-501 (Weld Wire Heat # 442011, Linde 80 Flux Type, Flux Lot # 8086)
5. Intermediate Shell Forging to Lower Shell Forging Circumferential Weld Seam WF-336 (Weld Wire Heat # 442002, Linde 80 Flux Type, Flux Lot # 8873)

The Byron Unit 1 surveillance program utilizes tangential and axial test specimens from the Intermediate Shell Forging. The surveillance weld metal was fabricated with weld wire Heat # 442002, Flux Type Linde 80, Lot # 8873.

At the time when the Byron Unit 1 surveillance program material was selected, it was believed that copper and phosphorus were the elements most important to the embrittlement of reactor vessel steels and the Nozzle Shell Forging was not considered a "beltline" material. The Intermediate Shell Forging had a higher initial RT_{NDT} and a lower initial USE value than the Lower Shell Forging. In addition, the Lower and Intermediate Shell Forgings had essentially the same copper and phosphorus content. Based on this comparison of the beltline forging materials (Intermediate and Lower Shell Forgings), the Intermediate Shell Forging was chosen for the surveillance program.

Weld seam WF-336, on the other hand, was considered the only weld in the beltline region and therefore was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 442002), the same type of flux (Linde 80), and the same flux lot (# 8873) as the Intermediate to Lower Shell Forging Circumferential Weld Seam.

Therefore, the materials selected for use in the Byron Unit 1 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

Based on this discussion, Criterion 1 is met for the Byron Unit 1 surveillance program.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-18054 [17].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Byron Unit 1 surveillance materials unambiguously.

Hence, the Byron Unit 1 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [21].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the forging.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of [1]. Byron Unit 1 has one circumferential weld that will be evaluated for credibility. This weld is Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336 and is fabricated from weld wire Heat # 442002, Linde 80 type flux, Lot # 8873. This weld metal heat is contained in both the Byron Unit 1 and the Byron Unit 2 surveillance programs. Since the weld in question utilizes data from other surveillance programs, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [22]. At this meeting the NRC presented five cases. Of the five cases, Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situations listed above for Byron Unit 1 surveillance weld metal. Note that for the forging materials, the straightforward method in [1] will be followed.

Following the NRC Case 4 guidelines, the Byron Unit 1 data will be evaluated first. Table D-1 provides the calculation of the interim CF for Byron Unit 1. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is

based on the actual surveillance weld metal measured shift values. In addition, only Byron Unit 1 data is being considered; therefore, no temperature adjustment is required.

Table D-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Byron Unit 1 Surveillance Data

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm², E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.409	0.752	28.7	21.58	0.57
	X	1.49	1.110	18.3	20.32	1.23
	W	2.26	1.221	49.5	60.42	1.49
	Y	3.97	1.355	27.8	37.66	1.83
Intermediate Shell Forging 5P-5933 (Axial)	U	0.409	0.752	18.6	13.99	0.57
	X	1.49	1.110	54.6	60.63	1.23
	W	2.26	1.221	29.5	36.01	1.49
	Y	3.97	1.355	11.7	15.85	1.83
				SUM:	266.46	10.25
	$CF_{5P\text{-}5933} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (266.46) \div (10.25) =$ **26.0°F**					
Byron Unit 1 Surveillance Weld Material (Heat # 442002)	U	0.409	0.752	5.2	3.91	0.57
	X	1.49	1.110	40.1	44.53	1.23
	W	2.26	1.221	50.6	61.77	1.49
	Y	3.97	1.355	76.7	103.90	1.83
				SUM:	214.10	5.12
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (214.10) \div (5.12) =$ **41.8°F**					

Notes:

(a) Taken from Table 5-1
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17]

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-2.

Table D-2 Byron Unit 1 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[(a)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
Intermediate Shell Forging 5P-5933 (Tangential)	U	26.0	0.409	0.752	28.7	19.6	9.1	Yes
	X		1.49	1.110	18.3	28.9	10.6	Yes
	W		2.26	1.221	49.5	31.7	17.8	No
	Y		3.97	1.355	27.8	35.2	7.4	Yes
Intermediate Shell Forging 5P-5933 (Axial)	U		0.409	0.752	18.6	19.6	1.0	Yes
	X		1.49	1.110	54.6	28.9	25.7	No
	W		2.26	1.221	29.5	31.7	2.2	Yes
	Y		3.97	1.355	11.7	35.2	23.5	No
Byron Unit 1 Surveillance Weld Material	U	41.8	0.409	0.752	5.2	31.4	26.2	Yes
	X		1.49	1.110	40.1	46.4	6.3	Yes
	W		2.26	1.221	50.6	51.0	0.4	Yes
	Y		3.97	1.355	76.7	56.6	20.1	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17].
(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

From a statistical point of view, $\pm$ 1σ would be expected to encompass 68% of the data. The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 17°F for base metal. Table D-2 indicates that three of the eight surveillance data points fall outside the $\pm$ 1σ of 17°F scatter band for surveillance base metals (62.5% within the scatter band); therefore, the forging data is deemed "non-credible" per the third criterion.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-2 indicates that all four surveillance data points (100%) fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when only the Byron Unit 1 data is considered.

Next, data from all sources for weld material with the same heat number is considered in order to evaluate the credibility of the weld metal using the NRC Case 4 guidelines. Data for the Byron Unit 1 surveillance weld material is also available from the Byron Unit 2 surveillance weld material. Since data are from multiple sources, the data may need to be adjusted for chemical and irradiation environment differences.

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. This is performed as follows:

Byron Unit 1 surveillance weld metal

Cu Wt. % = 0.02, Ni Wt. % = 0.69, Position 1.1 CF = 27°F (from Table 3-1 and Table 5-4)

Byron Unit 2 surveillance weld metal

Cu Wt. % = 0.02, Ni Wt. % = 0.71, Position 1.1 CF = 27°F (from Table 3-2 and Table 5-5)

The mean chemical composition is: Cu Wt. % = 0.02, Ni Wt. % = 0.70, Position 1.1 CF = 27°F

Since the mean chemical composition yields the same Position 1.1 CF as the Byron Units 1 and 2 surveillance weld materials, chemistry adjustments are not needed. Furthermore, as described in Section 5, since Byron Units 1 and 2 have similar operating temperatures, temperature adjustments are also not needed. Table D-3 provides the summary of the weld interim CF considering all available data.

Table D-3 Calculation of Interim Weld Chemistry Factor for the Credibility Evaluation Using all Available Surveillance Data

Material	Capsule	Capsule Fluence(a) ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Byron Unit 1 Surveillance Weld Material (Heat # 442002)	U	0.409	0.752	5.2	3.91	0.57
	X	1.49	1.110	40.1	44.53	1.23
	W	2.26	1.221	50.6	61.77	1.49
	Y	3.97	1.355	76.7	103.90	1.83
Byron Unit 2 Surveillance Weld Material (Heat # 442002)	U	0.406	0.750	8.7	6.53	0.56
	W	1.21	1.053	28.8	30.33	1.11
	X	2.18	1.211	54.2	65.66	1.47
	Y	4.19	1.366	58.7	80.19	1.87
				SUM:	392.82	10.13
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (392.82) \div (10.13) =$ **39.2°F**					

Notes:

(a) Taken from Table 5-3.
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18].

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-4.

Table D-4 Byron Unit 1 Calculated Surveillance Weld Metal Data Scatter about the Best-Fit Line using all Available Surveillance Data

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured(a) ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}(b) (°F)	<17°F (Base Metal) <28°F (Weld)
Byron Unit 1 Surveillance Weld Material	U	39.2	0.409	0.752	5.2	29.5	24.3	Yes
	X		1.49	1.110	40.1	43.5	3.4	Yes
	W		2.26	1.221	50.6	47.9	2.7	Yes
	Y		3.97	1.355	76.7	53.1	23.6	Yes
Byron Unit 2 Surveillance Weld Material	U		0.406	0.750	8.7	29.4	20.7	Yes
	W		1.21	1.053	28.8	41.3	12.5	Yes
	X		2.18	1.211	54.2	47.5	6.7	Yes
	Y		4.19	1.366	58.7	53.6	5.1	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18].

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-4 indicates that all eight surveillance data points (100%) fall within the ± 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when all available data for the Byron Unit 1 weld is considered.

Hence, Criterion 3 is met for the Byron Unit 1 surveillance program materials.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The Byron Unit 1 capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions and will not differ by more than 25°F.

Hence, Criterion 4 is met for the Byron Unit 1 surveillance program.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Byron Unit 1 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Byron Unit 1 surveillance program.

Hence, Criterion 5 is met for the Byron Unit 1 surveillance program.

Conclusion: Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The Byron Unit 1 surveillance forging data are deemed "non-credible"
- The Byron Units 1 and 2 surveillance weld data are deemed "credible"

D.2 BYRON UNIT 2 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [4], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Byron Unit 2 reactor vessel beltline region consists of the following materials:

1. Nozzle Shell Forging 4P-6107
2. Intermediate Shell Forging [49D329/49C297]-1-1
3. Lower Shell Forging [49D330/49C298]-1-1
4. Nozzle Shell Forging to Intermediate Shell Forging Circumferential Weld Seam WF-562 (Weld Wire Heat # 442011, Linde 80 Flux Type, Flux Lot # 8061)
5. Intermediate Shell Forging to Lower Shell Forging Circumferential Weld Seams WF-447 (Weld Wire Heat # 442002, Linde 80 Flux Type, Flux Lot # 8064)

The Byron Unit 2 surveillance program utilizes tangential and axial test specimens from the Lower Shell Forging. The surveillance weld metal was fabricated with weld wire Heat # 442002, Flux Type Linde 80, Lot # 8064.

At the time when the Byron Unit 2 surveillance program material was selected, it was believed that copper and phosphorus were the elements most important to the embrittlement of reactor vessel steels and the Nozzle Shell Forging was not considered a "beltline" material. The Lower Shell Forging had the same initial RT_{NDT} and a lower initial USE value than the Intermediate Shell Forging. In addition, the Lower Forging had a slightly higher copper content and approximately the same phosphorus content compared to the Intermediate Shell Forging. Based on this comparison of the beltline forging materials (Intermediate and Lower Shell Forgings), the Lower Shell Forging was chosen for the surveillance program.

Weld seam WF-447, on the other hand, was considered the only weld in the beltline region and therefore was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 442002), the same type flux (Linde 80), and the same flux lot (# 8064) as the Intermediate to Lower Shell Forging Circumferential Weld Seam.

Therefore, the materials selected for use in the Byron Unit 2 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

Based on the discussion above, Criterion 1 is met for the Byron Unit 2 surveillance program.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-18056 [18].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Byron Unit 2 surveillance materials unambiguously.

Hence, the Byron Unit 2 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [21].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the forging.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of [1]. Byron Unit 2 has one circumferential weld that will be evaluated for credibility. This weld is Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447 and is fabricated from weld wire Heat # 442002, Linde 80 type flux, Lot # 8064. This weld metal heat is contained in both the Byron Unit 1 and the Byron Unit 2 surveillance programs. Since the welds in question utilize data from other surveillance programs, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [22]. At this meeting the NRC presented five cases. Of the five cases, Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situation listed above for Byron Unit 2 surveillance weld metal. Note that for the forging materials, the straightforward method in [1] will be followed.

Following the NRC Case 4 guidelines, only the Byron Unit 2 data will be evaluated first. Table D-5 provides the calculation of the interim CF for Byron Unit 2. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is

based on the actual surveillance weld metal measured shift values. In addition, only Byron Unit 2 data is being considered; therefore, no temperature adjustment is required.

Table D-5 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Byron Unit 2 Surveillance Data

Material	Capsule	Capsule Fluence[a] ($x\ 10^{19}\ n/cm^2$, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [49D330/49C298]-1-1 (Tangential)	U	0.406	0.750	-4.8[d]	-3.60	0.56
	W	1.21	1.053	2.5	2.63	1.11
	X	2.18	1.211	14.9	18.05	1.47
	Y	4.19	1.366	44.5	60.79	1.87
Lower Shell Forging [49D330/49C298]-1-1 (Axial)	U	0.406	0.750	20.4	15.30	0.56
	W	1.21	1.053	32.1	33.81	1.11
	X	2.18	1.211	39.5	47.85	1.47
	Y	4 19	1.366	68.6	93.72	1.87
				SUM:	268.56	10.01
	$CF_{[49D330/49C298]-1-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (268.56) \div (10.01) =$ **26.8°F**					
Byron Unit 2 Surveillance Weld Material (Heat # 442002)	U	0.406	0.750	8.7	6.53	0.56
	W	1.21	1.053	28.8	30.33	1.11
	X	2.18	1.211	54.2	65.66	1.47
	Y	4.19	1.366	58.7	80.19	1.87
				SUM:	182.71	5.01
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (182.71) \div (5.01) =$ **36.5°F**					

Notes·

(a) Taken from Table 5-2

(b) FF = fluence factor = $f^{(0\ 28 - 0\ 10*\log(f))}$.

(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [18]

(d) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-6.

Table D-6 Byron Unit 2 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[a] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[b] (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Forging (Tangential)	U	26.8	0.406	0.750	-4.8[c]	20.1	24.9	No
	W		1.21	1.053	2.5	28.2	25.7	No
	X		2.18	1.211	14.9	32.5	17.6	No
	Y		4.19	1.366	44.5	36.6	7.9	Yes
Lower Shell Forging (Axial)	U		0.406	0.750	20.4	20.1	0.3	Yes
	W		1.21	1.053	32.1	28.2	3.9	Yes
	X		2.18	1.211	39.5	32.5	7.0	Yes
	Y		4.19	1.366	68.6	36.6	32.0	No
Byron Unit 2 Surveillance Weld Metal	U	36.5	0.406	0.750	8.7	27.4	18.7	Yes
	W		1.21	1.053	28.8	38.4	9.6	Yes
	X		2.18	1.211	54.2	44.2	10.0	Yes
	Y		4.19	1.366	58.7	49.9	8.8	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [18]

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

(c) Even though a reduction should not occur, using the negative measured ΔRT_{NDT} value produces the most conservative results for this credibility evaluation.

From a statistical point of view, ± 1σ would be expected to encompass 68% of the data. The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 17°F for base metal. Table D-6 indicates that four of the eight surveillance data points (50%) fall outside the ± 1σ of 17°F scatter band for surveillance base metals; <u>therefore, the forging data is deemed "non-credible" per the third criterion</u>.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-6 indicates that all four surveillance data points (100%) fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; <u>therefore, the weld material is deemed "credible" per the third criterion when only the Byron Unit 2 data is considered</u>.

Next, data from all sources is considered in order to evaluate the credibility of the weld metal using the NRC Case 4 guidelines. Data for the Byron Unit 2 surveillance weld material is also available from the Byron Unit 1 surveillance weld material. Since data are from multiple sources, the data may need to be adjusted for chemical and irradiation environment differences.

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. This is performed as follows:

Byron Unit 2 surveillance weld metal

Cu Wt. % = 0.02, Ni Wt. % = 0.71, Position 1.1 CF = 27°F (from Table 3-2 and Table 5-5)

Byron Unit 1 surveillance weld metal

Cu Wt. % = 0.02, Ni Wt. % = 0.69, Position 1.1 CF = 27°F (from Table 3-1 and Table 5-4)

The mean chemical composition is: Cu Wt. % = 0.02, Ni Wt. % = 0.70, Position 1.1 CF = 27°F

Since the mean chemical composition yields the same Position 1.1 CF as the Byron Units 1 and 2 surveillance weld materials, chemistry adjustments are not needed. Furthermore, as described in Section 5, since Byron Units 1 and 2 have similar operating temperatures temperature adjustments are also not needed. Table D-7 provides the summary of the weld interim CF considering all available data.

Table D-7 Calculation of Interim Weld Chemistry Factor for the Credibility Evaluation Using all Available Surveillance Data

Material	Capsule	Capsule Fluence[a] ($x\ 10^{19}$ n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Byron Unit 1 Surveillance Weld Material (Heat # 442002)	U	0.409	0.752	5.2	3.91	0.57
	X	1.49	1.110	40.1	44.53	1.23
	W	2.26	1.221	50.6	61.77	1.49
	Y	3.97	1.355	76.7	103.90	1.83
Byron Unit 2 Surveillance Weld Material (Heat # 442002)	U	0.406	0.750	8.7	6.53	0.56
	W	1.21	1.053	28.8	30.33	1.11
	X	2.18	1.211	54.2	65.66	1.47
	Y	4.19	1.366	58.7	80.19	1.87
				SUM:	392.82	10.13
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (392.82) \div (10.13) =$ **39.2°F**					

Notes:

(a) Taken from Table 5-3.
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18].

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-8.

Table D-8 Byron Unit 2 Calculated Surveillance Weld Metal Data Scatter about the Best-Fit Line using all Available Surveillance Data

Material	Capsule	CF (Slope $_{best-fit}$) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[(a)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
Byron Unit 1 Surveillance Weld Material	U	39.2	0.409	0.752	5.2	29.5	24.3	Yes
	X		1.49	1.110	40.1	43.5	3.4	Yes
	W		2.26	1.221	50.6	47.9	2.7	Yes
	Y		3.97	1.355	76.7	53.1	23.6	Yes
Byron Unit 2 Surveillance Weld Material	U		0.406	0.750	8.7	29.4	20.7	Yes
	W		1.21	1.053	28.8	41.3	12.5	Yes
	X		2.18	1.211	54.2	47.5	6.7	Yes
	Y		4.19	1.366	58.7	53.6	5.1	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [17] and [18]

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 28°F for weld metal. Table D-8 indicates that all eight surveillance data points (100%) fall within the ± 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion when all available data for the Byron Unit 2 weld is considered.

Hence, Criterion 3 is met for the Byron Unit 2 surveillance program materials.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The Byron Unit 2 capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions and will not differ by more than 25°F.

Hence, Criterion 4 is met for the Byron Unit 2 surveillance program.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Byron Unit 2 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Byron Unit 2 surveillance program.

<u>Hence, Criterion 5 is met for the Byron Unit 2 surveillance program.</u>

Conclusion: Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The Byron Unit 2 surveillance <u>forging</u> data are deemed "<u>non-credible</u>"
- The Byron Units 1 and 2 surveillance <u>weld</u> data are deemed "<u>credible</u>"

APPENDIX E PRESSURIZED THERMAL SHOCK EVALUATION

Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the reactor pressure vessel (RPV) under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [23]) that established screening criteria on Pressurized Water Reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. The RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 [1].

These accepted methods were used with the clad/base metal interface fluence values of Section 2 to calculate the following RT_{PTS} values for the Byron Units 1 and 2 RPV materials at 57 EFPY (EOLE). The EOLE RT_{PTS} calculations are summarized in Table E-1 and Table E-2.

PTS Conclusion

All of the beltline and extended beltline materials in the Byron Units 1 and 2 reactor vessels are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds at 57 EFPY. Therefore, all materials are acceptable.

Table E-1 RT_{PTS} Calculations for the Byron Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Heat Number	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials										
Nozzle Shell Forging	123J218	31	1.15	1.039	30	32.2	0	16.1	32.2	94.4
Intermediate Shell Forging	5P-5933	26	3.19	1.305	40	33.9	0	17.0	33.9	107.9
Intermediate Shell Forging Using Non-credible Byron Unit 1 Surveillance Data	5P-5933	26.0	3.19	1.305	40	33.9	0	17.0	33.9	107.9
Lower Shell Forging	5P-5951	26	3.18	1.304	10	33.9	0	17.0	33.9	77.8
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-501	442011	41	1.15	1.039	10	42.6	0	21.3	42.6	95.2
Nozzle to Intermediate Shell Forging Circ. Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	442011	31.2	1.15	1.039	10	32.4	0	14.0	28.0	70.4
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-336	442002	54	3.07	1.296	-30	70.0	0	28.0	56.0	96.0
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Byron Units 1 and 2 Surveillance Data	442002	78.3	3.07	1.296	-30	101.5	0	14.0	28.0	99.5
Reactor Vessel Extended Beltline Materials										
Inlet Nozzle 03-001	1V4684-3V1320	86	0.0133	0.133	-10	11.4	0	5.7	11.4	12.8
Inlet Nozzle 03-002	1V4684-3V1320	86	0.0133	0.133	-20	11.4	0	5.7	11.4	2.8
Inlet Nozzle 04-001	1V4695	95.8	0.0133	0.133	-20	12.7	0	6.4	12.7	5.4
Inlet Nozzle 04-002	1V4695	85.7	0.0133	0.133	-20	11.4	0	5.7	11.4	2.7
Outlet Nozzle 01-001	1V4656	77	0.0101	0.110	0	8.5	0	4.3	8.5	17.0
Outlet Nozzle 01-002	1V4656	77	0.0101	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-001	2V2557	77	0.0101	0.110	-20	8.5	0	4.3	8.5	-3.0
Outlet Nozzle 02-002	2V2557	77	0.0101	0.110	-10	8.5	0	4.3	8.5	7.0

Table E-1 RT_{PTS} Calculations for the Byron Unit 1 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Heat Number	CF(a) (°F)	Surface Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ(d) (°F)	Margin (°F)	RT_{PTS} (°F)
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-337	442002	139.2	0.0133	0.133	-10	18.5	0	9.2	18.5	26.9
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-419	1P5412	168.3	0.0101	0.110	-48.6	18.6	18.0(f)	28.0(e)	66.6	36.6
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-406	504	73.6	0.0101	0.110	10	8.1	0	4.1	8.1	26.2

Notes:

(a) Values are taken from Table 5-4.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix D, the intermediate shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of 10 CFR 50.61 [23], the base metal σ_Δ = 17°F when surveillance data is non-credible or not used to determine the CF, and σ_Δ = 8.5°F when credible surveillance data is used to determine the CF. The weld metal σ_Δ = 28°F when surveillance data is non-credible or not used, and σ_Δ = 14°F when credible surveillance data is used. However, σ_Δ need not exceed 0 5*ΔRT_{NDT} per [23].

(e) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.

(f) Table 9 of [6] Revision 2-A identifies σ_I = 18.0°F associated with the use of the generic $RT_{NDT(u)}$ value

Table E-2 RT_{PTS} Calculations for the Byron Unit 2 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Material ID/Heat Number	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials										
Nozzle Shell Forging	4P-6107	31	1.01	1.003	10	31.1	0	15.5	31.1	72.2
Intermediate Shell Forging	[49D329/49C297]-1-1	20	2.96	1.288	-20	25.8	0	12.9	25.8	31.5
Lower Shell Forging	[49D330/49C298]-1-1	37	2.90	1.283	-20	47.5	0	17.0	34.0	61.5
Lower Shell Forging Using Non-Credible Byron Unit 2 Surveillance Data	[49D330/49C298]-1-1	27.2	2.90	1.283	-20	34.9	0	17.0	34.0	48.9
Nozzle to Intermediate Shell Forging Circumferential Weld Seam WF-562	442011	41	1.05	1.014	40	41.6	0	20.8	41.6	123.1
Nozzle to Intermediate Shell Forging Circumferential Weld Seam Using Credible Braidwood Units 1 and 2 Surveillance Data	442011	31.2	1.05	1.014	40	31.6	0	14.0	28.0	99.6
Intermediate to Lower Shell Forging Circumferential Weld Seam WF-447	442002	54	2.82	1.276	10	68.9	0	28.0	56.0	134.9
Intermediate to Lower Shell Forging Circumferential Weld Seam Using Credible Byron Units 1 and 2 Surveillance Data	442002	78.3	2.82	1.276	10	99.9	0	14.0	28.0	137.9
Reactor Vessel Extended Beltline Materials										
Inlet Nozzle 01-001	51-2979	44	0.0129	0.130	-10	5.7	0	2.9	5.7	1.4
Inlet Nozzle 01-002	51-2979	44	0.0129	0.130	-20	5.7	0	2.9	5.7	-8.6
Inlet Nozzle 02-001	42-5105	44	0.0129	0.130	0	5.7	0	2.9	5.7	11.4
Inlet Nozzle 02-002	42-5105	44	0.0129	0.130	0	5.7	0	2.9	5.7	11.4
Outlet Nozzle 01-001	11-5052	58	0.00968	0.107	-10	6.2	0	3.1	6.2	2.4
Outlet Nozzle 01-002	11-5052	51	0.00968	0.107	-10	5.5	0	2.7	5.5	0.9
Outlet Nozzle 02-001	4-2953	58	0.00968	0.107	-20	6.2	0	3.1	6.2	-7.6
Outlet Nozzle 02-002	4-2956	58	0.00968	0.107	-10	6.2	0	3.1	6.2	2.4
Inlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403	167[e]	0.0129	0.130	-48.6	21.7	18.0[g]	28.0[f]	66.6	39.7

Table E-2 RT_{PTS} Calculations for the Byron Unit 2 Reactor Vessel Materials at 57 EFPY

Reactor Vessel Material	Material ID/Heat Number	CF[a] (°F)	Surface Fluence[b] (x 10^{19} n/cm², E > 1.0 MeV)	Surface FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[d] (°F)	Margin (°F)	RT_{PTS} (°F)
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-545	442010	172	0.00968	0.107	-48.6	18.4	18.0[g]	28.0[f]	66.6	36.4
Outlet Nozzle to Nozzle Shell Forging Circumferential Weld Seams WF-559	41403	167[e]	0.00968	0.107	-48.6	17.9	18.0[g]	28.0[f]	66.6	35.9

Notes:

(a) Values are taken from Table 5-5.
(b) Values are taken from Table 7-2.
(c) Values are taken from Table 3-2.
(d) Per Appendix D, the lower shell forging material surveillance data was determined to be non-credible, but the Byron Units 1 and 2 surveillance weld data was determined to be credible. Per [46], the Braidwood Units 1 and 2 surveillance weld data was determined to be credible. Therefore, per the guidance of 10 CFR 50.61 [23], the base metal σ_Δ = 17°F when surveillance data is non-credible or not used to determine the CF, and σ_Δ = 8.5°F when credible surveillance data is used to determine the CF. The weld metal σ_Δ = 28°F when surveillance data is non-credible or not used, and σ_Δ = 14°F when credible surveillance data is used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per [23].
(e) Value is required minimum per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.
(f) Value is required per condition from [6]. This condition must be met in order to use values from Table 9 of [6] Revision 2-A.
(g) Table 9 of [6] Revision 2-A identifies σ_I = 18 0°F associated with the use of the generic $RT_{NDT(u)}$ value.

APPENDIX F VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

F.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for all surveillance capsules withdrawn from service to date at Byron Unit 1 and Unit 2 are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [9]. One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ±20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 2 of this report.

F.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the four in-vessel neutron sensor sets and two sets of Ex-Vessel Neutron Dosimetry (EVND) dosimetry sets analyzed to date as part of the Byron Unit 1 and Unit 2 Reactor Vessels Materials Surveillance Program are presented.

Six irradiation capsules attached to the neutron pads were included in the Byron Unit 1 reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 58.5°, 121.5°, 238.5°, and 301.5° (31.5° from the core cardinal axis) and 61° and 241° (29° from the core cardinal axis). The irradiation history of each of these six in-vessel surveillance capsules is summarized as follows:

Unit 1		
Capsule	**Location**	**Irradiation History**
U	31.5° Dual	Cycle 1 (withdrawn for analysis)
X	31.5° Dual	Cycles 1–5 (withdrawn for analysis)
W	31.5° Single	Cycles 1–8 (withdrawn for analysis)
V	29° Dual	Cycles 1–12 (withdrawn for storage)
Z	31.5° Single	Cycles 1–12 (withdrawn for storage)
Y	29° Dual	Cycles 1–15 (withdrawn for analysis)

In addition to the four previously analyzed in-vessel surveillance capsules, two sets of EVND dosimetry sensors have been analyzed after withdrawal from Byron Unit 1. The location and time of irradiation for EVND sensor sets analyzed at Byron Unit 1 are summarized as follows:

Unit 1				
Capsule ID	**Azimuthal Location from Cardinal Axis**	**Gradient Chain ID Tag on which EVND Capsule is Located**	**Axial Elevation**	**Cycles of Irradiation**
A	0.5°	1S-1 00	Active Core Midplane	16
B	14.5°	1S-1 15	Active Core Midplane	16
C	29.5°	1S-1 30	Active Core Midplane	16
E	44.5°	1S-1 45	Active Core Midplane	16
G	0.5°	1S-2 00	Active Core Midplane	17–21
H	14.5°	1S-2 15	Active Core Midplane	17–21
I	29.5°	1S-2 30	Active Core Midplane	17–21
K	44.5°	1S-2 45	Active Core Midplane	17–21
D	44.5°	1S-1 45	Active Core Top	16
F	44.5°	1S-1 45	Active Core Bottom	16
J	44.5°	1S-2 45	Active Core Top	17–21
L	44.5°	1S-2 45	Active Core Bottom	17–21

Similarly, six irradiation capsules attached to the neutron pads were included in the Byron Unit 2 reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 58.5°, 121.5°, 238.5°, 301.5° (31.5° from the core cardinal axis), 61.0° and 241.0° (29.0° from the core cardinal axis). The irradiation history of each of these six in-vessel surveillance capsules is summarized as follows:

Unit 2		
Capsule	**Location**	**Irradiation History**
U	31.5° Dual	Cycle 1 (withdrawn for analysis)
W	31.5° Single	Cycles 1–4 (withdrawn for analysis)
X	31.5° Dual	Cycles 1–7 (withdrawn for analysis)
V	29° Dual	Cycles 1–11 (withdrawn for storage)
Z	31.5° Single	Cycles 1–11 (withdrawn for storage)
Y	29° Dual	Cycles 1–15 (withdrawn for analysis)

In addition to the four previously analyzed in-vessel surveillance capsules, two sets of EVND dosimetry sensors have been analyzed after withdrawal from Byron Unit 2. The location and time of irradiation for EVND sensor sets analyzed at Byron Unit 2 are summarized as follows:

Unit 2				
Capsule ID	**Azimuthal Location from Cardinal Axis**	**Gradient Chain ID Tag on which EVND Capsule is Located**	**Axial Elevation**	**Cycles of Irradiation**
A	0.5°	2S-1 00	Active Core Midplane	15
B	14.5°	2S-1 15	Active Core Midplane	15
C	29.5°	2S-1 30	Active Core Midplane	15
E	44.5°	2S-1 45	Active Core Midplane	15
G	0.5°	2S-2 00	Active Core Midplane	16–20
H	14.5°	2S-2 15	Active Core Midplane	16–20
I	29.5°	2S-2 30	Active Core Midplane	16–20
K	44.5°	2S-2 45	Active Core Midplane	16–20
D	44.5°	2S-1 45	Active Core Top	15
F	44.5°	2S-1 45	Active Core Bottom	15
J	44.5°	2S-2 45	Active Core Top	16–20
L	44.5°	2S-2 45	Active Core Bottom	16–20

The azimuthal locations included in the above tabulation represent the first octant equivalent azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of Byron Unit 1 and Unit 2 Surveillance Capsules U, X, W, and Y are summarized as follows:

Sensor Material	Reaction Of Interest	Capsule U	Capsule X	Capsule W	Capsule Y
Copper	$^{63}Cu(n,\alpha)^{60}Co$	X	X	X	X
Iron	$^{54}Fe(n,p)^{54}Mn$	X	X	X	X
Nickel	$^{58}Ni(n,p)^{58}Co$	X	X	X	
Uranium-238	$^{238}U(n,f)^{137}Cs$	X	X	X	X
Neptunium-237	$^{237}Np(n,f)\ ^{137}Cs$	X	X	X	X
Cobalt-Aluminum*	$^{59}Co(n,\gamma)^{60}Co$	X	X	X	X
*The cobalt-aluminum measurements include both bare wire and cadmium-covered sensors.					

Since all of the dosimetry monitors were located at the radial center of the material test specimen array, radial gradient corrections were not required for these reaction rates. Pertinent physical and nuclear characteristics of the in-vessel surveillance capsule passive neutron sensors are listed in Table F-1. The pertinent physical and nuclear characteristics of the EVND capsule passive neutron sensors are listed in Table F-9.

The use of passive monitors such as those listed above does not yield a direct measure of the energy-dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- The physical characteristics of each monitor,
- The operating history of the reactor,
- The energy response of each monitor, and
- The neutron energy spectrum at the monitor location.

Results from the radiometric counting of the neutron sensors from Byron Unit 1 Capsules U, X, W, and Y are documented in [28, 29, 30, and 17], respectively, and re-evaluated in [7]. Results from the radiometric counting of the neutron sensors from Byron Unit 2 Capsules U, W, X, and Y are documented in [31, 32, 33, and 18], respectively, and re-evaluated in [8]. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by Capsules U, X, W, and Y was based on the monthly power generation of each unit from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The startup and shutdown dates for each cycle at Byron Unit 1 are given in Table F-2. Similarly, the startup and shutdown dates for each cycle at Byron Unit 2 are given in Table F-3.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum \frac{P_j}{P_{ref}} C_j \left[1 - e^{-\lambda t_j}\right] \left[e^{-\lambda t_{d,j}}\right]}$$

where:

R = Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).

A = Measured specific activity (dps/g).

N_0	=	Number of target element atoms per gram of sensor.
F	=	Atom fraction of the target isotope in the target element.
Y	=	Number of product atoms produced per reaction.
P_j	=	Average core power level during irradiation period j (MW).
P_{ref}	=	Maximum or reference power level of the reactor (MW).
C_j	=	Calculated ratio of $\phi(E > 1.0\ \text{MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0\ \text{MeV})$ over the entire irradiation period.
λ	=	Decay constant of the product isotope (1/sec).
t_j	=	Length of irradiation period j (sec).
t_{d_j}	=	Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology discussed in Section 2, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel-cycle-specific neutron flux values are used to compute values for cycle-dependent C_j values at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets at both Byron Unit 1 and Unit 2, additional corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U and ^{237}Np sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the Byron Unit 1 fission sensor reaction rates are summarized as follows:

Unit 1				
Correction	Capsule U	Capsule X	Capsule W	Capsule Y
^{235}U Impurity/Pu Build-in	0.868	0.826	0.801	0.744
^{238}U(γ,f)	0.966	0 967	0.970	0.968
Net ^{238}U Correction	0.839	0.799	0.777	0.720
^{237}Np(γ,f)	0.990	0.990	0.991	0.991

Similarly, the correction factors applied to the Byron Unit 2 fission sensor reaction rates are summarized as follows:

Unit 2				
Correction	Capsule U	Capsule W	Capsule X	Capsule Y
^{235}U Impurity/Pu Build-in	0.868	0.837	0.803	0.736
^{238}U(γ,f)	0.966	0.970	0.967	0.968
Net ^{238}U Correction	0.839	0.812	0.777	0.712
^{237}Np(γ,f)	0.990	0.991	0.990	0.991

These factors were applied in a multiplicative fashion to the decay corrected uranium and neptunium fission sensor reaction rates.

Results of the sensor reaction rate determinations for Byron Unit 1 Capsules U, X, W, and Y are given in Table F-4 through Table F-7. Results of the sensor reaction rate determinations for Byron Unit 2 Capsules U, W, X, and Y are given in Table F-22 through Table F-25. In Table F-4 through Table F-7 and Table F-22 through Table F-25, the measured specific activities, decay-corrected saturated specific activities, and computed reaction rates for each sensor indexed to the radial center of the capsule are listed. The fission sensor reaction rates are listed both with and without the applied corrections for ^{238}U impurities, plutonium build-in, and gamma-ray-induced fission effects.

Table F-8 lists the EVND capsule designations for the Byron Unit 1 and Unit 2 EVND capsules. Similarly, the nuclear parameters listed in Table F-9 are used to evaluate the EVND capsule neutron sensors irradiated in Cycle 16 and Cycles 17-21 for Unit 1 and Cycle 15 and Cycles 16-20 for Unit 2. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycle 16 at Byron Unit 1 are reported in Table F-10 through Table F-15. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycles 17 through 21 at Byron Unit 1 are reported in Table F-16 through Table F-21. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycle 15 at Byron Unit 2 are reported in Table F-26 through Table F-31. The pertinent measured and calculated data for the ex-vessel dosimetry irradiated in Cycles 16 through 20 at Byron Unit 2 are reported in Table F-32 through Table F-37.

F.1.2 Least-Squares Evaluation of Sensor Sets

Least-squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations resulting in a Best-Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as $\phi(E > 1.0\text{ MeV})$ or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\varphi_g \pm \delta_{\varphi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the Byron Unit 1 and Unit 2 surveillance capsule dosimetry, the FERRET Code [34] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters ($\phi(E > 1.0\text{ MeV})$ and dpa) along with associated uncertainties for the five in-vessel capsules analyzed to date.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Byron Unit 1 and Unit 2 application, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section F.1.1. The dosimetry reaction cross sections and uncertainties were obtained from the Sandia National Laboratories Radiation Metrology Laboratory (SNLRML) dosimetry cross-section library [35]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB)" [36].

The uncertainties associated with the measured reaction rates, dosimetry cross sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The

assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" [37].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the Byron Unit 1 and Unit 2 surveillance capsule sensor sets.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	**Uncertainty**
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{46}Ti (n,p) ^{46}Sc$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	10%
$^{93}Nb (n,n') ^{93m}Nb$	5% - 10%
$^{237}Np(n,f)^{137}Cs$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations, and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the Byron Unit 1 and Unit 2 surveillance programs, the following uncertainties in the fission spectrum averaged cross sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08-4.16%
$^{46}Ti(n,p)^{46}Sc$	4.50-4.87%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{238}U(n,f)^{137}Cs$	0.54-0.64%
$^{93}Nb\ (n,n')\ ^{93m}Nb$	6.96-7.23%
$^{237}Np(n,f)^{137}Cs$	10.32-10.97%
$^{59}Co(n,\gamma)^{60}Co$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectra input to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta\, e^{-H}$$

where

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Byron Unit 1 and Unit 2 calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

F.1.3 Comparisons of Measurements and Calculations

Byron Unit 1 Neutron Dosimetry Benchmark

In this section, comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations are presented for Byron Unit 1. These comparisons are provided on two levels. In the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second level, calculated values of neutron exposure rates in terms of fast neutron fluence rate ϕ (E > 1.0 MeV) and iron atom displacements are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table F-38, comparisons of measured-to-calculation (M/C) ratios are listed for the threshold sensors contained in the in-vessel Capsules U, X, W, and Y. From Table F-38, it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.91 to 1.07 with an overall average of 0.98 and an associated standard deviation of 11.9%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table F-39, similar comparisons are provided for the sensor sets withdrawn from the midplane axial elevation measurement locations in the reactor cavity. From Table F-39, it is noted that for the individual threshold foils, the average M/C ratio ranges from 0.85 to 1.03 with an overall average of 0.91 and an associated standard deviation of 6.5%. From Table F-40, it is also noted that the M/C ratios for the EVND Set 1S-1 and 1S-2 capsules are in good agreement. Finally, as in the case of the in-vessel comparisons, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

Comparisons of the M/C ratios for the two sensor sets withdrawn from the off-midplane measurement locations in the reactor cavity are provided in Table F-40.

In Table F-41 and Table F-42, best-estimate to calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set are provided for the in-vessel and midplane ex-vessel irradiations, respectively. For the in-vessel capsules the average BE/C ratio is seen to be 0.95 with an associated uncertainty of 11.3% for neutron fluence rate (E > 1.0 MeV) and 0.96 with an associated uncertainty of 10.6% for the iron atom displacement rate. The corresponding average BE/C ratio from the midplane ex-vessel irradiations is 0.96 with an uncertainty of 6.4% for neutron fluence rate (E > 1.0 MeV) and 1.00 with an uncertainty of 5.0% for the iron atom displacement rate.

The M/C data sets listed in Table F-38 and Table F-39, as well as the BE/C data sets given in Table F-41 and Table F-42, provide a validation of the plant-specific neutron transport calculations described in Section 2 of this report. Each of these data comparisons shows that for both in-vessel and midplane ex-vessel locations the measurements and calculations agree well within the 20% criterion specified in Regulatory Guide 1.190. In fact, both the average M/C results and BE/C results fall within the 13% (1σ) uncertainty assigned to the fast neutron exposure of the absolute transport calculations.

The measurements to calculation comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	Unit 1 In-Vessel		Unit 1 Ex-Vessel Midplane		Unit 1 Combined	
	Avg. M/C	% Unc. (1σ)	Avg. M/C	% Unc. (1σ)	Avg. M/C	% Unc. (1σ)
$^{63}Cu(n,\alpha)$	1.07	2.3%	0.89	6.6%	0.98	4.6%
$^{46}Ti(n,p)$	-	-	0.88	7.3%	-	-
$^{54}Fe(n,p)$	0.94	5.6%	0.92	7.2%	0.93	6.4%
$^{58}Ni(n,p)$	0.91	11.9%	0.85	7.6%	0.88	10.1%
$^{93}Nb(n,n')$	-	-	1.03	5.4%	-	-
$^{238}U(Cd)$ (n,f)	1.06	14.2%	-	-	-	-
$^{237}Np(Cd)(n,f)$	0.93	14.8%	-	-	-	-
Linear Average	0.98	11.9%	0.91	9.6%	0.95	7.7%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) are summarized as follows:

Parameter	Unit 1 In-Vessel		Unit 1 Ex-Vessel Midplane		Unit 1 Combined	
	Avg. BE/C	% Unc. (1σ)	Avg. BE/C	% Unc. (1σ)	Avg. BE/C	% Unc. (1σ)
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.95	11.3%	0.96	6.4%	0.95	6.5%
dpa/s	0.96	10.6%	1.00	5.0%	0.98	5.8%

These data comparisons show similar and consistent results with the linear average M/C ratio of 0.95 in good agreement with the resultant least-squares BE/C ratios of 0.95 for neutron fluence rate (E > 1.0 MeV) and 0.98 for iron atom displacement rate. The comparisons demonstrate that the calculated results for Byron Unit 1 provided in Section 2 of this report are validated within the context of the assigned 13% uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by Reference 9 is easily met.

Byron Unit 2 Neutron Dosimetry Benchmark

In this section, comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations are presented for Byron Unit 2. These comparisons are provided on two levels. In the first instance, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second instance, calculated values of neutron exposure rates in terms of neutron fluence (E > 1.0 MeV) and iron atom displacements are compared with the best estimate exposure rates obtained from the least-squares evaluation.

In Table F-43, comparisons of M/C ratios are listed for the threshold sensors contained in in-vessel Capsules U, W, X, and Y. From Table F-43 it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.95 to 1.14 with an overall average of 1.03 and an associated standard deviation of 8.3%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table F-44, similar comparisons are provided for the sensor sets withdrawn from the midplane axial elevation measurement locations in the reactor cavity. From Table F-44, it is noted that for the individual threshold foils, the average M/C ratio ranges from 0.86 to 1.06 with an overall average of 0.92 and an associated standard deviation of 10.3%. From Table F-44, it is also noted that the M/C ratios for the EVND Set 2S-1 and 2S-2 capsules are in good agreement. Finally, as in the case of the in-vessel comparisons, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

Comparisons of the M/C ratios for the two sensor sets withdrawn from the off-midplane measurement locations in the reactor cavity are provided in Table F-45.

In Table F-46 and Table F-47, best-estimate to calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set are provided for the in-vessel and midplane ex-vessel irradiations, respectively. For the in-vessel capsules the average BE/C ratio is seen to be 0.99 with an associated uncertainty of 4.3% for neutron fluence rate (E > 1.0 MeV) and 0.99 with an associated uncertainty of 3.0% for the iron atom displacement rate. The corresponding average BE/C ratio from the midplane ex-vessel irradiations is 0.95 with an uncertainty of 5.5% for neutron fluence rate (E > 1.0 MeV) and 0.97 with an uncertainty of 4.6% for the iron atom displacement rate.

The M/C data sets listed in Table F-43 and Table F-44, as well as the BE/C data sets given in Table F-46 and Table F-47, provide a validation of the plant-specific neutron transport calculations described in Section 2 of this report. Each of these data comparisons shows that for both in-vessel and midplane ex-vessel locations the measurements and calculations agree well within the 20% criterion specified in Regulatory Guide 1.190. In fact, both the average M/C results and BE/C results fall within the 13% (1σ) uncertainty assigned to the fast neutron exposure of the absolute transport calculations.

The measurements to calculation comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	Unit 2 In-Vessel		Unit 2 Midplane Ex-Vessel		Unit 2 Combined	
	Average	% Standard Deviation	Average	% Standard Deviation	Average	% Standard Deviation
	M/C	(1σ)	M/C	(1σ)	M/C	(1σ)
^{63}Cu(n,α)	1.14	2.3%	0.89	8.4%	1.02	3.9%
^{46}Ti(n,p)	-	-	0.89	7.9%	-	-
^{54}Fe(n,p)	0.98	0.8%	0.92	6.8%	0.95	3.3%
^{58}Ni(n,p)	0.95	5.8%	0.86	7.9%	0.91	4.7%
^{93}Nb(n,n′)	-	-	1.06	5.6%	-	-
^{238}U(Cd) (n,f)	1.08	8.5%	-	-	-	-
^{237}Np(Cd)(n,f)	1.00	6.2%	-	-	-	-
Linear Average	1.03	8.3%	0.92	10.3%	0.98	6.5%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) is summarized as follows:

Parameter	Unit 2 In-Vessel		Unit 2 Midplane Ex-Vessel		Unit 2 Combined	
	Average	% Standard Deviation	Average	% Standard Deviation	Average	% Standard Deviation
	BE/C	(1σ)	BE/C	(1σ)	BE/C	(1σ)
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.99	4.3%	0.95	5.5%	0.97	3.5%
dpa/s	0.99	3.0%	0.97	4.6%	0.98	2.7%

These data comparisons show similar and consistent results with the linear average M/C ratio of 0.98 in excellent agreement with the resultant least-squares BE/C ratios of 0.97 and 0.98 for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, respectively. The comparisons demonstrate that the calculated results for Byron Unit 2 provided in Section 2 of this report are validated within the context of the assigned 13% calculational uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by Reference 9 is easily met.

Table F-1 Nuclear Parameters Used in the Evaluation of Byron Unit 1 and Unit 2 In-Vessel Surveillance Capsule Neutron Sensors

Monitor Material	Reaction of Interest	Target Atom Fraction	90% Response Range (MeV)[a]	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	0.6917	4.9 – 11.9	5.272 y	
Iron	^{54}Fe (n,p)	0.0585	2.1 – 8.5	312.1 d	
Nickel	^{58}Ni (n,p)	0.6808	1.5 – 8.3	70.82 d	
Uranium-238	^{238}U (n,f)	1.0000	1.3 – 6.9	30.07 y	6.02
Neptunium-237	^{237}Np (n,f)	1.0000	0.3 – 3.8	30.07 y	6.17 – 6.27
Cobalt-Aluminum	^{59}Co (n,γ)	0.0015	non-threshold	5.272 y	

Note:

(a) The 90% response range is defined such that, in the neutron spectrum characteristic of the Byron Unit 1 and Unit 2 surveillance capsules, approximately 90% of the sensor response is due to neutrons in the energy range specified with approximately 5% of the total response due to neutrons with energies below the lower limit and 5% of the total response due to neutrons with energies above the upper limit

Table F-2 Byron Unit 1 Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	March 1, 1985	February 14, 1987
2	June 3, 1987	September 3, 1988
3	November 9, 1988	January 5, 1990
4	March 4, 1990	September 6, 1991
5	November 7, 1991	February 5, 1993
6	April 10, 1993	September 8, 1994
7	November 2, 1994	April 5, 1996
8	July 3, 1996	November 7, 1997
9	March 9, 1998	March 27, 1999
10	April 25, 1999	September 23, 2000
11	October 13, 2000	March 12, 2002
12	March 30, 2002	September 22, 2003
13	October 14, 2003	February 27, 2005
14	March 25, 2005	September 10, 2006
15	October 16, 2006	March 23, 2008
16	April 14, 2008	September 14, 2009
17	October 8, 2009	March 13, 2011
18	April 24, 2011	September 10, 2012
19	October 7, 2012	March 10, 2014
20	March 27, 2014	September 14, 2015
21	October 1, 2015	March 6, 2017

Table F-3 Byron Unit 2 Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	February 6, 1987	January 7, 1989
2	March 6, 1989	September 1, 1990
3	November 21, 1990	February 28, 1992
4	April 30, 1992	September 3, 1993
5	October 25, 1993	February 10, 1995
6	March 25, 1995	August 8, 1996
7	October 5, 1996	April 11, 1998
8	May 18, 1998	October 23, 1999
9	November 16, 1999	April 7, 2001
10	April 22, 2001	September 16, 2002
11	October 7, 2002	March 22, 2004
12	April 9, 2004	September 25, 2005
13	October 12, 2005	April 1, 2007
14	May 3, 2007	October 5, 2008
15	October 24, 2008	April 18, 2010
16	May 8, 2010	September 18, 2011
17	October 10, 2011	April 8, 2013
18	May 1, 2013	September 29, 2014
19	October 23, 2014	April 18, 2016
20	May 17, 2016	October 2, 2017

Table F-4 Measured Sensor Activities and Reaction Rates for Byron Unit 1 Surveillance Capsule U

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
87-1686	^{63}Cu (n,α) ^{60}Co	5.37E+04	4.14E+05	6.32E-17	5.95E-17	5.95E-17
87-1692	^{63}Cu (n,α) ^{60}Co	4.87E+04	3.76E+05	5.73E-17		
87-1698	^{63}Cu (n,α) ^{60}Co	4.93E+04	3.80E+05	5.80E-17		
87-1687	^{54}Fe (n,p) ^{54}Mn	1.46E+06	3.77E+06	5.99E-15	5.70E-15	5.70E-15
87-1693	^{54}Fe (n,p) ^{54}Mn	1.35E+06	3.49E+06	5.54E-15		
87-1699	^{54}Fe (n,p) ^{54}Mn	1.36E+06	3.52E+06	5.58E-15		
87-1688	^{58}Ni (n,p) ^{58}Co	1.04E+07	5.39E+07	7.72E-15	7.30E-15	7.30E-15
87-1694	^{58}Ni (n,p) ^{58}Co	9.55E+06	4.95E+07	7.09E-15		
87-1700	^{58}Ni (n,p) ^{58}Co	9.55E+06	4.95E+07	7.09E-15		
87-1683	^{59}Co (n,γ) ^{60}Co	1.07E+07	8.25E+07	5.38E-12	5.03E-12	5.03E-12
87-1684	^{59}Co (n,γ) ^{60}Co	9.61E+06	7.41E+07	4.84E-12		
87-1689	^{59}Co (n,γ) ^{60}Co	1.08E+07	8.33E+07	5.43E-12		
87-1690	^{59}Co (n,γ) ^{60}Co	8.83E+06	6.81E+07	4.44E-12		
87-1695	^{59}Co (n,γ) ^{60}Co	9.40E+06	7.25E+07	4.73E-12		
87-1696	^{59}Co (n,γ) ^{60}Co	1.06E+07	8.18E+07	5.33E-12		
87-1685	^{59}Co(Cd) (n,γ) ^{60}Co	5.16E+06	3.98E+07	2.60E-12	2.64E-12	2.64E-12
87-1691	^{59}Co(Cd) (n,γ) ^{60}Co	5.35E+06	4.13E+07	2.69E-12		
87-1697	^{59}Co(Cd) (n,γ) ^{60}Co	5.20E+06	4.01E+07	2.62E-12		
87-1681	^{238}U (n,f) ^{137}Cs	1.50E+05	5.82E+06	3.82E-14	3.82E-14	3.20E-14
87-1682	^{237}Np (n,f) ^{137}Cs	1.30E+06	5.04E+07	3.22E-13	3.22E-13	3.19E-13

Note:

(a) Measured activity decay corrected to July 1, 1987

Table F-5 Measured Sensor Activities and Reaction Rates for Byron Unit 1 Surveillance Capsule X

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
93-3125	^{63}Cu (n,α) ^{60}Co	1.41E+05	3.22E+05	4.92E-17	4.60E-17	4.60E-17
93-3131	^{63}Cu (n,α) ^{60}Co	1.29E+05	2.95E+05	4.50E-17		
93-3137	^{63}Cu (n,α) ^{60}Co	1.26E+05	2.88E+05	4.39E-17		
93-3127	^{54}Fe (n,p) ^{54}Mn	1.49E+06	2.93E+06	4.65E-15	4.39E-15	4.39E-15
93-3133	^{54}Fe (n,p) ^{54}Mn	1.37E+06	2.70E+06	4.28E-15		
93-3139	^{54}Fe (n,p) ^{54}Mn	1.36E+06	2.68E+06	4.25E-15		
93-3126	^{58}Ni (n,p) ^{58}Co	6.01E+06	4.43E+07	6.35E-15	6.04E-15	6.04E-15
93-3132	^{58}Ni (n,p) ^{58}Co	5.60E+06	4.13E+07	5.92E-15		
93-3138	^{58}Ni (n,p) ^{58}Co	5.55E+06	4.10E+07	5.86E-15		
93-3122	^{59}Co (n,γ) ^{60}Co	2.56E+07	5.85E+07	3.82E-12	3.48E-12	3.48E-12
93-3123	^{59}Co (n,γ) ^{60}Co	2.33E+07	5.33E+07	3.48E-12		
93-3128	^{59}Co (n,γ) ^{60}Co	2.10E+07	4.80E+07	3.13E-12		
93-3129	^{59}Co (n,γ) ^{60}Co	2.50E+07	5.72E+07	3.73E-12		
93-3134	^{59}Co (n,γ) ^{60}Co	2.42E+07	5.53E+07	3.61E-12		
93-3135	^{59}Co (n,γ) ^{60}Co	2.08E+07	4.76E+07	3.10E-12		
93-3124	^{59}Co(Cd) (n,γ) ^{60}Co	1.27E+07	2.90E+07	1.89E-12	1.89E-12	1.89E-12
93-3130	^{59}Co(Cd) (n,γ) ^{60}Co	1.27E+07	2.90E+07	1.89E-12		
93-3136	^{59}Co(Cd) (n,γ) ^{60}Co	1.27E+07	2.90E+07	1.89E-12		
93-3120	^{238}U (n,f) ^{137}Cs	5.52E+05	4.69E+06	3.08E-14	3.08E-14	2.46E-14
93-3121	^{237}Np (n,f) ^{137}Cs	3.54E+06	3.01E+07	1.92E-13	1.92E-13	1.90E-13

Note:

(a) Measured activity decay corrected to August 8, 1993.

Table F-6 Measured Sensor Activities and Reaction Rates for Byron Unit 1 Surveillance Capsule W

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
9805-280	^{63}Cu (n,α) ^{60}Co	1.75E+05	3.27E+05	4.99E-17	4.64E-17	4.64E-17
9805-286	^{63}Cu (n,α) ^{60}Co	1.58E+05	2.95E+05	4.50E-17		
9805-291	^{63}Cu (n,α) ^{60}Co	1.55E+05	2.90E+05	4.42E-17		
9805-281	^{54}Fe (n,p) ^{54}Mn	1.31E+06	3.21E+06	5.09E-15	4.67E-15	4.67E-15
9805-288	^{54}Fe (n,p) ^{54}Mn	1.16E+06	2.84E+06	4.51E-15		
9805-293	^{54}Fe (n,p) ^{54}Mn	1.14E+06	2.79E+06	4.43E-15		
9805-282	^{58}Ni (n,p) ^{58}Co	3.70E+06	5.02E+07	7.18E-15	6.71E-15	6.71E-15
9805-287	^{58}Ni (n,p) ^{58}Co	3.34E+06	4.53E+07	6.48E-15		
9805-292	^{58}Ni (n,p) ^{58}Co	3.33E+06	4.51E+07	6.46E-15		
9805-278	^{59}Co (n,γ) ^{60}Co	2.63E+07	4.91E+07	3.21E-12	3.37E-12	3.37E-12
9805-283	^{59}Co (n,γ) ^{60}Co	2.97E+07	5.55E+07	3.62E-12		
9805-284	^{59}Co (n,γ) ^{60}Co	2.55E+07	4.76E+07	3.11E-12		
9805-289	^{59}Co (n,γ) ^{60}Co	2.86E+07	5.34E+07	3.49E-12		
9805-290	^{59}Co (n,γ) ^{60}Co	2.80E+07	5.23E+07	3.41E-12		
9805-279	^{59}Co(Cd) (n,γ) ^{60}Co	1.56E+07	2.92E+07	1.90E-12	1.94E-12	1.94E-12
9805-285	^{59}Co(Cd) (n,γ) ^{60}Co	1.63E+07	3.05E+07	1.99E-12		
9805-275	^{238}U (n,f) ^{137}Cs	1.04E+06	5.73E+06	3.76E-14	3.76E-14	2.91E-14
9805-276	^{237}Np (n,f) ^{137}Cs	7.35E+06	4.05E+07	2.58E-13	2.58E-13	2.56E-13

Note:

(a) Measured activity decay corrected to July 1, 1998.

Table F-7 Measured Sensor Activities and Reaction Rates for Byron Unit 1 Surveillance Capsule Y

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
30146708005	^{63}Cu (n,α) ^{60}Co	8.69E+04	2.77E+05	4.23E-17	3.98E-17	3.98E-17
30146708011	^{63}Cu (n,α) ^{60}Co	8.07E+04	2.57E+05	3.93E-17		
30146708017	^{63}Cu (n,α) ^{60}Co	7.76E+04	2.48E+05	3.78E-17		
30146708007	^{54}Fe (n,p) ^{54}Mn	7.15E+03	2.60E+06	4.13E-15	3.70E-15	3.70E-15
30146708013	^{54}Fe (n,p) ^{54}Mn	6.22E+03	2.26E+06	3.59E-15		
30146708019	^{54}Fe (n,p) ^{54}Mn	5.89E+03	2.14E+06	3.40E-15		
30146708003	^{59}Co (n,γ) ^{60}Co	1.48E+07	4.72E+07	3.08E-12	2.75E-12	2.75E-12
30146708004	^{59}Co (n,γ) ^{60}Co	1.48E+07	4.72E+07	3.08E-12		
30146708008	^{59}Co (n,γ) ^{60}Co	1.28E+07	4.08E+07	2.66E-12		
30146708009	^{59}Co (n,γ) ^{60}Co	1.18E+07	3.76E+07	2.46E-12		
30146708014	^{59}Co (n,γ) ^{60}Co	1.35E+07	4.31E+07	2.81E-12		
30146708015	^{59}Co (n,γ) ^{60}Co	1.17E+07	3.73E+07	2.44E-12		
30146708010	^{59}Co(Cd) (n,γ) ^{60}Co	7.58E+06	2.42E+07	1.58E-12	1.55E-12	1.55E-12
30146708016	^{59}Co(Cd) (n,γ) ^{60}Co	7.35E+06	2.34E+07	1.53E-12		
30146708001	^{238}U (n,f) ^{137}Cs	1.34E+06	4.68E+06	3.50E-14	3.50E-14	2.52E-14
30146708002	^{237}Np (n,f) ^{137}Cs	8.24E+06	2.88E+07	2.10E-13	2.10E-13	2.08E-13

Notes:

(a) Measured activity decay corrected to May 5, 2015.

Table F-8 Multiple Foil Sensor Set Locations within the Reactor Cavity for Byron Unit 1 and Unit 2

Azimuth (Degrees)	Capsule Identification – EVND Sets 1S-1 and 2S-1		
	Core Top	Core Midplane	Core Bottom
0		A	
15		B	
30		C	
45	D	E	F

Azimuth (Degrees)	Capsule Identification – EVND Sets 1S-2 and 2S-2		
	Core Top	Core Midplane	Core Bottom
0		G	
15		H	
30		I	
45	J	K	L

Table F-9 Nuclear Parameters Used in the Evaluation of Byron Unit 1 EVND Capsules Set 1S-1 and 1S-2 and Byron Unit 2 EVND Capsules Set 2S-1 and 2S-2

Reaction of Interest	Atomic Weight[(a)] (g/g-atom)	Target Atom Fraction[(b)]	Product Half-life[(b,c)] (days)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	63.546	0.6917	1925.5
$^{46}Ti\ (n,p)\ ^{46}Sc$	47.867	0.0825	83.79
$^{54}Fe\ (n,p)\ ^{54}Mn$	55.845	0.05845	312.11
$^{58}Ni\ (n,p)\ ^{58}Co$	58.693	0.68077	70.82
$^{93}Nb\ (n,n')\ ^{93m}Nb$	92.906	1.0	5890.0
$^{59}Co\ (n,\gamma)\ ^{60}Co$	58.933	0.00438	1925.5
$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	58.933	0.00438	1925.5

Notes.

(a) Atomic weight data were taken from the Chart of the Nuclides 17th Edition, dated 2009 [38]

(b) Half-life and target atom fraction data for ^{63}Cu (n,α), ^{46}Ti(n,p), ^{54}Fe(n,p), ^{58}Ni(n,p), and ^{93}Nb (n,n') reactions were taken from ASTM Standard E 1005-10 [39].

(c) The half-life for the ^{59}Co (n,γ) reaction was taken from ASTM Standard E 1005-10 [39]

Table F-10 Reaction Rates for Byron Unit 1 EVND Capsule A; 0.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868004	^{63}Cu (n,α) ^{60}Co	4.49E+02	2.79E+03	4.25E-19	4.25E-19
3020868005	^{46}Ti (n,p) ^{46}Sc	1.86E+03	5.74E+03	5.53E-18	5.53E-18
3020868001	^{54}Fe (n,p) ^{54}Mn	9.76E+03	1.93E+04	3.07E-17	3.07E-17
3020868002	^{54}Fe (n,p) ^{54}Mn	9.78E+03	1.94E+04	3.07E-17	
3020868003	^{58}Ni (n,p) ^{58}Co	7.26E+04	2.73E+05	3.90E-17	3.90E-17
3020868095	^{93}Nb (n,n′) ^{93m}Nb	5.21E+04	9.01E+05	1.39E-16	1.39E-16
3020868006	^{59}Co (n,γ) ^{60}Co	3.10E+05	1.93E+06	4.30E-14	4.30E-14
3020868007	^{59}Co(Cd) (n,γ) ^{60}Co	1.89E+05	1.17E+06	2.62E-14	2.62E-14

Note.

(a) Measured activity decay corrected to January 25, 2010.

Table F-11 Reaction Rates for Byron Unit 1 EVND Capsule B; 14.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868011	^{63}Cu (n,α) ^{60}Co	5.73E+02	3.56E+03	5.43E-19	5.43E-19
3020868012	^{46}Ti (n,p) ^{46}Sc	2.55E+03	7.87E+03	7.58E-18	7.58E-18
3020868008	^{54}Fe (n,p) ^{54}Mn	1.40E+04	2.77E+04	4.40E-17	4.32E-17
3020868009	^{54}Fe (n,p) ^{54}Mn	1.35E+04	2.68E+04	4.24E-17	
3020868010	^{58}Ni (n,p) ^{58}Co	1.02E+05	3.83E+05	5.48E-17	5.48E-17
3020868096	^{93}Nb (n,n′) ^{93m}Nb	7.10E+04	1.23E+06	1.89E-16	1.89E-16
3020868013	^{59}Co (n,γ) ^{60}Co	5.22E+05	3.24E+06	7.24E-14	7.24E-14
3020868014	^{59}Co(Cd) (n,γ) ^{60}Co	2.92E+05	1.81E+06	4.05E-14	4.05E-14

Note.

(a) Measured activity decay corrected to January 25, 2010.

Table F-12 Reaction Rates for Byron Unit 1 EVND Capsule C; 29.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868018	^{63}Cu (n,α) ^{60}Co	4.69E+02	2.91E+03	4.44E-19	4.44E-19
3020868019	^{46}Ti (n,p) ^{46}Sc	2.04E+03	6.30E+03	6.07E-18	6.07E-18
3020868015	^{54}Fe (n,p) ^{54}Mn	1.14E+04	2.26E+04	3.58E-17	3.58E-17
3020868016	^{54}Fe (n,p) ^{54}Mn	1.14E+04	2.26E+04	3.58E-17	
3020868017	^{58}Ni (n,p) ^{58}Co	8.61E+04	3.23E+05	4.63E-17	4.63E-17
3020868097	^{93}Nb (n,n') ^{93m}Nb	6.91E+04	1.20E+06	1.84E-16	1.84E-16
3020868020	^{59}Co (n,γ) ^{60}Co	5.36E+05	3.33E+06	7.44E-14	7.44E-14
3020868021	^{59}Co(Cd) (n,γ) ^{60}Co	3.05E+05	1.89E+06	4.23E-14	4.23E-14

Note:

(a) Measured activity decay corrected to January 25, 2010.

Table F-13 Reaction Rates for Byron Unit 1 EVND Capsule D; 44.5° at 182.88 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868025	^{63}Cu (n,α) ^{60}Co	1.21E+02	7.51E+02	1.15E-19	1.15E-19
3020868026	^{46}Ti (n,p) ^{46}Sc	6.22E+02	1.92E+03	1.85E-18	1.85E-18
3020868022	^{54}Fe (n,p) ^{54}Mn	3.00E+03	5.94E+03	9.43E-18	9.86E-18
3020868023	^{54}Fe (n,p) ^{54}Mn	3.27E+03	6.48E+03	1.03E-17	
3020868024	^{58}Ni (n,p) ^{58}Co	2.95E+04	1.11E+05	1.59E-17	1.59E-17
3020868098	^{93}Nb (n,n') ^{93m}Nb	2.42E+04	4.19E+05	6.46E-17	6.46E-17
3020868027	^{59}Co (n,γ) ^{60}Co	1.48E+05	9.19E+05	2.05E-14	2.05E-14
3020868028	^{59}Co(Cd) (n,γ) ^{60}Co	9.93E+04	6.17E+05	1.38E-14	1.38E-14

Note:

(a) Measured activity decay corrected to January 25, 2010.

Table F-14 Reaction Rates for Byron Unit 1 EVND Capsule E; 44.5° at 0 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868032	^{63}Cu (n,α) ^{60}Co	3.55E+02	2.20E+03	3.36E-19	3.36E-19
3020868033	^{46}Ti (n,p) ^{46}Sc	1.51E+03	4.66E+03	4.49E-18	4.49E-18
3020868029	^{54}Fe (n,p) ^{54}Mn	9.07E+03	1.80E+04	2.85E-17	2.80E-17
3020868030	^{54}Fe (n,p) ^{54}Mn	8.76E+03	1.74E+04	2.75E-17	
3020868031	^{58}Ni (n,p) ^{58}Co	6.82E+04	2.56E+05	3.66E-17	3.66E-17
3020868099	^{93}Nb (n,n') ^{93m}Nb	6.08E+04	1.05E+06	1.62E-16	1.62E-16
3020868034	^{59}Co (n,γ) ^{60}Co	3.31E+05	2.06E+06	4.59E-14	4.59E-14
3020868035	^{59}Co(Cd) (n,γ) ^{60}Co	2.21E+05	1.37E+06	3.07E-14	3.07E-14

Note:

(a) Measured activity decay corrected to January 25, 2010

Table F-15 Reaction Rates for Byron Unit 1 EVND Capsule F; 44.5° at -182.88 cm Relative to Core Midplane (1S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
3020868039	^{63}Cu (n,α) ^{60}Co	1.42E+02	8.82E+02	1.35E-19	1.35E-19
3020868040	^{46}Ti (n,p) ^{46}Sc	7.03E+02	2.17E+03	2.09E-18	2.09E-18
3020868036	^{54}Fe (n,p) ^{54}Mn	3.98E+03	7.89E+03	1.25E-17	1.20E-17
3020868037	^{54}Fe (n,p) ^{54}Mn	3.68E+03	7.29E+03	1.16E-17	
3020868038	^{58}Ni (n,p) ^{58}Co	3.29E+04	1.24E+05	1.77E-17	1.77E-17
3020868100	^{93}Nb (n,n') ^{93m}Nb	2.46E+04	4.25E+05	6.56E-17	6.56E-17
3020868041	^{59}Co (n,γ) ^{60}Co	1.86E+05	1.16E+06	2.58E-14	2.58E-14
3020868042	^{59}Co(Cd) (n,γ) ^{60}Co	1.26E+05	7.82E+05	1.75E-14	1.75E-14

Note

(a) Measured activity decay corrected to January 25, 2010

Table F-16 Reaction Rates for Byron Unit 1 EVND Capsule G; 0.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926001	^{54}Fe (n,p) ^{54}Mn	1.56E+04	2.02E+04	3.21E-17	3.19E-17
30218926002	^{54}Fe (n,p) ^{54}Mn	1.54E+04	2.00E+04	3.17E-17	
30218926003	^{58}Ni (n,p) ^{58}Co	1.10E+05	2.85E+05	4.08E-17	4.08E-17
30218926004	^{63}Cu (n,α) ^{60}Co	1.56E+03	2.75E+03	4.20E-19	4.20E-19
30218926005	^{46}Ti (n,p) ^{46}Sc	2.58E+03	5.79E+03	5.58E-18	5.58E-18
30218926006	^{93}Nb (n,n′) ^{93m}Nb	2.37E+05	9.33E+05	1.44E-16	1.44E-16
30218926007	^{59}Co (n,γ) ^{60}Co	1.14E+06	2.01E+06	4.49E-14	4.49E-14
30218926008	^{59}Co(Cd) (n,γ) ^{60}Co	6.74E+05	1.19E+06	2.66E-14	2.66E-14

Note:

(a) Measured activity decay corrected to June 1, 2017.

Table F-17 Reaction Rates for Byron Unit 1 EVND Capsule H; 14.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926009	^{54}Fe (n,p) ^{54}Mn	2.24E+04	2.87E+04	4.55E-17	4.56E-17
30218926010	^{54}Fe (n,p) ^{54}Mn	2.25E+04	2.88E+04	4.57E-17	
30218926011	^{58}Ni (n,p) ^{58}Co	1.63E+05	4.17E+05	5.97E-17	5.97E-17
30218926012	^{63}Cu (n,α) ^{60}Co	2.08E+03	3.66E+03	5.58E-19	5.58E-19
30218926013	^{46}Ti (n,p) ^{46}Sc	3.65E+03	8.10E+03	7.80E-18	7.80E-18
30218926014	^{93}Nb (n,n′) ^{93m}Nb	3.37E+05	1.33E+06	2.04E-16	2.04E-16
30218926015	^{59}Co (n,γ) ^{60}Co	1.85E+06	3.25E+06	7.26E-14	7.26E-14
30218926016	^{59}Co(Cd) (n,γ) ^{60}Co	1.06E+06	1.86E+06	4.16E-14	4.16E-14

Note.

(a) Measured activity decay corrected to June 1, 2017.

Table F-18 Reaction Rates for Byron Unit 1 EVND Capsule I; 29.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926017	^{54}Fe (n,p) ^{54}Mn	1.93E+04	2.46E+04	3.91E-17	3.88E-17
30218926018	^{54}Fe (n,p) ^{54}Mn	1.91E+04	2.44E+04	3.86E-17	
30218926019	^{58}Ni (n,p) ^{58}Co	1.41E+05	3.60E+05	5.15E-17	5.15E-17
30218926020	^{63}Cu (n,α) ^{60}Co	1.71E+03	3.00E+03	4.58E-19	4.58E-19
30218926021	^{46}Ti (n,p) ^{46}Sc	3.06E+03	6.78E+03	6.53E-18	6.53E-18
30218926022	^{93}Nb (n,n') ^{93m}Nb	3.20E+05	1.26E+06	1.94E-16	1.94E-16
30218926023	^{59}Co (n,γ) ^{60}Co	1.93E+06	3.39E+06	7.57E-14	7.57E-14
30218926024	^{59}Co(Cd) (n,γ) ^{60}Co	1.10E+06	1.93E+06	4.31E-14	4.31E-14

Note:

(a) Measured activity decay corrected to June 1, 2017.

Table F-19 Reaction Rates for Byron Unit 1 EVND Capsule J; 44.5° at 182.88 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926025	^{54}Fe (n,p) ^{54}Mn	5.10E+03	6.61E+03	1.05E-17	1.09E-17
30218926026	^{54}Fe (n,p) ^{54}Mn	5.48E+03	7.11E+03	1.13E-17	
30218926027	^{58}Ni (n,p) ^{58}Co	4.75E+04	1.27E+05	1.81E-17	1.81E-17
30218926028	^{63}Cu (n,α) ^{60}Co	4.48E+02	7.86E+02	1.20E-19	1.20E-19
30218926029	^{46}Ti (n,p) ^{46}Sc	9.25E+02	2.14E+03	2.06E-18	2.06E-18
30218926030	^{93}Nb (n,n') ^{93m}Nb	1.20E+05	4.72E+05	7.27E-17	7.27E-17
30218926031	^{59}Co (n,γ) ^{60}Co	5.10E+05	8.95E+05	2.00E-14	2.00E-14
30218926032	^{59}Co(Cd) (n,γ) ^{60}Co	3.55E+05	6.23E+05	1.39E-14	1.39E-14

Note:

(a) Measured activity decay corrected to June 1, 2017.

Table F-20 Reaction Rates for Byron Unit 1 EVND Capsule K; 44.5° at 0 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926033	^{54}Fe (n,p) ^{54}Mn	1.44E+04	1.86E+04	2.95E-17	2.93E-17
30218926034	^{54}Fe (n,p) ^{54}Mn	1.42E+04	1.84E+04	2.91E-17	
30218926035	^{58}Ni (n,p) ^{58}Co	1.07E+05	2.79E+05	3.99E-17	3.99E-17
30218926036	^{63}Cu (n,α) ^{60}Co	1.26E+03	2.22E+03	3.38E-19	3.38E-19
30218926037	^{46}Ti (n,p) ^{46}Sc	2.15E+03	4.86E+03	4.68E-18	4.68E-18
30218926038	^{93}Nb (n,n') ^{93m}Nb	2.98E+05	1.17E+06	1.81E-16	1.81E-16
30218926039	^{59}Co (n,γ) ^{60}Co	1.21E+06	2.13E+06	4.75E-14	4.75E-14
30218926040	^{59}Co(Cd) (n,γ) ^{60}Co	8.10E+05	1.42E+06	3.18E-14	3.18E-14

Note.

(a) Measured activity decay corrected to June 1, 2017.

Table F-21 Reaction Rates for Byron Unit 1 EVND Capsule L; 44.5° at -182.88 cm Relative to Core Midplane (1S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
30218926041	^{54}Fe (n,p) ^{54}Mn	6.32E+03	8.17E+03	1.30E-17	1.24E-17
30218926042	^{54}Fe (n,p) ^{54}Mn	5.76E+03	7.45E+03	1.18E-17	
30218926043	^{58}Ni (n,p) ^{58}Co	4.82E+04	1.25E+05	1.79E-17	1.79E-17
30218926044	^{63}Cu (n,α) ^{60}Co	4.63E+02	8.14E+02	1.24E-19	1.24E-19
30218926045	^{46}Ti (n,p) ^{46}Sc	9.71E+02	2.19E+03	2.11E-18	2.11E-18
30218926046	^{93}Nb (n,n') ^{93m}Nb	1.11E+05	4.36E+05	6.73E-17	6.73E-17
30218926047	^{59}Co (n,γ) ^{60}Co	6.75E+05	1.19E+06	2.65E-14	2.65E-14
30218926048	^{59}Co(Cd) (n,γ) ^{60}Co	4.52E+05	7.94E+05	1.77E-14	1.77E-14

Note:

(a) Measured activity decay corrected to June 1, 2017.

Table F-22 Measured Sensor Activities and Reaction Rates for Byron Unit 2 Surveillance Capsule U

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
89-0697	^{63}Cu (n,α) ^{60}Co	5.36E+04	4.18E+05	6.38E-17	6.17E-17	6.17E-17
89-0702	^{63}Cu (n,α) ^{60}Co	5.02E+04	3.92E+05	5.97E-17		
89-0698	^{54}Fe (n,p) ^{54}Mn	1.43E+06	3.99E+06	6.34E-15	5.98E-15	5.98E-15
89-0703	^{54}Fe (n,p) ^{54}Mn	1.33E+06	3.71E+06	5.89E-15		
89-0707	^{54}Fe (n,p) ^{54}Mn	1.29E+06	3.60E+06	5.72E-15		
89-0699	^{58}Ni (n,p) ^{58}Co	8.64E+06	5.80E+07	8.30E-15	7.74E-15	7.74E-15
89-0704	^{58}Ni (n,p) ^{58}Co	7.73E+06	5.19E+07	7.43E-15		
89-0708	^{58}Ni (n,p) ^{58}Co	7.80E+06	5.23E+07	7.49E-15		
89-0693	^{238}U (n,f) ^{137}Cs	1.70E+05	6.59E+06	4.33E-14	4.33E-14	3.63E-14
89-0694	^{237}Np (n,f) ^{137}Cs	1.34E+06	5.19E+07	3.26E-13	3.26E-13	3.23E-13
89-0695	^{59}Co (n,γ) ^{60}Co	1.11E+07	8.66E+07	5.65E-12	5.70E-12	5.70E-12
89-0700	^{59}Co (n,γ) ^{60}Co	1.13E+07	8.81E+07	5.75E-12		
89-0705	^{59}Co (n,γ) ^{60}Co	1.12E+07	8.73E+07	5.70E-12		
89-0696	^{59}Co(Cd) (n,γ) ^{60}Co	5.63E+06	4.39E+07	2.86E-12	2.79E-12	2.79E-12
89-0701	^{59}Co(Cd) (n,γ) ^{60}Co	5.00E+06	3.90E+07	2.54E-12		
89-0706	^{59}Co(Cd) (n,γ) ^{60}Co	5.82E+06	4.54E+07	2.96E-12		

Note

(a) Measured activity decay corrected to May 17, 1989.

Table F-23 Measured Sensor Activities and Reaction Rates for Byron Unit 2 Surveillance Capsule W

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
94-0690	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.37E+05	3.57E+05	5.44E-17	5.05E-17	5.05E-17
94-0695	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.23E+05	3.20E+05	4.89E-17		
94-0700	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.21E+05	3.15E+05	4.81E-17		
94-0692	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.63E+06	3.18E+06	5.05E-15	4.70E-15	4.70E-15
94-0697	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.47E+06	2.87E+06	4.56E-15		
94-0702	$^{54}Fe\ (n,p)\ ^{54}Mn$	1.45E+06	2.83E+06	4.49E-15		
94-0691	$^{58}Ni\ (n,p)\ ^{58}Co$	6.42E+06	4.95E+07	7.08E-15	6.60E-15	6.60E-15
94-0696	$^{58}Ni\ (n,p)\ ^{58}Co$	5.81E+06	4.48E+07	6.41E-15		
94-0701	$^{58}Ni\ (n,p)\ ^{58}Co$	5.73E+06	4.41E+07	6.32E-15		
94-0686	$^{238}U\ (n,f)\ ^{137}Cs$	5.04E+05	5.15E+06	3.38E-14	3.38E-14	2.75E-14
94-0687	$^{237}Np\ (n,f)\ ^{137}Cs$	3.85E+06	3.93E+07	2.47E-13	2.47E-13	2.45E-13
94-0688	$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.30E+07	5.99E+07	3.91E-12	3.90E-12	3.90E-12
94-0698	$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.29E+07	5.96E+07	3.89E-12		
94-0689	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	1.15E+07	3.00E+07	1.95E-12	1.98E-12	1.98E-12
94-0693	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	1.18E+07	3.07E+07	2.01E-12		

Note
(a) Measured activity decay corrected to March 18, 1994.

Table F-24 Measured Sensor Activities and Reaction Rates for Byron Unit 2 Surveillance Capsule X

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
99-0201	^{63}Cu (n,α) ^{60}Co	1.81E+05	3.31E+05	5.05E-17	4.75E-17	4.75E-17
99-0206	^{63}Cu (n,α) ^{60}Co	1.66E+05	3.04E+05	4.63E-17		
99-0211	^{63}Cu (n,α) ^{60}Co	1.64E+05	3.00E+05	4.57E-17		
99-0203	^{54}Fe (n,p) ^{54}Mn	1.37E+06	3.12E+06	4.96E-15	4.67E-15	4.67E-15
99-0208	^{54}Fe (n,p) ^{54}Mn	1.25E+06	2.85E+06	4.52E-15		
99-0213	^{54}Fe (n,p) ^{54}Mn	1.25E+06	2.85E+06	4.52E-15		
99-0202	^{58}Ni (n,p) ^{58}Co	2.47E+06	4.77E+07	6.84E-15	6.56E-15	6.56E-15
99-0207	^{58}Ni (n,p) ^{58}Co	2.35E+06	4.54E+07	6.50E-15		
99-0212	^{58}Ni (n,p) ^{58}Co	2.29E+06	4.43E+07	6.34E-15		
99-0363	^{238}U (n,f) ^{137}Cs	8.51E+05	4.96E+06	3.25E-14	3.25E-14	2.53E-14
99-0198	^{237}Np (n,f) ^{137}Cs	7.49E+06	4.36E+07	2.74E-13	2.74E-13	2.71E-13
99-0199	^{59}Co (n,γ) ^{60}Co	3.11E+07	5.69E+07	3.71E-12	3.66E-12	3.66E-12
99-0204	^{59}Co (n,γ) ^{60}Co	3.06E+07	5.59E+07	3.65E-12		
99-0209	^{59}Co (n,γ) ^{60}Co	3.03E+07	5.54E+07	3.61E-12		
99-0200	^{59}Co(Cd) (n,γ) ^{60}Co	1.60E+07	2.93E+07	1.91E-12	1.92E-12	1.92E-12
99-0205	^{59}Co(Cd) (n,γ) ^{60}Co	1.61E+07	2.94E+07	1.92E-12		
99-0210	^{59}Co(Cd) (n,γ) ^{60}Co	1.63E+07	2.98E+07	1.94E-12		

Note:

(a) Measured activity decay corrected to January 20, 1999.

Table F-25 Measured Sensor Activities and Reaction Rates for Byron Unit 2 Surveillance Capsule Y

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
299005	^{63}Cu (n,α) ^{60}Co	1.24E+05	2.90E+05	4.42E-17	4.10E-17	4.10E-17
299010	^{63}Cu (n,α) ^{60}Co	1.11E+05	2.59E+05	3.96E-17		
299015	^{63}Cu (n,α) ^{60}Co	1.10E+05	2.57E+05	3.92E-17		
299007	^{54}Fe (n,p) ^{54}Mn	3.83E+04	2.57E+06	4.08E-15	3.89E-15	3.89E-15
299012	^{54}Fe (n,p) ^{54}Mn	3.62E+04	2.43E+06	3.86E-15		
299017	^{54}Fe (n,p) ^{54}Mn	3.51E+04	2.36E+06	3.74E-15		
299001	^{238}U (n,f) ^{137}Cs	1.53E+06	4.81E+06	3.59E-14	3.59E-14	2.56E-14
299002	^{237}Np (n,f) ^{137}Cs	9.08E+06	2.85E+07	2.05E-13	2.05E-13	2.03E-13
299003	^{59}Co (n,γ) ^{60}Co	1.84E+07	4.30E+07	2.81E-12	2.72E-12	2.72E-12
299008	^{59}Co (n,γ) ^{60}Co	1.80E+07	4.21E+07	2.74E-12		
299013	^{59}Co (n,γ) ^{60}Co	1.71E+07	4.00E+07	2.61E-12		
299004	$^{59}Co(Cd)$ (n,γ) ^{60}Co	1.00E+07	2.34E+07	1.52E-12	1.48E-12	1.48E-12
299009	$^{59}Co(Cd)$ (n,γ) ^{60}Co	9.36E+06	2.19E+07	1.43E-12		
299014	$^{59}Co(Cd)$ (n,γ) ^{60}Co	9.85E+06	2.30E+07	1.50E-12		

Note:

(a) Measured activity decay corrected to June 12, 2015.

Table F-26 Reaction Rates for Byron Unit 2 EVND Capsule A; 0.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190004	^{63}Cu (n,α) ^{60}Co	4.46E+02	2.68E+03	4.09E-19	4.09E-19
34190005	^{46}Ti (n,p) ^{46}Sc	1.49E+03	5.48E+03	5.28E-18	5.28E-18
34190001	^{54}Fe (n,p) ^{54}Mn	9.30E+03	1.89E+04	2.99E-17	2.95E-17
34190002	^{54}Fe (n,p) ^{54}Mn	9.03E+03	1.83E+04	2.91E-17	
34190003	^{58}Ni (n,p) ^{58}Co	5.74E+04	2.65E+05	3.80E-17	3.80E-17
34190008	^{93}Nb (n,n') ^{93m}Nb	5.36E+04	8.89E+05	1.37E-16	1.37E-16
34190006	^{59}Co (n,γ) ^{60}Co	3.07E+05	1.84E+06	3.21E-14	3.21E-14
34190007	^{59}Co(Cd) (n,γ) ^{60}Co	1.89E+05	1.14E+06	1.98E-14	1.98E-14

Note:

(a) Measured activity decay corrected to September 20, 2010.

Table F-27 Reaction Rates for Byron Unit 2 EVND Capsule B; 14.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190012	^{63}Cu (n,α) ^{60}Co	6.16E+02	3.70E+03	5.64E-19	5.64E-19
34190013	^{46}Ti (n,p) ^{46}Sc	2.14E+03	7.88E+03	7.59E-18	7.59E-18
34190009	^{54}Fe (n,p) ^{54}Mn	1.39E+04	2.82E+04	4.47E-17	4.27E-17
34190010	^{54}Fe (n,p) ^{54}Mn	1.26E+04	2.56E+04	4.06E-17	
34190011	^{58}Ni (n,p) ^{58}Co	8.59E+04	3.97E+05	5.69E-17	5.69E-17
34190016	^{93}Nb (n,n') ^{93m}Nb	8.08E+04	1.34E+06	2.07E-16	2.07E-16
34190014	^{59}Co (n,γ) ^{60}Co	5.86E+05	3.52E+06	6.13E-14	6.13E-14
34190015	^{59}Co(Cd) (n,γ) ^{60}Co	3.12E+05	1.87E+06	3.26E-14	3.26E-14

Note

(a) Measured activity decay corrected to September 20, 2010.

Table F-28 Reaction Rates for Byron Unit 2 EVND Capsule C; 29.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190020	^{63}Cu (n,α) ^{60}Co	4.93E+02	2.96E+03	4.52E-19	4.52E-19
34190021	^{46}Ti (n,p) ^{46}Sc	1.75E+03	6.44E+03	6.20E-18	6.20E-18
34190017	^{54}Fe (n,p) ^{54}Mn	1.14E+04	2.31E+04	3.67E-17	3.65E-17
34190018	^{54}Fe (n,p) ^{54}Mn	1.13E+04	2.29E+04	3.64E-17	
34190019	^{58}Ni (n,p) ^{58}Co	7.00E+04	3.24E+05	4.64E-17	4.64E-17
34190024	^{93}Nb (n,n′) ^{93m}Nb	7.86E+04	1.30E+06	2.01E-16	2.01E-16
34190022	^{59}Co (n,γ) ^{60}Co	5.59E+05	3.36E+06	5.84E-14	5.84E-14
34190023	^{59}Co(Cd) (n,γ) ^{60}Co	3.32E+05	1.99E+06	3.47E-14	3.47E-14

Note:

(a) Measured activity decay corrected to September 20, 2010.

Table F-29 Reaction Rates for Byron Unit 2 EVND Capsule D; 44.5° at 182.88 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190028	^{63}Cu (n,α) ^{60}Co	1.10E+02	6 60E+02	1.01E-19	1.01E-19
34190029	^{46}Ti (n,p) ^{46}Sc	4.63E+02	1.70E+03	1.64E-18	1.64E-18
34190025	^{54}Fe (n,p) ^{54}Mn	2.53E+03	5.13E+03	8.14E-18	8.48E-18
34190026	^{54}Fe (n,p) ^{54}Mn	2.74E+03	5.56E+03	8.82E-18	
34190027	^{58}Ni (n,p) ^{58}Co	2.04E+04	9.43E+04	1.35E-17	1.35E-17
34190032	^{93}Nb (n,n′) ^{93m}Nb	2.36E+04	3.92E+05	6.04E-17	6.04E-17
34190030	^{59}Co (n,γ) ^{60}Co	1.39E+05	8.35E+05	1.45E-14	1.45E-14
34190031	^{59}Co(Cd) (n,γ) ^{60}Co	9.31E+04	5.59E+05	9.73E-15	9.73E-15

Note.

(a) Measured activity decay corrected to September 20, 2010.

Table F-30 Reaction Rates for Byron Unit 2 EVND Capsule E; 44.5° at 0 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190036	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	3.84E+02	2.31E+03	3.52E-19	3.52E-19
34190037	$^{46}Ti\ (n,p)\ ^{46}Sc$	1.34E+03	4.93E+03	4.75E-18	4.75E-18
34190033	$^{54}Fe\ (n,p)\ ^{54}Mn$	9.39E+03	1.91E+04	3 02E-17	2.96E-17
34190034	$^{54}Fe\ (n,p)\ ^{54}Mn$	9.00E+03	1.83E+04	2.90E-17	
34190035	$^{58}Ni\ (n,p)\ ^{58}Co$	6.01E+04	2.78E+05	3.98E-17	3.98E-17
34190040	$^{93}Nb\ (n,n')\ ^{93m}Nb$	7.09E+04	1.18E+06	1.82E-16	1.82E-16
34190038	$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.71E+05	2.23E+06	3.88E-14	3.88E-14
34190039	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	2.40E+05	1.44E+06	2.51E-14	2.51E-14

Note:

(a) Measured activity decay corrected to September 20, 2010.

Table F-31 Reaction Rates for Byron Unit 2 EVND Capsule F; 44.5° at -182.88 cm Relative to Core Midplane (2S-1)

Sample	Target Isotope	Measured Activity (dps/g)[a]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
34190044	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.73E+02	1.04E+03	1.58E-19	1.58E-19
34190045	$^{46}Ti\ (n,p)\ ^{46}Sc$	7.18E+02	2.64E+03	2.55E-18	2.55E-18
34190041	$^{54}Fe\ (n,p)\ ^{54}Mn$	4.82E+03	9.78E+03	1.55E-17	1.49E-17
34190042	$^{54}Fe\ (n,p)\ ^{54}Mn$	4.42E+03	8.97E+03	1.42E-17	
34190043	$^{58}Ni\ (n,p)\ ^{58}Co$	3.33E+04	1.54E+05	2.21E-17	2.21E-17
34190048	$^{93}Nb\ (n,n')\ ^{93m}Nb$	3.25E+04	5.39E+05	8.32E-17	8.32E-17
34190046	$^{59}Co\ (n,\gamma)\ ^{60}Co$	1.94E+05	1.17E+06	2.03E-14	2.03E-14
34190047	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	1.25E+05	7.51E+05	1.31E-14	1.31E-14

Note:

(a) Measured activity decay corrected to September 20, 2010.

Table F-32 Reaction Rates for Byron Unit 2 EVND Capsule G; 0.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-004	^{63}Cu (n,α) ^{60}Co	1.47E+03	2.60E+03	3.97E-19	3.97E-19
-005	^{46}Ti (n,p) ^{46}Sc	2.23E+03	5.72E+03	5.51E-18	5.51E-18
-001	^{54}Fe (n,p) ^{54}Mn	1.40E+04	1.91E+04	3.03E-17	3.02E-17
-002	^{54}Fe (n,p) ^{54}Mn	1.39E+04	1.90E+04	3.01E-17	
-003	^{58}Ni (n,p) ^{58}Co	9.50E+04	2.85E+05	4.09E-17	4.09E-17
-006	^{93}Nb (n,n′) ^{93m}Nb	2.18E+05	8.56E+05	1.32E-16	1.32E-16
-007	^{59}Co (n,γ) ^{60}Co	9.71E+05	1.72E+06	2.99E-14	2.99E-14
-008	$^{59}Co(Cd)$ (n,γ) ^{60}Co	6.11E+05	1.08E+06	1.88E-14	1.88E-14

Note

(a) Measured activity decay corrected to January 15, 2018.

Table F-33 Reaction Rates for Byron Unit 2 EVND Capsule H; 14.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-012	^{63}Cu (n,α) ^{60}Co	1.92E+03	3.41E+03	5.20E-19	5.20E-19
-013	^{46}Ti (n,p) ^{46}Sc	2.85E+03	7.47E+03	7.20E-18	7.20E-18
-009	^{54}Fe (n,p) ^{54}Mn	1.87E+04	2.59E+04	4.11E-17	4.09E-17
-010	^{54}Fe (n,p) ^{54}Mn	1.85E+04	2.57E+04	4.07E-17	
-011	^{58}Ni (n,p) ^{58}Co	1.15E+05	3.54E+05	5.06E-17	5.06E-17
-014	^{93}Nb (n,n′) ^{93m}Nb	2.99E+05	1.18E+06	1.81E-16	1.81E-16
-015	^{59}Co (n,γ) ^{60}Co	1.78E+06	3 16E+06	5.51E-14	5.51E-14
-016	$^{59}Co(Cd)$ (n,γ) ^{60}Co	9.57E+05	1.70E+06	2 96E-14	2.96E-14

Note:

(a) Measured activity decay corrected to January 15, 2018

Table F-34 Reaction Rates for Byron Unit 2 EVND Capsule I; 29.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-020	^{63}Cu (n,α) ^{60}Co	1.49E+03	2.65E+03	4.04E-19	4.04E-19
-021	^{46}Ti (n,p) ^{46}Sc	2.24E+03	5.96E+03	5.74E-18	5.74E-18
-017	^{54}Fe (n,p) ^{54}Mn	1.57E+04	2.19E+04	3.47E-17	3.45E-17
-018	^{54}Fe (n,p) ^{54}Mn	1.55E+04	2.16E+04	3.43E-17	
-019	^{58}Ni (n,p) ^{58}Co	1.06E+05	3.31E+05	4.74E-17	4.74E-17
-022	^{93}Nb (n,n') ^{93m}Nb	2.87E+05	1.13E+06	1.74E-16	1.74E-16
-023	^{59}Co (n,γ) ^{60}Co	1.73E+06	3.07E+06	5.35E-14	5.35E-14
-024	^{59}Co(Cd) (n,γ) ^{60}Co	1.04E+06	1.85E+06	3.22E-14	3.22E-14

Note.

(a) Measured activity decay corrected to January 15, 2018

Table F-35 Reaction Rates for Byron Unit 2 EVND Capsule J; 44.5° at 182.88 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)[(a)]	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-028	^{63}Cu (n,α) ^{60}Co	4.08E+02	7.16E+02	1.09E-19	1.09E-19
-029	^{46}Ti (n,p) ^{46}Sc	7.60E+02	1.92E+03	1.85E-18	1.85E-18
-025	^{54}Fe (n,p) ^{54}Mn	4.37E+03	5.83E+03	9.25E-18	9.77E-18
-026	^{54}Fe (n,p) ^{54}Mn	4.87E+03	6.50E+03	1.03E-17	
-027	^{58}Ni (n,p) ^{58}Co	3.98E+04	1.18E+05	1.69E-17	1.69E-17
-030	^{93}Nb (n,n') ^{93m}Nb	1.08E+05	4.23E+05	6.52E-17	6.52E-17
-031	^{59}Co (n,γ) ^{60}Co	4.77E+05	8.38E+05	1.46E-14	1.46E-14
-032	^{59}Co(Cd) (n,γ) ^{60}Co	3.16E+05	5.55E+05	9.66E-15	9.66E-15

Note:

(a) Measured activity decay corrected to January 15, 2018

Table F-36 Reaction Rates for Byron Unit 2 EVND Capsule K; 44.5° at 0 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-036	^{63}Cu (n,α) ^{60}Co	1.20E+03	2.12E+03	3.24E-19	3.24E-19
-037	^{46}Ti (n,p) ^{46}Sc	1.88E+03	4.87E+03	4.70E-18	4.70E-18
-033	^{54}Fe (n,p) ^{54}Mn	1.32E+04	1.81E+04	2.86E-17	2.85E-17
-034	^{54}Fe (n,p) ^{54}Mn	1.31E+04	1.79E+04	2.84E-17	
-035	^{58}Ni (n,p) ^{58}Co	9.18E+04	2.79E+05	4.00E-17	4.00E-17
-038	^{93}Nb (n,n') ^{93m}Nb	2.82E+05	1.11E+06	1.71E-16	1.71E-16
-039	^{59}Co (n,γ) ^{60}Co	1.16E+06	2.05E+06	3.57E-14	3.57E-14
-040	^{59}Co(Cd) (n,γ) ^{60}Co	7.33E+05	1.30E+06	2.26E-14	2.26E-14

Note

(a) Measured activity decay corrected to January 15, 2018.

Table F-37 Reaction Rates for Byron Unit 2 EVND Capsule L; 44.5° at -182.88 cm Relative to Core Midplane (2S-2)

Sample	Target Isotope	Measured Activity (dps/g)(a)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)
-044	^{63}Cu (n,α) ^{60}Co	4.82E+02	8.59E+02	1.31E-19	1.31E-19
-045	^{46}Ti (n,p) ^{46}Sc	8.80E+02	2.40E+03	2.31E-18	2.31E-18
-041	^{54}Fe (n,p) ^{54}Mn	6.19E+03	8.80E+03	1.40E-17	1.33E-17
-042	^{54}Fe (n,p) ^{54}Mn	5.62E+03	7.99E+03	1.27E-17	
-043	^{58}Ni (n,p) ^{58}Co	4.37E+04	1.40E+05	2.00E-17	2.00E-17
-046	^{93}Nb (n,n') ^{93m}Nb	1.08E+05	4.25E+05	6.55E-17	6.55E-17
-047	^{59}Co (n,γ) ^{60}Co	6.07E+05	1.08E+06	1.88E-14	1.88E-14
-048	^{59}Co(Cd) (n,γ) ^{60}Co	4.07E+05	7.25E+05	1.26E-14	1.26E-14

Note:

(a) Measured activity decay corrected to January 15, 2018.

Table F-38 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 1 In-Vessel Capsules

Reaction	M/C				Avg. M/C	% Unc. (1σ)
	Capsule					
	U	X	W	Y		
^{63}Cu (n,α) ^{60}Co	1.07	1.04	1.12	1.08	1.07	2.3%
^{54}Fe (n,p) ^{54}Mn	0.90	0.90	1.03	0.93	0.94	5.6%
^{58}Ni (n,p) ^{58}Co	0.82	0.88	1.05	-	0.91	11.9%
$^{238}U(Cd)$ (n,f) ^{137}Cs	0.92	0.93	1.19	1.18	1.06	14.2%
$^{237}Np(Cd)$ (n,f) ^{137}Cs	0.93	0.74	1.07	1.01	0.93	14.8%
Linear Average of M/C Results					0.98	11.9%

Table F-39 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 1 Ex-Vessel Midplane Capsules

Reaction	M/C								Avg. M/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
^{63}Cu (n,α) ^{60}Co	0.92	0.95	0.80	0.88	0.89	0.96	0.81	0.87	0.89	6.6%
^{46}Ti (n,p) ^{46}Sc	0.89	0.96	0.79	0.84	0.89	0.98	0.83	0.86	0.88	7.3%
^{54}Fe (n,p) ^{54}Mn	0.93	0.99	0.83	0.88	0.95	1.03	0.87	0.90	0.92	7.2%
^{58}Ni (n,p) ^{58}Co	0.85	0.89	0.75	0.79	0.87	0.96	0.82	0.84	0.85	7.6%
^{93}Nb (n,n') ^{93m}Nb	1.06	1.05	0.96	0.97	1.08	1.11	0.98	1.05	1.03	5.4%
Linear Average of M/C Results									0.91	6.5%

Table F-40 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 1 Ex-Vessel Off-Midplane Capsules

Reaction	M/C			
	Capsule			
	D	F	J	L
^{63}Cu (n,α) ^{60}Co	0.68	0.85	0.67	0.76
^{46}Ti (n,p) ^{46}Sc	0.78	0.94	0.82	0.92
^{54}Fe (n,p) ^{54}Mn	0.70	0.90	0.72	0.90
^{58}Ni (n,p) ^{58}Co	0.77	0.91	0.83	0.90
^{93}Nb (n,n') ^{93m}Nb	0.87	0.93	0.92	0.92

Table F-41 Comparison of Calculated and Best-Estimate Exposure Rates for Byron Unit 1 In-Vessel Capsules

Reaction	BE/C				Avg. BE/C	% Unc. (1σ)
	Capsule					
	U	X	W	Y		
Fast Neutron Fluence rate (E > 1.0 MeV)	0.87	0.85	1.07	1.01	0.95	11.3%
dpa/s	0.89	0.85	1.06	1.02	0.96	10.6%

Table F-42 Comparison of Calculated and Best-Estimate Exposure Rates for Byron Unit 1 Ex-Vessel Midplane Capsules

Reaction	BE/C								Avg. BE/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.97	0.99	0.87	0.89	1.00	1.05	0.91	0.96	0.96	6.4%
dpa/s	1.02	1.02	0.93	0.94	1.04	1.07	0.96	1.01	1.00	5.0%

Table F-43 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 2 In-Vessel Capsules

Reaction	M/C				Avg. M/C	% Unc. (1σ)
	Capsule					
	U	W	X	Y		
^{63}Cu (n,α) ^{60}Co	1.14	1.17	1.11	1.12	1.14	2.3%
^{54}Fe (n,p) ^{54}Mn	0.97	0.98	0.99	0.98	0.98	0.8%
^{58}Ni (n,p) ^{58}Co	0.89	0.98	0.99	-	0.95	5.8%
^{238}U(Cd) (n,f) ^{137}Cs	1.07	1.06	0.99	1.21	1.08	8.5%
^{237}Np(Cd) (n,f) ^{137}Cs	0.96	0.96	1.09	0.99	1.00	6.2%
Linear Average of M/C Results					1.03	8.3%

Table F-44 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 2 Ex-Vessel Midplane Capsules

Reaction	M/C Capsule A	B	C	E	G	H	I	K	Avg. M/C	% Unc. (1σ)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	0.92	0.99	0.80	0.90	0.89	0.96	0.77	0.87	0.89	8.4%
$^{46}Ti\ (n,p)\ ^{46}Sc$	0.89	0.97	0.79	0.87	0.92	0.97	0.79	0.90	0.89	7.9%
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.94	0.98	0.82	0.91	0.96	0.99	0.84	0.92	0.92	6.8%
$^{58}Ni\ (n,p)\ ^{58}Co$	0.86	0.93	0.74	0.84	0.93	0.87	0.81	0.89	0.86	7.4%
$^{93}Nb\ (n,n')\ ^{93m}Nb$	1.09	1.16	1.02	1.06	1.06	1.07	0.95	1.06	1.06	5.6%
Linear Average of M/C Results									0.92	10.3%

Table F-45 Comparison of Measured and Calculated Threshold Foil Reaction Rates for Byron Unit 2 Ex-Vessel Off-Midplane Capsules

Reaction	M/C Capsule D	F	J	L
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	0.55	1.03	0.64	0.85
$^{46}Ti\ (n,p)\ ^{46}Sc$	0.63	1.18	0.77	1.07
$^{54}Fe\ (n,p)\ ^{54}Mn$	0.55	1.15	0.68	1.03
$^{58}Ni\ (n,p)\ ^{58}Co$	0.60	1.18	0.81	1.06
$^{93}Nb\ (n,n')\ ^{93m}Nb$	0.75	1.22	0.88	0.96

Table F-46 Comparison of Calculated and Best-Estimate Exposure Rates for Byron Unit 2 In Vessel Capsules

Reaction	BE/C Capsule U	W	X	Y	Avg. BE/C	% Unc. (1σ)
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.94	0.97	0.99	1.04	0.99	4.3%
dpa/s	0.96	0.98	1.00	1.03	0.99	3.0%

Table F-47 Comparison of Calculated and Best-Estimate Exposure Rates for Byron Unit 2 Ex-Vessel Midplane Capsules

Reaction	BE/C								Avg. BE/C	% Unc. (1σ)
	Capsule									
	A	B	C	E	G	H	I	K		
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.96	1.01	0.86	0.93	0.99	0.97	0.88	0.96	0.95	5.5%
dpa/s	0.99	1.04	0.91	0.97	1.00	0.99	0.91	0.98	0.97	4.6%

APPENDIX G APPLICABILITY OF BAW-2308 TO BYRON AND BRAIDWOOD NOZZLE-TO-SHELL WELDS

G.1 INTRODUCTION

BAW-2308 [6] was developed to establish alternative unirradiated reference temperatures based on the "Master Curve" method for reactor vessel (RV) welds produced using Linde 80 flux. The report makes use of fracture toughness testing and American Society of Mechanical Engineers (ASME) Code Case N-629 [40] to develop heat-specific and generic Linde 80 reference temperatures. The alternatives to RT_{NDT} (termed RTT_0) along with corresponding margin terms (σ_I) developed in [6] were approved by the NRC, with the latest approved version of the report being Revision 2-A [6].

The focus of [6] is on Babcock & Wilcox Owners Group (B&WOG) RV working group beltline Linde 80 welds, as these welds were the materials expected to see a benefit from using the Master Curve method at that time since they were the limiting materials. However, since the development of [6], areas beyond the traditional beltline region of the RV termed the "extended beltline" must now also be analyzed consistent with Regulatory Issue Summary (RIS) 2014-11 [14]. Since the extended beltline was not considered in the development of [6], materials exist in reactor vessel extended beltline regions to which [6] applies that were not identified when [6] was completed.

One group of materials often comprising a part of the extended beltline is the primary inlet and outlet nozzle-to-shell welds. At Byron Units 1 and 2, as well as Braidwood Units 1 and 2, the nozzle-to-shell welds were determined to be Linde 80 welds. This attachment demonstrates that the work completed in [6] is applicable to certain Byron and Braidwood nozzle-to-shell welds.

G.2 BYRON AND BRAIDWOOD NOZZLE-TO-SHELL MATERIALS

Table G-1 and Table G-2 summarize the material properties for the Byron and Braidwood nozzle-to-shell welds, respectively. Each of the welds in these tables is a Linde 80 flux type and Mn-Mo-Ni filler type weld fabricated using single wire automatic submerged arc welding. The filler wire qualification tests for each of these welds are dated from 1973 to 1976. All of the information in Tables G-1 and G-2 come directly from original B&W weld certification records.

Table G-1 Byron Units 1 and 2 Nozzle-to-Shell Weld Material Properties

Material	Heat #	Flux Lot	Cu wt. %	Ni wt. %	Ultimate Strength (ksi)	Yield Strength (ksi)
Byron Unit 1[(a)]						
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-419	1P5412	8969	0.178	0.69	92.5	76
Byron Unit 2						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-559	41403	8061	0.15	0.59	81.25	64.5
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-545	442010	8060	0.22	0.63	81	61.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-559	41403	8061	0.15	0.59	81.25	64.5

Note:

(a) Additional nozzle-to-shell weld materials exist for Byron Unit 1; however, use of the material properties from [6] for these additional materials is not considered in this Appendix

Table G-2 Braidwood Units 1 and 2 Nozzle-to-Shell Weld Material Properties

Material	Heat #	Flux Lot	Cu wt. %	Ni wt. %	Ultimate Strength (ksi)	Yield Strength (ksi)
Braidwood Unit 1						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-598	41403	0852	0.29	0.56	87	67.75
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-598	41403	0852	0.29	0.56	87	67.75
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-588	41403	8119	0.29	0.63	82.5	65.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-579	442010	8119	0.25	0.63	85	68
Braidwood Unit 2						
Inlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-654	41404	0261	0.18	0.52	83.5	66.25
Outlet Nozzle to Nozzle Shell Forging Circ. Weld Seams WF-654	41404	0261	0.18	0.52	83.5	66.25

G.3 COMPARISON OF WELD MATERIALS

Comparison of the identifiers for the Byron and Braidwood nozzle-to-shell welds provides evidence that these welds fall into the same class. The identification numbers (WF-) in Tables G-1 and G-2 show that the naming convention utilized for the Byron and Braidwood nozzle-to-shell welds is consistent with those utilized for Linde 80 welds in [6] (see [6], Revision 1 Table 3-2). Additionally, some of the heat numbers in Tables G-1 and G-2 are identical to Heat numbers reported in [6], Revision 1-A Table 3-2, while the majority of those not specifically listed in [6], Revision 1-A are extremely similar to a Heat number reported in [6], Revision 1-A. Furthermore, the four-digit flux lot naming convention is also consistent with Linde 80 weld lot numbers for beltline welds. These naming conventions, in combination with the fact that these welds were fabricated by the same supplier (B&W) in approximately the same time period (1960's to 1970's) as the beltline Linde 80 welds discussed in [6] ensures that the results of [6] apply to the Byron and Braidwood nozzle-to-shell welds.

For Byron Unit 2 and Braidwood Units 1 and 2, as indicated by bare wire versus groove weld chemistries reported in the CMTRs, these weld wires were Cu coated when used for the nozzle-to-shell welds. It is noted, however, that the beltline welds of the Byron and Braidwood units did not use Cu coated weld wires. For the Byron Unit 1 weld in Table G-1, bare wire chemistry values are not available; however, based on the Cu value, it is believed this weld was also copper coated. The copper content of these welds fall within or just outside of the range of copper values listed in Table 1-2 of [6], Revision 1. Additionally, the tensile properties (ultimate strength and yield strength) of the Byron and Braidwood nozzle-to-shell

welds fall well within the same range as the Linde 80 materials shown in Figures 3-1 and 3-2 of [6], Revision 1.

Table 3-3 of [6], Revision 1 contains a comparison of T_{NDT} values. For the majority of the welds listed in Tables G-1 and G-2, full drop-weight test results are unavailable. Only one weld record contains full test drop-weight results, and the corresponding T_{NDT} value was measured to be -60°F. This value is consistent with the T_{NDT} values in Table 3-2 of [6], Revision 1-A. The remaining welds contain drop-weight results at only one temperature with a result of two "no break" specimens. Thus, T_{NDT} cannot be determined precisely for the majority of the welds listed in Tables G-1 and G-2. Since less focus was placed on the extended beltline welds during fabrication of the RV, it is likely that documentation of full drop-weight testing was considered unnecessary at the time of fabrication.

Another common characteristic of Linde 80 welds are having low initial upper-shelf energy, compared to other reactor vessel welds. The Byron and Braidwood nozzle-to-shell welds also exhibit this characteristic, as the upper-shelf energy values for the majority of the Byron and Braidwood nozzle-to-shell welds are less than 80 ft-lbs, and the maximum USE value amongst these welds is only 85 ft-lbs. The upper shelf energy (USE) of the Linde 80 weld materials meet the 10 CFR 50, Appendix G requirements, i.e., > 50 ft-lbs at end of license, as analyzed in WCAP-18054-NP [17] (for Byron Unit 1), WCAP-18056-NP [18] (for Byron Unit 2).

It is noted that the nozzle-to-shell weld thickness is typically greater than the beltline weld thickness; however, as described in [41] Question 4, a difference in weld thickness over a range of 8 to 12 inches would not be expected to lead to systemic differences in material properties. Additionally, a large portion of the data utilized in [6] is based on nozzle dropout (ND) welds. The ND welds considered in [6] have a comparable thickness to the Byron and Braidwood nozzle-to-shell weld thickness. Since the σ term evaluation completed in [6] "accounts for any variation between the measured toughness of the source material (RVSP *[reactor vessel surveillance program]* block, ND or MD-1 *[Midland Unit 1]* beltline weld) and the actual vessel weld", applicable uncertainty is already considered in the [6] results.

Given the above discussion, it is concluded that the use of material properties from [6] is appropriate for the Byron and Braidwood nozzle-to-shell welds listed in Tables G-1 and G-2. Since the specific heats relevant to these welds were not analyzed in [6], the generic "all heats" RTT_0 and uncertainty values will be utilized. It is noted that the generic values contain an added 20°F conservatism compared to the heat-specific values, as described in [42].