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RA-21-0320 10 CFR 50.90
December 20, 2021

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

MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-369, 50-370 / RENEWED LICENSE NOS. NPF-9 AND NPF-17

SUBJECT: License Amendment Request (LAR) to Revise the Applicability Term for Pressure and Temperature Limit Curves of Technical Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

Ladies and Gentlemen,

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) hereby requests an amendment to revise the Technical Specifications (TS) for McGuire Nuclear Station (MNS), Units 1 and 2. The proposed change revises MNS TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," to reflect that Figures 3.4.3-1, 3.4.3-2 and 3.4.3-5 (Unit 1 P/T limit curves) are applicable up to 54 effective full power years (EFPY) and that Figures 3.4.3-3, 3.4.3-4 and 3.4.3-6 (Unit 2 P/T limit curves) are applicable up to 38.6 EFPY.

The Enclosure provides a description and assessment of the proposed change. Attachment 1 provides the existing MNS TS pages marked to show the proposed change. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides Westinghouse report, WCAP-17455-NP, "McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," Revision 0.

Duke Energy has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and has determined that the proposed changes do not involve a significant hazards consideration. The basis for this determination is included in the Enclosure.

Duke Energy requests approval of the proposed amendment within one year of the date this submittal is accepted by the NRC staff for review. Once approved, Duke Energy will implement the license amendment within 120 days.

In accordance with 10 CFR 50.91, a copy of this LAR is being sent to the designated North Carolina state official.

There are no new regulatory commitments contained in this submittal.

If there are any questions or additional information is needed, please contact Mr. Lee Grzeck, Fleet Licensing Manager (Acting), at (980) 373-1530.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 20, 2021.

Sincerely,

Thomas D. Ray

Thomas D. Ray, P.E.
Site Vice President
McGuire Nuclear Station

Enclosure: Description and Assessment of the Proposed Change

Attachments:

1. Technical Specification 3.4.3 Markup
2. Technical Specification 3.4.3 Retyped Pages
3. WCAP-17455-NP, "McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations"

cc: L. Dudes, Regional Administrator USNRC Region II
G. A. Hutto, USNRC Senior Resident Inspector – MNS
J. Klos, NRR Project Manager – MNS

Enclosure
Description and Assessment of the Proposed Change

1 SUMMARY DESCRIPTION

Duke Energy Carolinas, LLC (Duke Energy) hereby requests an amendment to revise the Technical Specifications (TS) for McGuire Nuclear Station (MNS), Units 1 and 2. The proposed change revises MNS TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," to reflect that Figures 3.4.3-1, 3.4.3-2 and 3.4.3-5 (Unit 1 P/T limit curves) are applicable up to 54 effective full power years (EFPY) and that Figures 3.4.3-3, 3.4.3-4 and 3.4.3-6 (Unit 2 P/T limit curves) are applicable up to 38.6 EFPY.

2 DETAILED DESCRIPTION

2.1 System Design and Operation

All components of the reactor coolant system (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. TS Limiting Condition for Operation (LCO) 3.4.3 limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

TS 3.4.3 contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

2.2 Current Technical Specification Requirements

MNS TS LCO 3.4.3 requires that RCS pressure, RCS temperature, and RCS heatup and cooldown rates be maintained within the limits specified in the P/T limit curves: Figures 3.4.3-1, 3.4.3-2, and 3.4.3-5 (Unit 1) and 3.4.3-3, 3.4.3-4, and 3.4.3-6 (Unit 2). Each figure is stated as applicable for a specified number of effective full power years (EFPY), as stated in the Figure descriptions, listed below.

- 3.4.3-1, “McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”
- 3.4.3-2, “McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”
- 3.4.3-3, “McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”
- 3.4.3-4, “McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”
- 3.4.3-5, “McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”

- 3.4.3-6, “McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641”

Additionally, each figure describes the Limiting Adjusted Reference Temperature (ART) values at the applicable EFPY (see example image below of Figure 3.4.3-1).

RCS P/T Limits
3.4.3

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

The P/T Limit Curve figures in Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, and 3.4.3-4 also contain a note stating that the criticality limit is based on the inservice hydrostatic test temperature for the service period up to 34 EFPY.

2.3 Reason for Proposed Change

NUREG-1772, “Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2” (Reference 3), which was published in March 2003, states in Section 4.2.3.2:

> “Normally, the P/T limits are calculated for several years into the future and remain valid for an established period of time not to exceed the current operating license expiration. The current P/T limit curves for the McGuire units are acceptable through 16 EFPYs of power operation; the current P/T limit curves for the Catawba units are acceptable though 15 EFPYs of power operation. Section 50.90 of 10 CFR requires licensees to submit new P/T limit curves for operating reactors for review and have the curves approved and implemented into the Technical Specifications for the reactor units prior to the expiration of the most current P/T limit curves approved in the Technical Specifications. The applicant will be required to submit the extended-period-of-operation P/T limit curves for the McGuire and Catawba RVs and have the curves approved against the criteria of 10 CFR Part 50, Appendix G, and implemented into the Technical Specifications prior to operation of the reactors during the extended operating terms for the units.
>
> The issue raised in open item 4.2-1 on the McGuire 1 TLAAs for neutron irradiation embrittlement (i.e., the McGuire 1 TLAAs for PTS, USE, and P/T limits), as stated in the staff's letter of September 9, 2002, does not change the staff's conclusion that the applicant is required to submit P/T limit curves for the period of extended operation before it begins operation beyond the first 40 years. However, since the P/T limits for McGuire 1 are based on the RTNDT value for the RV lower shell longitudinal welds fabricated from material heat No. 21935/12008, any P/T curves for McGuire 1 for the extended period of operation, when submitted to the staff for review and approval, will need to account for all relevant surveillance capsule data for this heat as obtained from the Diablo Canyon 2 RV material surveillance program. The staff will evaluate the extended-period-of-operation P/T limit curves for the McGuire and Catawba RVs prior to expiration of the 40-year, current-operating-term P/T limit curves for the units. The staff's review of the extended-period-of-operation P/T limit curves, when submitted, will ensure that the operations of the RCS for the McGuire and Catawba units will be done in a manner that ensures the integrity of the

RCS during the extended periods of operation for the McGuire and Catawba units as required by 10 CFR 54.21 (c)(1)(ii)."

Subsequent to the publication of NUREG-1772, the current P/T limit curves (TS Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6) were developed and submitted for NRC approval in 2002 (Reference 4). The curves were subsequently approved by the NRC for 34 EFPY (Reference 5).

Duke Energy calculates that MNS Units 1 and 2 will reach 34 EFPY as early as the Spring of 2023. Prior to this, new P/T limit curves will need to be developed, or it must be shown that the current P/T limit curves are applicable for an extended period of time, resulting in an update to the previously mentioned figure descriptions in TS 3.4.3. Extending the applicability date of the existing P/T limit curves is desirable relative to using new P/T limit curves in order to avoid revisions of plant procedures and the retraining of plant operators.

2.4 Description of Proposed Change

The proposed change revises the figure descriptions listed above as follows (changes from the above descriptions in Section 2.2 in **bold**).

- 3.4.3-1, "McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First **54** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"
- 3.4.3-2, "McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First **54** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"
- 3.4.3-3, "McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First **38.6** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"
- 3.4.3-4, "McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First **38.6** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"
- 3.4.3-5, "McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First **54** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"
- 3.4.3-6, "McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First **38.6** EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641"

In addition, the statement in each Figure, "Limiting ART values at 34 EFPY," as shown above in Section 2.2 revises the applicability term statement for the limiting ART values (i.e., Limiting ART values at 34 EFPY) to reflect 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2.

Finally, the proposed change revises the "Criticality Limit" note in the Figures as described in Section 2.2 to reflect an applicability period of 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2.

The proposed change only revises the applicability term associated with the TS figures. The P/T limit curves and limiting ART values remain unchanged. The TS markup provided in Attachment 1 reflects the proposed changes described above.

3 TECHNICAL EVALUATION

In 2013 the NRC approved License Amendments 269 and 249 for MNS Units 1 and 2, respectively (Reference 1). These amendments approved a Measurement Uncertainty Recapture (MUR) Power Uprate, increasing each unit's authorized core power level from 3411 megawatts thermal (MWt) to 3469 MWt, an increase of approximately 1.7% rated thermal power. Westinghouse report WCAP-17455-NP, "McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel and Neutron Fluence Evaluations" was an input into the MUR License Amendment Request (LAR) (Reference 2). WCAP-17455-NP provided reactor vessel (RV) integrity and neutron fluence evaluations for the MUR, in addition to developing new ART values based on the MUR Power Uprate and evaluating them for EFPY applicability. The technical basis for the McGuire Units 1 and 2 P/T limit applicability evaluations is documented in WCAP-17455-NP (Attachment 3 to this letter) and summarized in Section IV.I.C.iii of Enclosure 2 of the MUR LAR (Reference 2). The evaluation concluded that the P/T curves remain valid through 34 EFPY with the MUR power uprate, because the limiting ART values remain applicable up to 54 EFPY (Unit 1) and 38.6 EFPY (Unit 2).

Heatup and cooldown limit curves are calculated using the most limiting value of the reference nil-ductility transition temperature (RTNDT) corresponding to the limiting material in the beltline region of the reactor pressure vessel (RPV). The most limiting RTNDT of the material in the core (beltline) region of the RPV was determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (Δ RTNDT). RTNDT increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RTNDT at any time period in the reactor's life, Δ RTNDT due to the radiation exposure associated with that time period was added to the original unirradiated RTNDT. Using the ART values, P/T limit curves were determined in accordance with the requirements of 10 CFR 50, Appendix G.

The existing P/T limit curves for 34 EFPY in MNS TS 3.4.3 for Units 1 & 2 were originally developed in Westinghouse report WCAP-15192, Revision 2 and WCAP-15201, Revision 2, respectively (Enclosures 4 and 5 of Reference 4, respectively). To confirm the applicability of the P-T limit curves developed in WCAP-15192, Revision 2 for McGuire Unit 1 and in WCAP-15201, Revision 2 for McGuire Unit 2 (Enclosures 4 and 5 of Reference 4, respectively) for the MUR power uprate LAR, the limiting reactor vessel material ART values with consideration of the MUR power uprate were shown in Attachment 3 to be less than the limiting beltline material ART values used in development of the existing 34 EFPY P-T limit curves contained in Reference 4. The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was used along with the surface fluence of Section 2 to calculate ART values for the McGuire Units 1 and 2 reactor vessel materials at 34 EFPY and 54 EFPY. The ART calculations with consideration of the MUR power uprate are summarized in Attachment 3 to this letter, Tables 8.1-1 through 8.1-4 for McGuire Unit 1 and in Tables 8.2-1 through 8.2-4 for McGuire Unit 2.

As shown in Attachment 3, the calculated limiting material ART values are less than those used for the development of the current TS P/T limit curves in Reference 4. The new ART values determined for MUR included re-evaluation of surveillance data credibility, a recalculation of chemistry factors, and the consideration of MUR power uprate fluence projections. The lower ART values are primarily due to the re-evaluation of the Diablo Canyon Unit 2 surveillance weld data utilized in the ART calculations. The extension to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2 is supported by Westinghouse report WCAP-17445-NP (Attachment 3 to this letter). As evidenced in the Attachment 3 analysis, the limiting ART values at 54 EFPY for Unit 1 remain less than the limiting ART values used in the development of the existing Unit 1 TS P/T limit

curves. This means that at 54 EFPY, the existing limiting ART values in Figures 3.4.3-1, 3.4.3-2, and 3.4.3-5 remain conservative and bounding. Therefore, the limiting ART values shown in these figures are not revised by the proposed change. A more detailed discussion is contained in Section 8 of Attachment 3 to this letter.

10 CFR 50, Appendix G requires that P/T limits be developed to bound all ferritic materials in the RPV. Regulatory Issue Summary 2014-11 (Reference 6) clarifies that P/T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may define P/T curves that are more limiting because the consideration of stress levels from structural discontinuities (such as RPV inlet and outlet nozzles) may produce a lower allowable pressure. As part of the response to an NRC condition imposed on the MUR LAR approval, MNS submitted letter MNS-14-034 (Reference 7) to the NRC on May 14, 2014, which described the evaluation of the vessel nozzles and other ferritic reactor coolant pressure boundary (RCPB) components at 54 EFPY. That report concluded that the RV nozzle P/T limits are bounded by the traditional beltline curves.

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements and guidance documents are applicable to the proposed change.

10 CFR 50.36

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications,” establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36(c) TSs will include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) Surveillance Requirements (SRs), (4) design features; and (5) administrative controls.

MNS LCO 3.4.3 limits the pressure and temperature changes during RCS heatup and cooldown (i.e., to the right and below the P/T curves in Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6), to prevent non-ductile RPV failure. The proposed change increases the applicability of the P/T limits to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2, which does impact the function of the LCO to limit RCS operation to within approved P/T limits. Based on the determination that the existing Figures 3.4.3-1, 3.4.3-2, and 3.4.3-5 are applicable up to 54 EFPY, and Figures 3.4.3-3, 3.4.3-4, and 3.4.3-6 are applicable up to 38.6 EFPY (see Section 3 above and Attachment 3), Duke Energy concludes that LCO 3.4.3 will continue to meet 10 CFR 50.36(c)(2)(i) with the proposed change by providing for the requisite functional capability or performance level required for safe operation.

MNS SR 3.4.3.1 verifies that RCS operation is within the limits of Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6 when RCS pressure and temperature conditions are undergoing planned changes. The proposed change increases the applicability of the P/T limits to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2, which does not impact the function of SR 3.4.3.1 to verify that the RCS is operated within approved P/T limits. Therefore, Duke Energy concludes Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6 (with an increase in applicability of the P/T limits) in SR 3.4.3.1 continues to meet 10 CFR 50.36(c)(3) by providing sufficient test, calibration or inspection requirements to

assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits and that the LCO will be met.

MNS TS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System,” addresses the condition when the accumulator is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3, Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6. This includes the Applicability Note in LCO 3.4.12, Condition B and Required Action B.1, and Condition C and Required Action C.2. LCO 3.4.12 provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. The Notes permit accumulator surveillances to be performed when the accumulator pressure does not exceed the P/T limits. Action statements B and C address an improperly isolated accumulator. The proposed change in this LAR increases the applicability of the P/T limit curves to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2, which does not impact the Conditions and Required Actions in LCO 3.4.12. Therefore, Duke Energy concludes the use of the proposed Figures 3.4.3-1, 3.4.3-2, and 3.4.3-5 with increased applicability up to 54 EFPY, and Figures 3.4.3-3, 3.4.3-4, and 3.4.3-6 with increased applicability up to 38.6 EFPY (see Section 3 above and Attachment 3) continues to meet 10 CFR 50.36(c)(2)(i) by providing for the requisite functional capability or performance level required for safe operation.

<u>10 CFR 50.60</u>

Section 50.60 of 10 CFR, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR 50, Appendices G, “Fracture Toughness Requirements,” and H, “Reactor Vessel Material Surveillance Program Requirements.” The proposed change only increases the applicability term of the P/T limit curves to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2 and does not propose to revise any other aspects of the limit curves. The existing curves will meet the 10 CFR 50.60 requirements up to 54 EFPY and 38.6 EFPY. Therefore, the proposed change to increase the applicability terms to 54 EFPY and 38.6 EFPY is consistent with the requirements of 10 CFR 50.60.

<u>10 CFR 50, Appendix G</u>

Appendix G to 10 CFR 50 requires that the P/T limits for the facility’s reactor pressure vessel (RPV) be at least as conservative as those obtained by following the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The P/T limits in TS 3.4.3 are unchanged. Only the applicability term is proposed to be increased to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2 instead of 34 EFPY. Therefore, Duke Energy concludes for the proposed change that the MNS Units 1 and 2 RPV will continue to meet RPV integrity regulatory requirements through 54 EFPY and 38.6 EFPY, respectively.

<u>10 CFR 50, Appendix H</u>

Appendix H to 10 CFR 50 establishes requirements for each facility related to its RPV material surveillance. These regulatory requirements will continue to be met for the proposed change with the surveillance capsule removal schedule prescribed in Section 9 of Attachment 3 to this submittal.

Regulatory Guide (RG) 1.99, Revision 2

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," contains guidance on methodologies the NRC considers acceptable for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. This RG was used for the calculation of ART values at the 1/4T and 3/4T locations that were previously calculated in Reference 4 and are shown on the existing MNS P/T limit curves. The proposed change does not alter the existing ART values in Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6. The proposed change only increases the applicability term to 54 EFPY for Unit 1 and 38.6 EFPY for Unit 2. Therefore, the proposed change has no effect on the application of RG 1.99, Revision 2.

RG 1.190

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001, describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence. The analysis methodologies used to calculate the MNS RPV fluences for the MUR LAR associated with the proposed change satisfy the guidance set forth in RG 1.190.

The proposed change does not affect plant compliance with any of the above regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 Precedent

Entergy submitted a LAR for Palisades Nuclear Plant on March 7, 2011 (Reference 9), which extended the applicability of the P/T limit curves for TS Figures 3.4.3-1 and 3.4.3-2 in addition to the LTOP setpoint limit curve in Figure 3.4.12-1. The Palisades LAR was also based on a Westinghouse applicability evaluation. The NRC approved the LAR in a Safety Evaluation dated January 19, 2012 (Reference 10)

Duke Energy submitted a LAR for the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 in 2018 which reduced the applicability of the P/T limit curves in TS Figures 3.4.3-1 and 3.4.3-2 from 50 EFPY to 46.3 EFPY due to the removal of part length shield assemblies (PLSAs) and migration to 24-month cycles (Reference 11). Although the LAR sought approval for a reduction of the EFPY applicability, it was also based on a Westinghouse applicability evaluation which was provided along with the submittal. Like this LAR, the P/T curves and ART values were unchanged, only the applicability of the existing curves. The NRC approved the LAR in a Safety Evaluation dated August 16, 2018 (Reference 12).

4.3 Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC (Duke Energy) hereby requests an amendment to revise the Technical Specifications (TS) for McGuire Nuclear Station (MNS), Units 1 and 2. The proposed change revises MNS TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," to reflect that Figures 3.4.3-1, 3.4.3-2 and 3.4.3-5 (Unit 1 P/T limit curves) are applicable up to 54 effective full power years (EFPY) and that Figures 3.4.3-3, 3.4.3-4 and 3.4.3-6 (Unit 2 P/T limit curves) are applicable up to 38.6 EFPY.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   No changes are being made to the existing P/T limit curves in TS 3.4.3 Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6. The P/T limit curves are only being revised to change the applicability period of 54 EFPY (Unit 1) and 38.6 EFPY (Unit 2).

   The changes to the TS figures are applicable to normal plant operations and do not influence the probability of occurrence or safety analysis considerations for design basis accidents. Consequently, there will be no change to the probability or consequences of accidents previously evaluated. Operating the facility in accordance with the P/T limit curves ensures that stresses caused by the thermal gradient through the Reactor Pressure Vessel (RPV) beltline material remain bounded by the stress analyses. The proposed amendment does not involve operation of required structures, systems, or components in a manner or configuration different than previously recognized or evaluated. No radiological barriers are affected by the change.

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

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

   Response: No

   No changes are being made to the existing P/T limit curves in TS 3.4.3 Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6. The P/T limit curves are only being revised to change the applicability period of 54 EFPY (Unit 1) and 38.6 EFPY (Unit 2).

   The change does not involve a modification of plant structures, systems, or components. The change will not affect the manner in which the plant is operated and will not degrade the reliability of structures, systems, or components. Equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, and supporting system performance will not be affected. No new failure modes or mechanisms will be introduced as a result of this proposed change.

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

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

   Response: No

   No changes are being made to the existing P/T limit curves in TS 3.4.3 Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.3-4, 3.4.3-5, and 3.4.3-6. The P/T limit curves are only

being revised to change the applicability period of 54 EFPY (Unit 1) and 38.6 EFPY (Unit 2).

The proposed amendment does not involve: 1) a physical alteration of the plant, 2) a change to any set points for parameters associated with protection or mitigation actions nor 3) any adverse impact on the fission product barriers or parameters associated with licensed safety limits. There are no changes to either the containment analysis or to the analysis for any design basis event.

Appendix G to 10 CFR 50 describes the conditions that require P/T limits and provides the general bases for these limits. Operating limits based on the criteria of Appendix G, as defined by applicable regulations, codes standards, provide reasonable assurance that non-ductile or rapidly propagating failure will not occur. The P/T limits are prescribed for all plant modes to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary. Calculation of P/T limits in accordance with the criteria of Appendix G to 10 CFR 50 and applicable regulatory requirements ensures that adequate margins of safety are maintained and there is no significant reduction in a margin of safety.

No change is being made to the existing P/T limit curves, only the applicability period associated with the P/T limits is being extended. Since the P/T limits and limiting ART values remain unchanged there is no reduction in a margin of safety.

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined.
There is no change or impact on any safety analysis assumption or on any other parameter affecting the course of an accident analysis supporting the basis of any Technical Specification. The proposed change does not involve any increase in calculated off-site dose consequences.

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

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 the proposed revision to the figures in TS 3.4.3, (2) the proposed revision will be implemented in a manner consistent with the Commission’s regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical

exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6 REFERENCES

1. McGuire Nuclear Station, Units 1 and 2, “Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC ME8213 and 8214),” dated May 16, 2013 (ADAMS Accession No. ML13073A041).
2. Letter from R. Repko to USNRC, “License Amendment Request for Measurement Uncertainty Recapture Power Uprate,” dated March 5, 2012 (ADAMS Accession No. ML12082A210).
3. NUREG 1772, “Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 & 2,” dated March 31, 2003 (ADAMS Accession No. ML030830181)
4. Letter from D.M. Jamil to the USNRC, “License Amendment Request, TS Changes to TS 3.3.2, TS 3.4.3, TS 3.4.12,” dated December 12, 2002 (ADAMS Accesion No. ML023610223).
5. Safety Evaluation Report, “McGuire Nuclear Station, Units 1 and 2, License Amendment, Revise Technical Specification for Vessel Pressure-Temperature Limits Curves and Revise Low-Temperature Overpressure Protection Limits,” dated July 3, 2003 (ADAMS Accession No. ML031780107).
6. NRC Regulatory Issue Summary 2014-11, “Information on Licensing Application for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” dated October 14, 2014 (ADAMS Accession No. ML14149A165).
7. MNS-14-034, Letter from S.D. Capps to the USNRC, “McGuire Units 1 and 2 Pressure – Temperature Limit Curve Analyses Required by License Conditions Issued as Part of Measurement Uncertainty Recapture License Amendments (TAC NOS. ME8213 and ME8214),” dated May 14, 2014 (Not Publicly Available).
8. PNP 2011-016, “License Amendment Request for Primary Coolant System Pressure-Temperature Limits,” dated March 7, 2011 (ADAMS Accession No. ML110730082).
9. “Palisades Nuclear Plant – Issuance of Amendment re: Primary Coolant System Pressure-Temperature Limits,” dated January 19, 2012 (ADAMS Accession No. ML113480303).
10. RNP-RA/17-0082, “H.B. Robinson Steam Electric Plant, Unit 2 – License Amendment Request Proposing to Revise Technical Specification 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” dated February 07, 2018 (ADAMS Accession No. ML18038B289).
11. “Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendment No. 260 Regarding Request to Revise Technical Specification Reactor Coolant System Pressure and Temperature Limits to Reflect 24-Month Fuel Cycles (EPID L-2018-LLA-0033),” dated August 16, 2018 (ADAMS Accession No. ML18200A042).
12. Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” dated May, 1988.

Attachment 1

Technical Specification 3.4.3 Markup

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT ~~34~~ 54 EFPY: 1/4T, 202°F
3/4T, 146°F

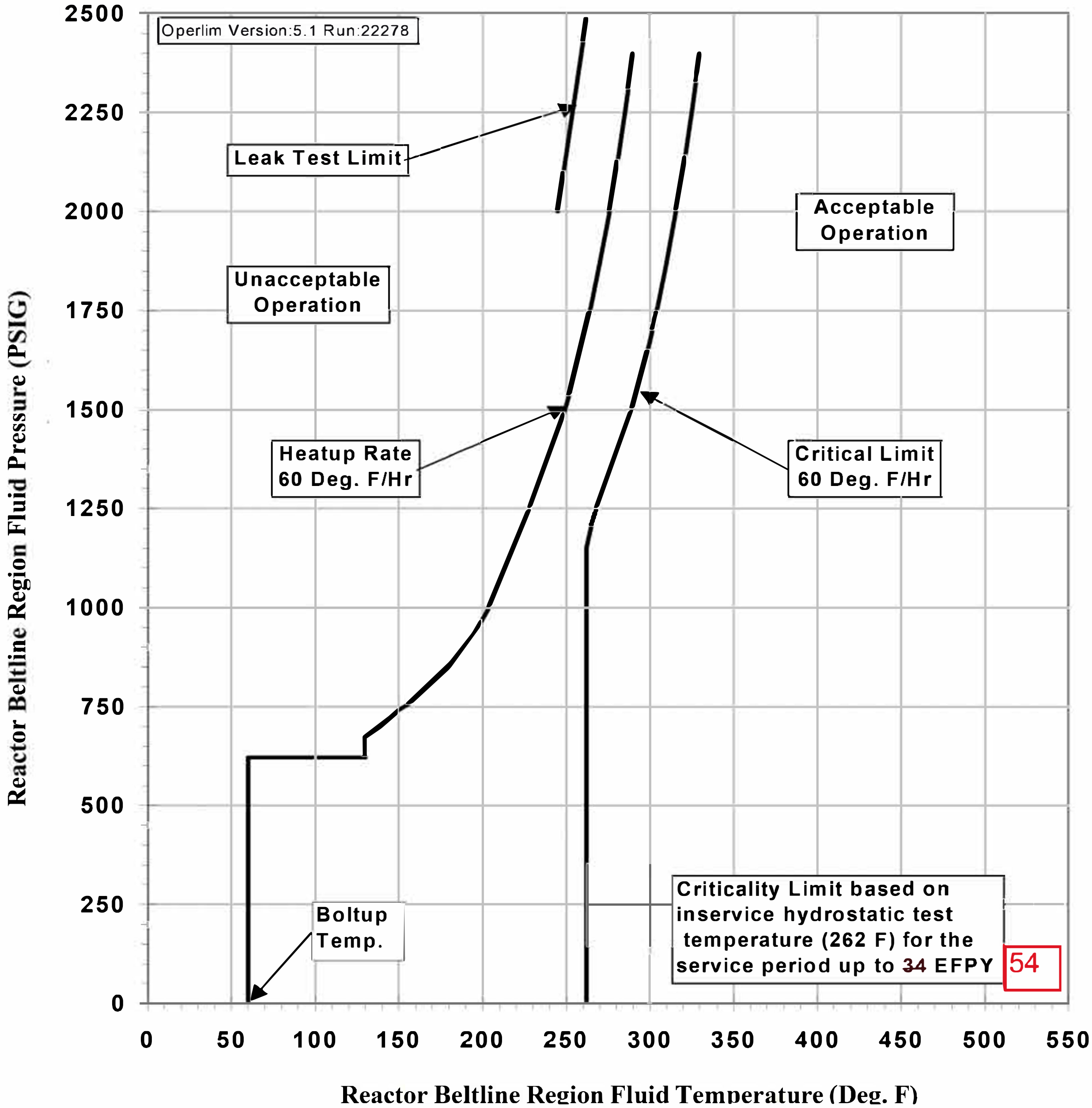


Figure 3.4.3-1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First ~~34~~ 54 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

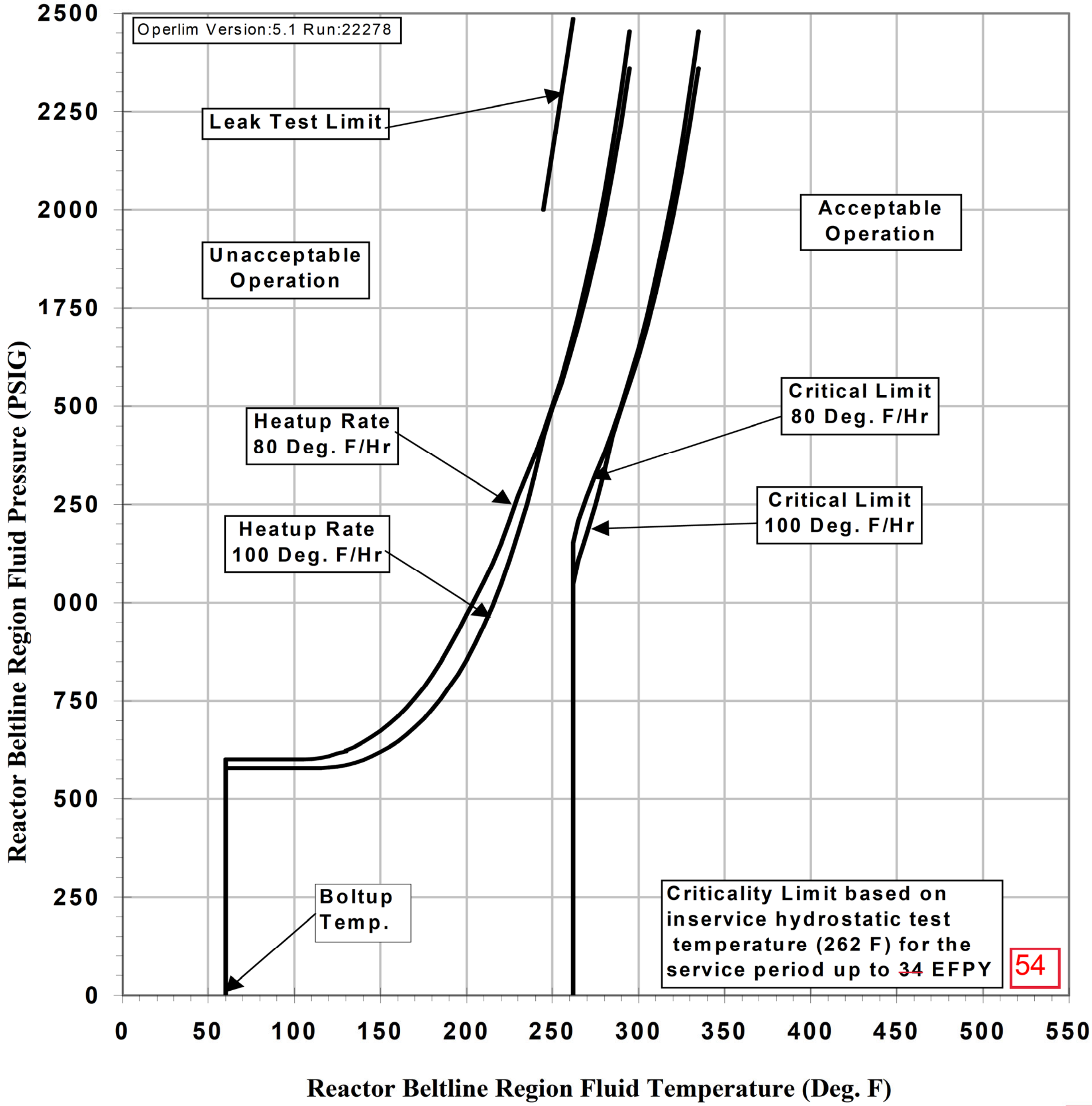
MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD

LIMITING ART VALUES AT ~~34~~ EFPY: 1/4T, 202°F

3/4T, 146°F

54



54

Figure 3.4.3-2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First ~~34~~ EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART VALUES AT ~~34~~ 38.6 EFPY: 1/4T, 123°F

3/4T, 91°F

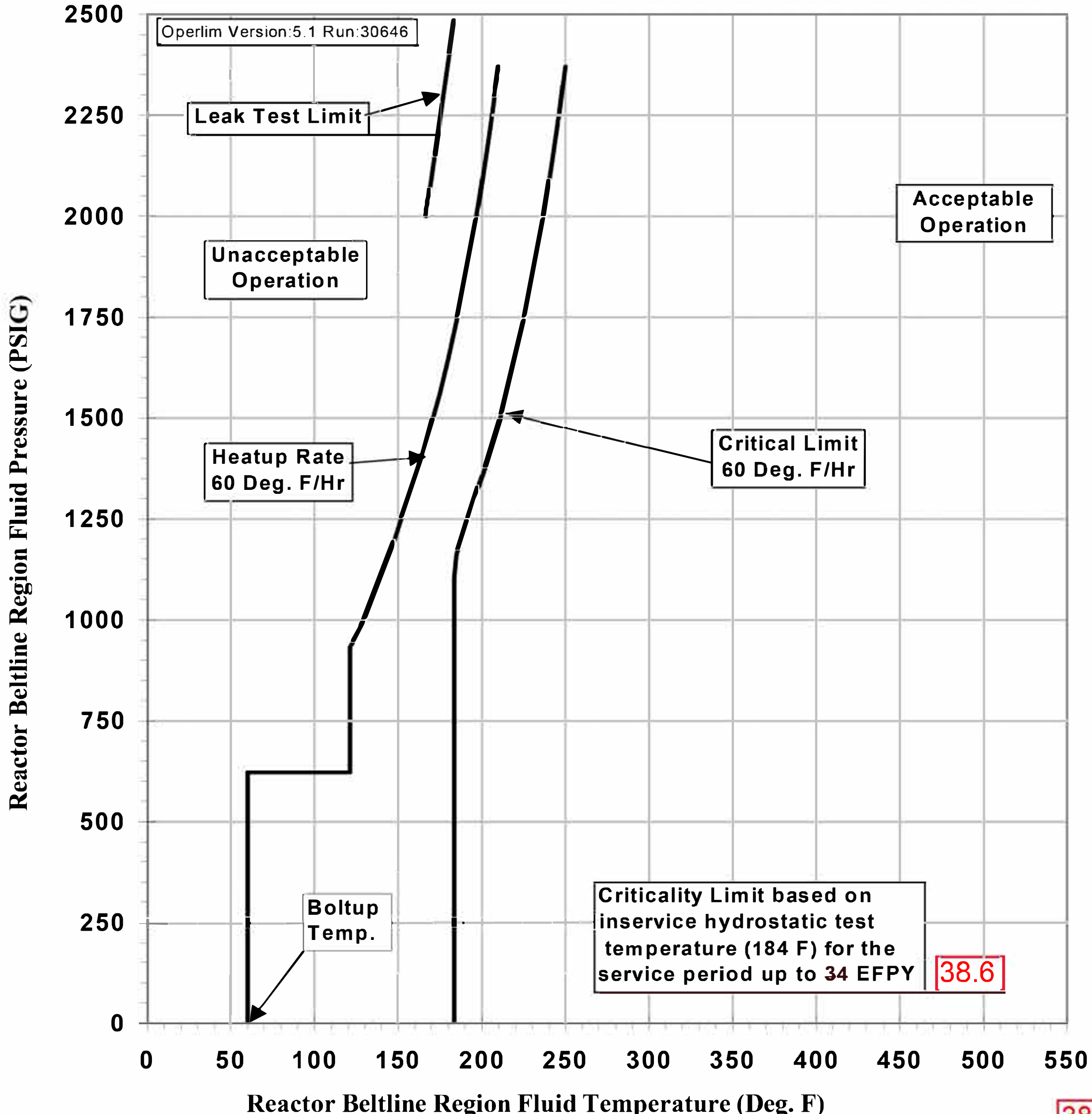


Figure 3.4.3-3 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First ~~34~~ 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART VALUES AT ~~34~~ 38.6 EFPY: 1/4T, 123°F

3/4T, 91°F

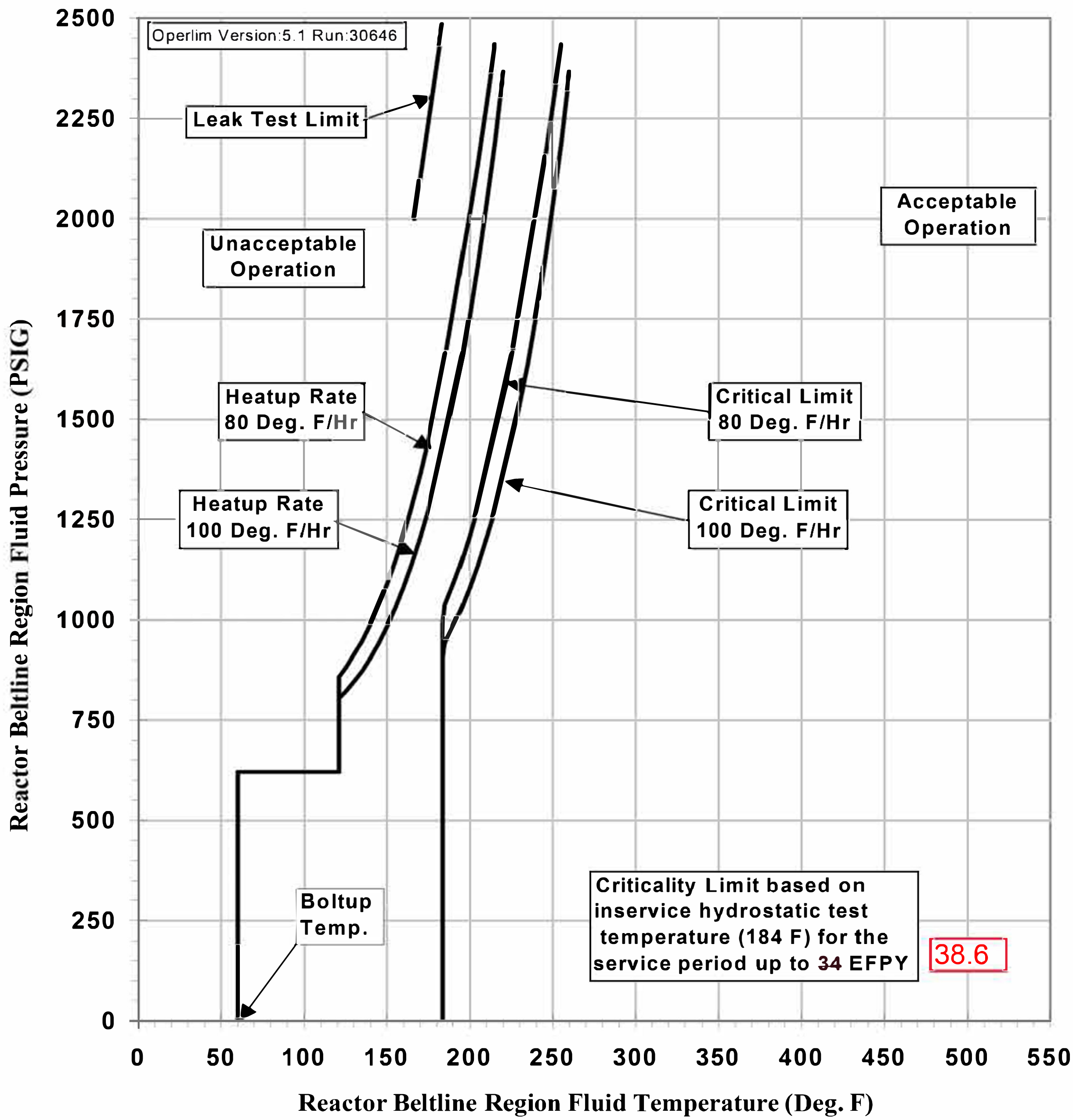


Figure 3.4.3-4 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First ~~34~~ 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT ~~34~~ 54 EFPY: 1/4T, 202°F
3/4T, 146°F

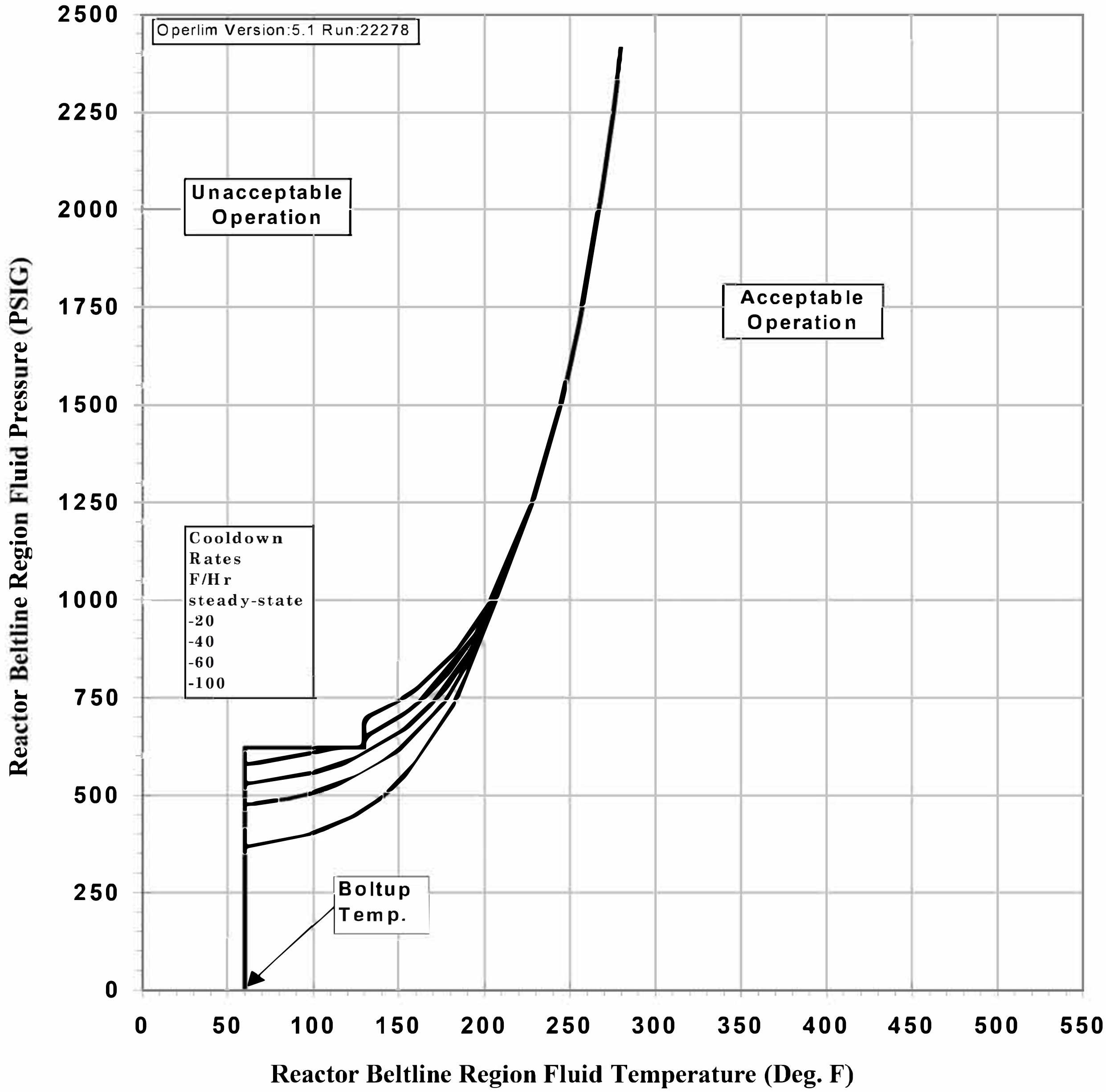


Figure 3.4.3-5 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First ~~34~~ 54 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

<u>MATERIAL PROPERTY BASIS</u>

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART VALUES AT ~~34~~ 38.6 EFPY: 1/4T, 123°F

3/4T, 91°F

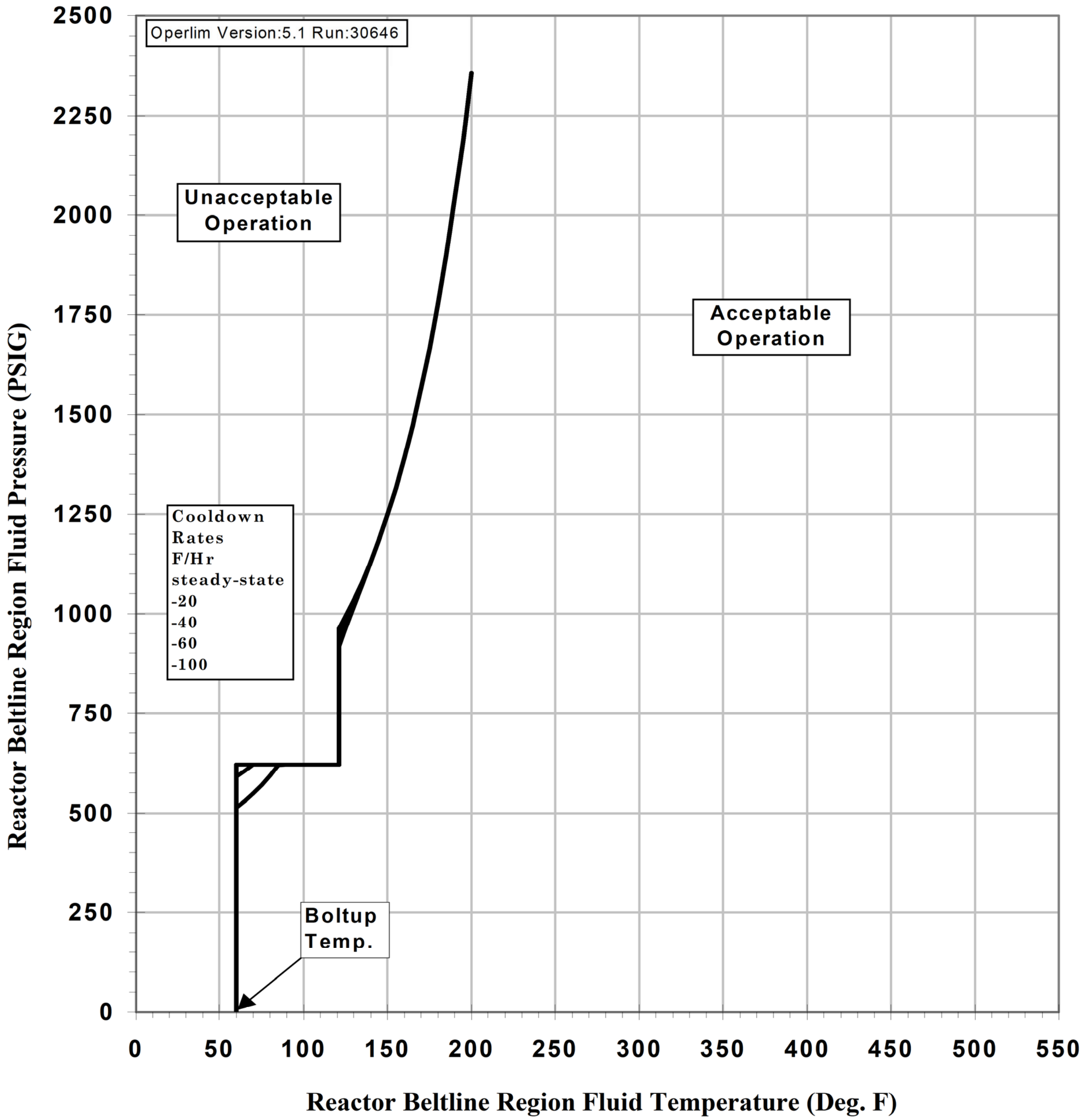


Figure 3.4.3-6 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First ~~34~~ 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

Attachment 2

Technical Specification 3.4.3 Retyped Pages

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD

LIMITING ART VALUES AT 54 EFPY:	1/4T,	202°F
	3/4T,	146°F

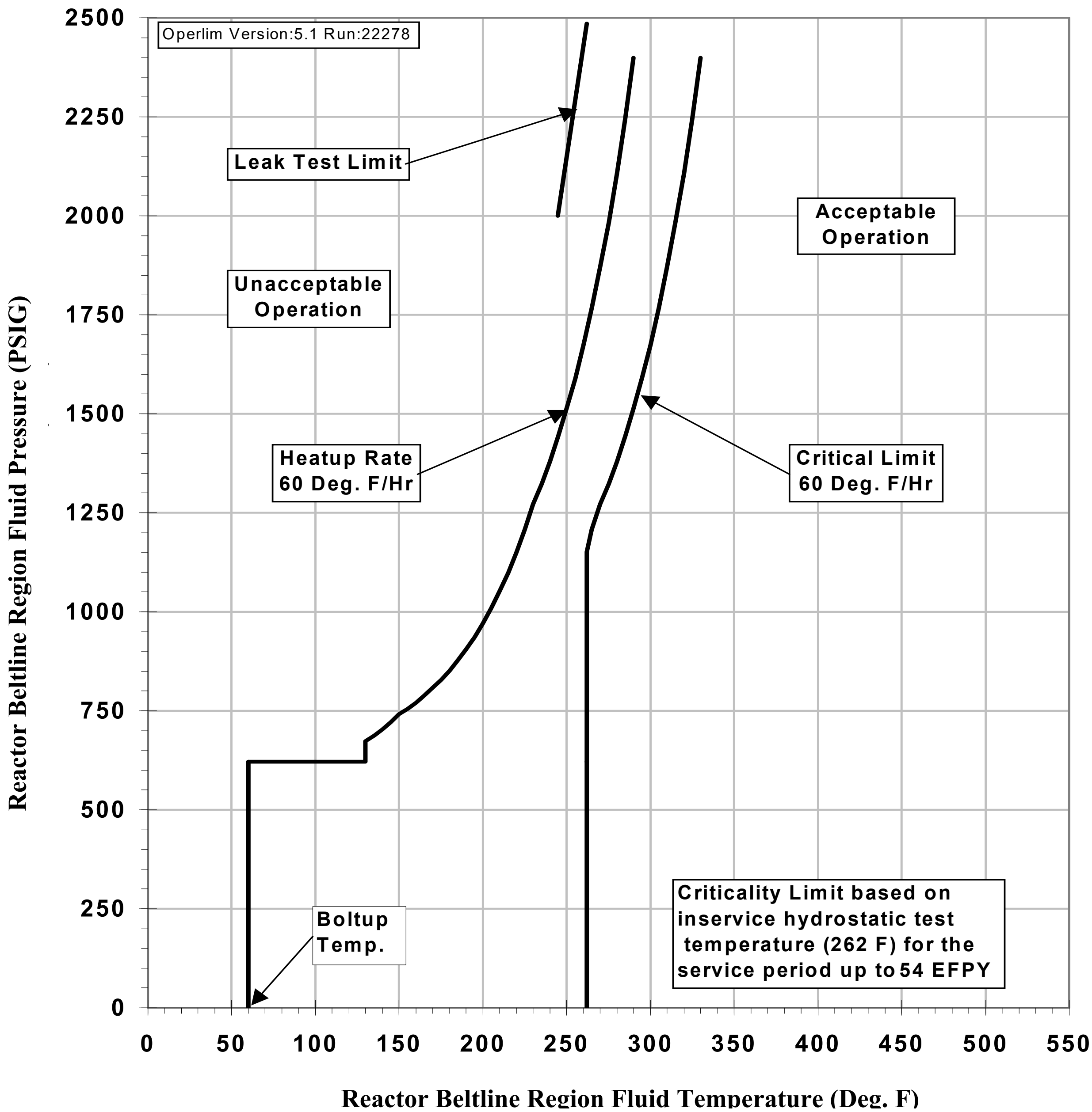


Figure 3.4.3-1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 54 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 54 EFPY: 1/4T, 202°F
3/4T, 146°F

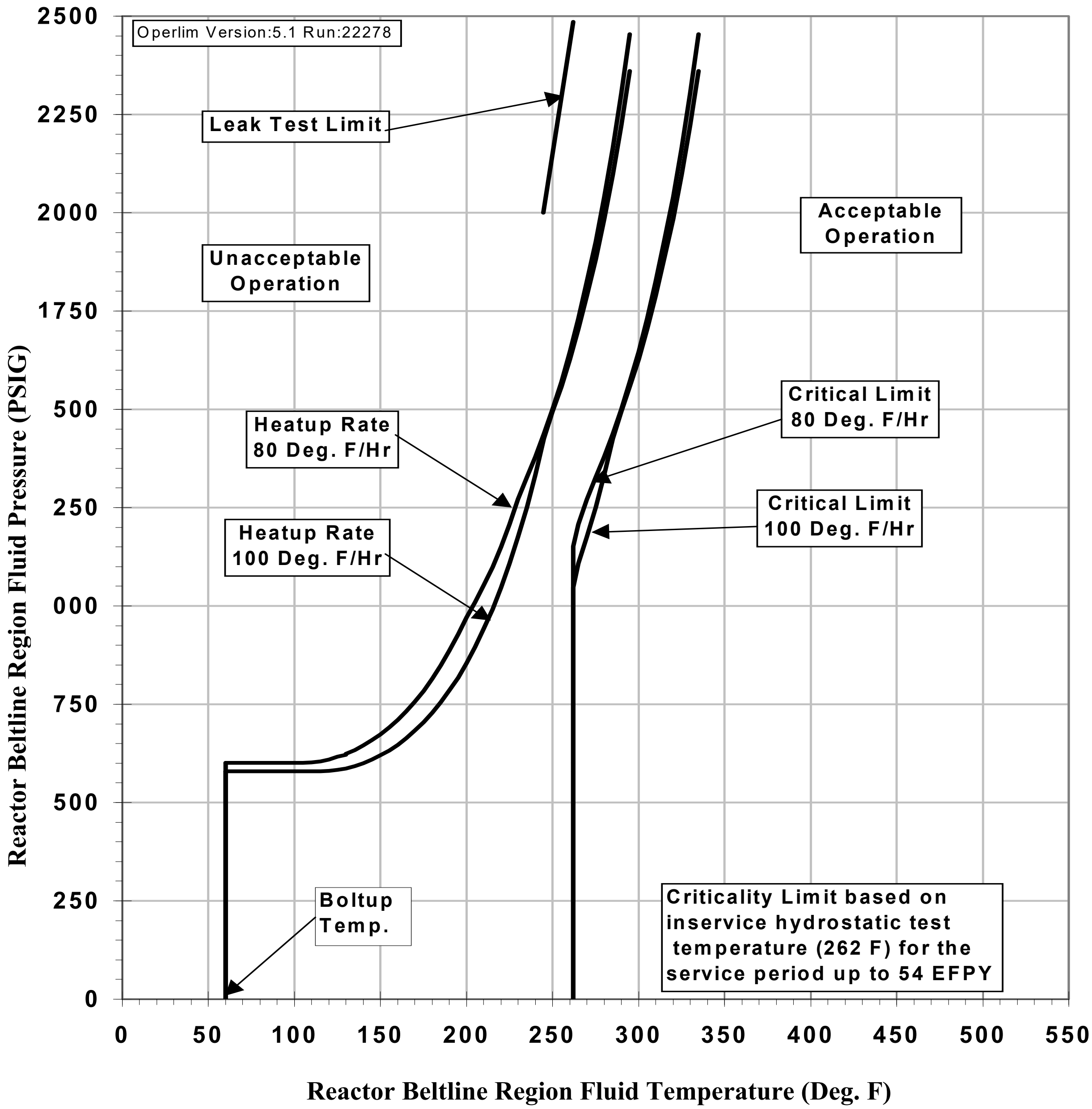


Figure 3.4.3-2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 54 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART VALUES AT 38.6 EFPY: 1/4T, 123°F
3/4T, 91°F

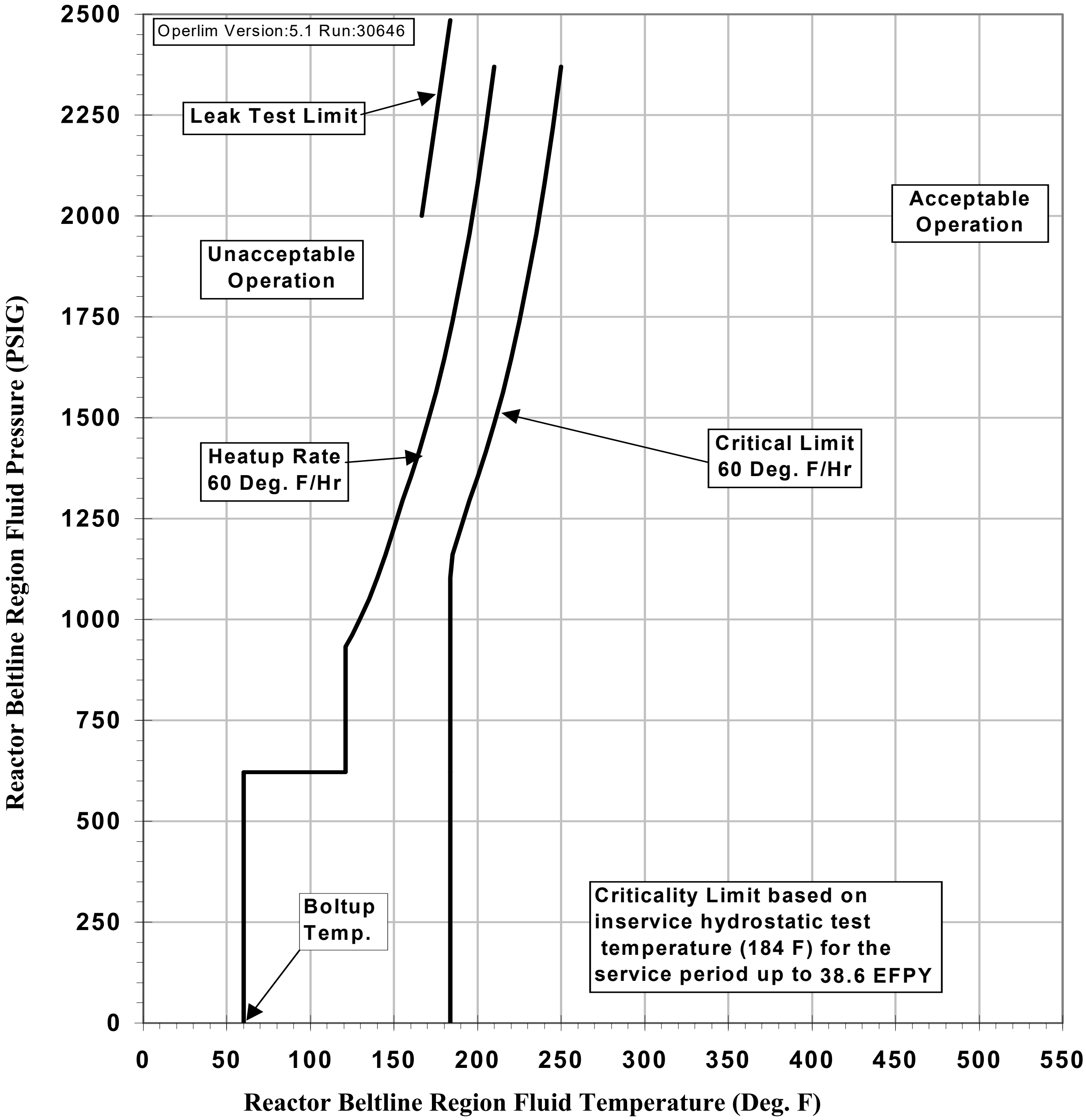


Figure 3.4.3-3 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 38.6 EFPY: 1/4T, 123°F
3/4T, 91°F

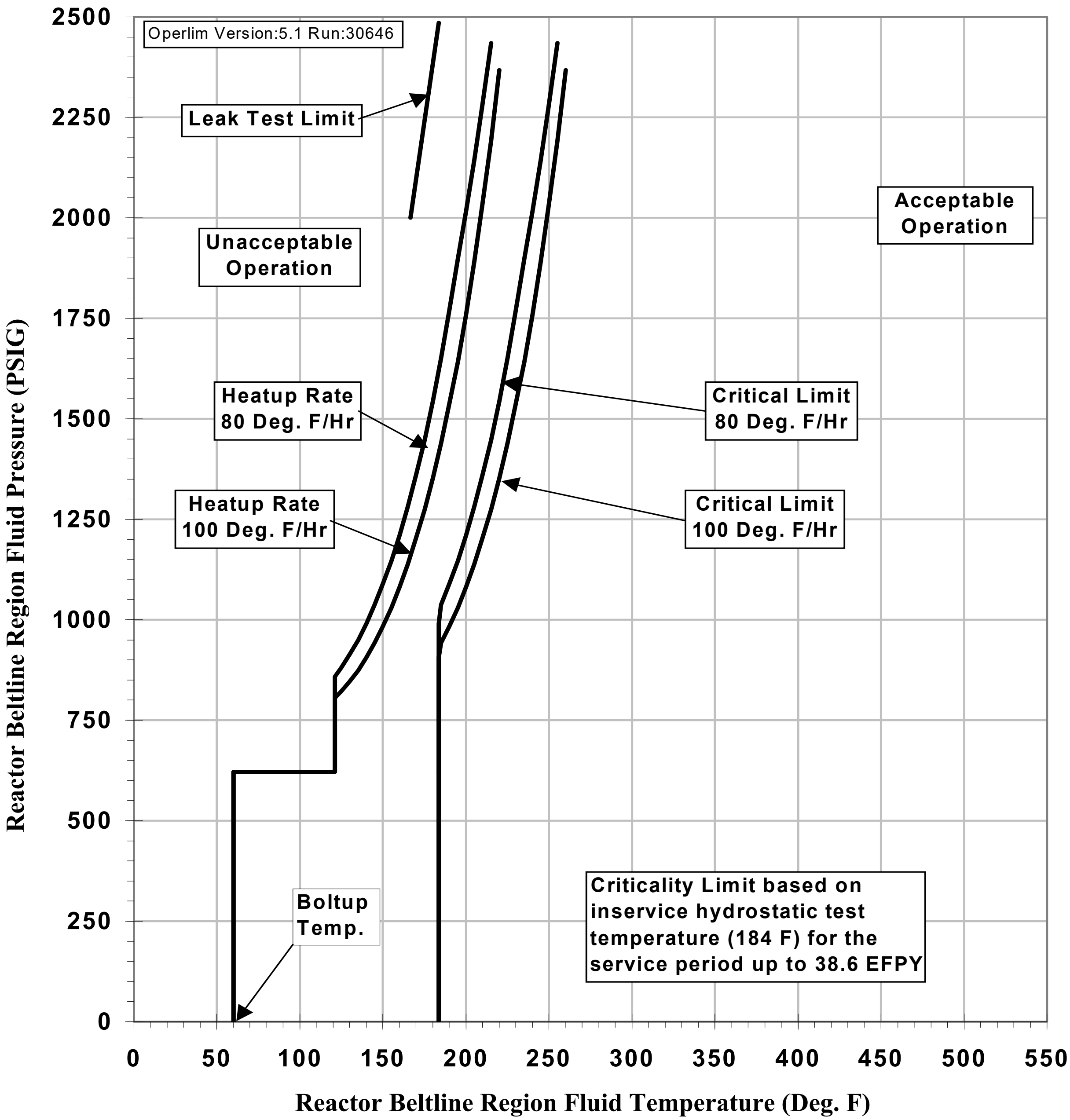


Figure 3.4.3-4 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 54 EFPY: 1/4T, 202°F
3/4T, 146°F

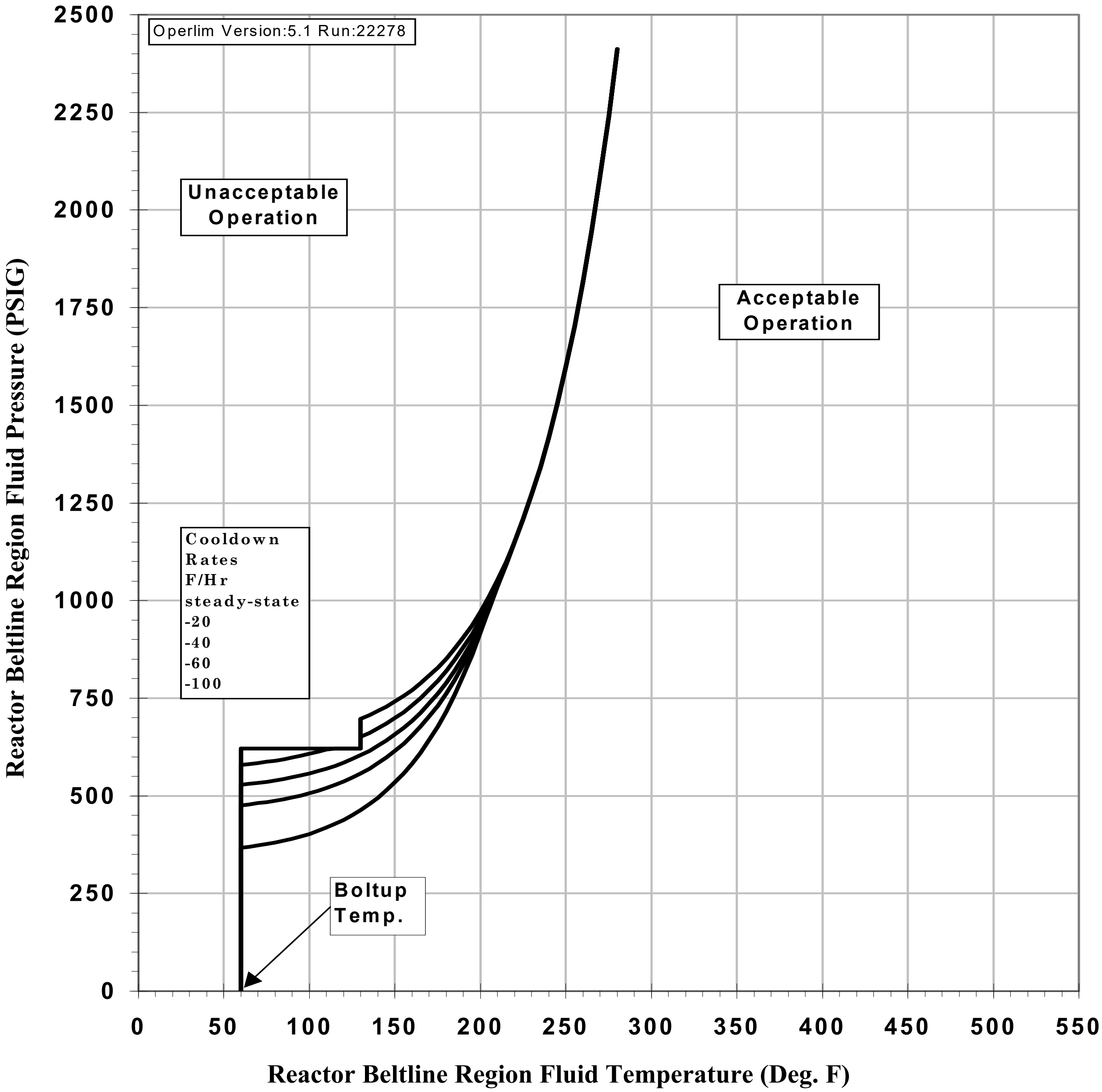


Figure 3.4.3-5 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 54 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 38.6 EFPY: 1/4T, 123°F
3/4T, 91°F

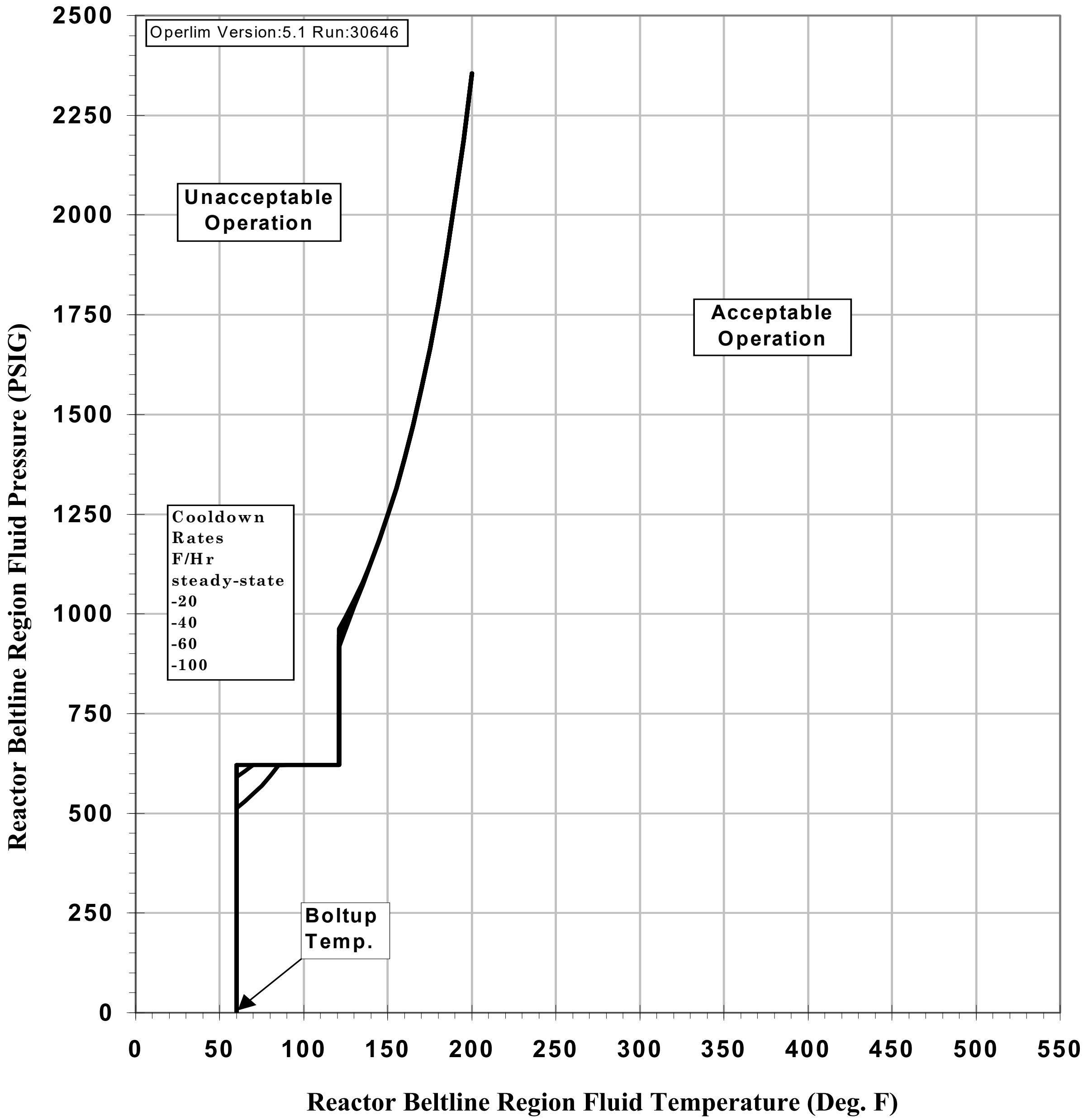


Figure 3.4.3-6 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 38.6 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

Attachment 3
WCAP-17455-NP, “McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations”

WCAP-17455-NP
Revision 0

February 2012

McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations

Westinghouse

WCAP-17455-NP
Revision 0

McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations

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Aging Management and License Renewal Services

Jianwei Chen*
Radiation Engineering and Analysis

February 2012

Reviewer: Amy E. Freed*
Aging Management and License Renewal Services

Reviewer: Melissa A. Hunter*
Radiation Engineering and Analysis

Approved: Michael G. Semmler*, Acting Manager
Aging Management and License Renewal Services

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

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EXECUTIVE SUMMARY

This report presents the reactor vessel (RV) integrity and neutron fluence evaluations for the McGuire Units 1 and 2 measurement uncertainty recapture (MUR) power uprate to 3479 MWt. The RV integrity evaluations must be shown to meet the applicable U.S. Nuclear Regulatory Commission (NRC) requirements through the end of the licensed operating period. McGuire Units 1 and 2 are each licensed for 60 years of operation, which pertains to 54 effective full power years (EFPY) and is deemed end-of-life extension (EOLE).

Appendix A contains the credibility evaluation for the McGuire Units 1 and 2 surveillance materials. Conclusions for the surveillance data credibility evaluations are contained in Appendix A of this report.

Appendix B contains the Emergency Response Guideline (ERG) limits classification for McGuire Units 1 and 2. The ERG limits were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Conclusions for the ERG limits evaluations are contained in Appendix B of this report.

The conclusions to the RV integrity evaluations are as follows:

EOLE Pressurized Thermal Shock (PTS)

All of the McGuire Units 1 and 2 reactor vessel materials are projected to remain below the 10 CFR 50.61 screening criteria values of 270°F for axially oriented welds and plates / forgings, and 300°F for circumferentially oriented welds, through EOLE (54 EFPY). See Section 6 for more details.

EOLE Upper-Shelf Energy (USE)

All of the McGuire Units 1 and 2 reactor vessel materials are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOLE (54 EFPY). See Section 7 for more details.

Applicability of Pressure-Temperature (P-T) Limit Curves

The current 34 EFPY McGuire Units 1 and 2 P-T limit curves are contained in Technical Specifications Figures 3.4.3-1 through 3.4.3-6. With a re-evaluation of surveillance data credibility, a recalculation of chemistry factors, and the consideration of MUR power uprate fluence projections, the applicability of the P-T limit curves may either remain unchanged or be extended. See Section 8 for more details.

Surveillance Capsule Withdrawal Schedules

All in-vessel surveillance capsules have been removed from the McGuire Units 1 and 2 reactor vessels. The guidelines of ASTM E185-82 are met, as required by 10 CFR 50, Appendix H, with consideration of the MUR power uprate. Ex-Vessel Neutron Dosimetry is installed in McGuire Units 1 and 2 such that neutron fluence may be monitored during the period of extended operation in accordance with the Generic Aging Lessons Learned Report. See Section 9 for more details.

1 METHOD DISCUSSION

Adjusted Reference Temperature (ART)

Per Regulatory Guide 1.99, Revision 2 (Reference 1), the following equations and variables are to be used for calculating ART values at the clad/base metal interface and at the reactor vessel 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations.

ART (°F) = Initial RT_{NDT} + ΔRT_{NDT} + Margin [Eqn. 1]

Where,

Initial RT_{NDT} (°F) = Reference temperature of the unirradiated material

Margin (°F) $= 2\sqrt{\sigma_I^{\,2} + \sigma_\Delta^{\,2}}$ [Eqn. 2]

Where,

σ_I is the standard deviation for the Initial RT_{NDT} (note that σ_I is referred to as σ_U in 10 CFR 50.61). If the initial RT_{NDT} is a measured value, σ_I is estimated from the precision of the test method; per WCAP-14040-A, Revision 4 (Reference 2), σ_I = 0°F when the initial RT_{NDT} is a measured value. Per 10 CFR 50.61 (Reference 3), when the initial RT_{NDT} is not a measured value and a generic mean initial RT_{NDT} value is used for welds with the welding flux types identified in 10 CFR 50.61, then σ_I = 17°F.

σ_Δ is the standard deviation for ΔRT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not credible or not used*

σ_Δ = 8.5°F when credible surveillance capsule data is used*

For welds:

σ_Δ = 28°F when surveillance capsule data is not credible or not used*

σ_Δ = 14°F when credible surveillance capsule data is used*

* σ_Δ not to exceed 0.5*ΔRT_{NDT} per Regulatory Guide 1.99, Revision 2

Shift in Reference Temperature (ΔRT_{NDT}) Calculations:

ΔRT_{NDT} (°F) = CF * FF [Eqn. 3]

Where,

CF (°F) = chemistry factor based on the copper (Cu) and nickel (Ni) weight % of the material or based on the results of surveillance capsule test data. If the weight percent of Cu and Ni is used to determine the CF, then the CF is obtained from Table 1 or Table 2 of Regulatory Guide 1.99, Revision 2. If surveillance capsule data is used to determine the CF, then the CF is determined as follows:

$$CF = \frac{\sum_{i=1}^{n}[A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n}[f_i^{(0.56-0.20\log f_i)}]} \qquad \text{[Eqn. 4]}$$

Where:

n = The number of surveillance data points

A_i = The measured value of ΔRT_{NDT} (°F)**

f_i = fluence for each surveillance data point (x10^{19} n/cm^2 (E > 1.0 MeV))

**If the surveillance weld copper and nickel content differs from that of the vessel weld, then the measured values of ΔRT_{NDT} (A_i in the preceding equation for CF) shall be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld (CF_{VW}) to that for the surveillance weld (CF_{SW}) based on the Cu and Ni content of the materials.

ΔRT_{NDT} (°F) = (measured ΔRT_{NDT}) * (CF_{VW} / CF_{SW}) [Eqn. 5]

FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$ [Eqn. 6]

Where,

f = Vessel inner wall surface fluence, 1/4T fluence, or 3/4T fluence [x10^{19} n/cm^2 (E > 1.0 MeV)]. The neutron fluence at any depth in the vessel wall is calculated as follows:

f (x10^{19} n/cm^2 (E > 1.0 MeV)) = $f_{surf} * e^{-0.24*(x)}$ [Eqn. 7]

Where,

f_{surf} = Vessel inner wall surface fluence, x10^{19} n/cm^2 (E > 1.0 MeV)

x = The depth into the vessel wall from the inner surface, inches

Upper-Shelf Energy (USE)

The predicted decrease in USE is determined as a function of fluence and copper content using either of the following:

- Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 1), Position 1.2, or
- Surveillance program test results and Figure 2 of Regulatory Guide 1.99, Revision 2, Position 2.2.

Both methods require the use of the 1/4 thickness (1/4T) vessel fluence.

Reactor Vessel/Core Inlet Temperature (T_{cold})

Regulatory Guide 1.99, Revision 2, Position 1.3 identifies limitations of applicability for the calculations of reference temperature and upper-shelf energy. Nominal irradiation temperature is one of the limitations, wherein Regulatory Guide 1.99 indicates that the nominal irradiation temperature for which the procedures are valid is 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation over 590°F may be considered to produce less embrittlement.

It is concluded that McGuire Units 1 and 2 operate within the 525°F and 590°F range. Thus, the Regulatory Guide 1.99, Revision 2 correlations are applicable.

Pressurized Thermal Shock (PTS)

The PTS Rule, 10 CFR 50.61 (Reference 3), requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected pressurized thermal shock reference temperature (RT_{PTS}) values accepted by the U.S. NRC for each reactor vessel beltline material at the end-of-life (EOL) fluence of the plant. This assessment must specify the basis for the projected value of RT_{PTS} for each vessel beltline material, including the assumptions regarding core-loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

Per 10 CFR 50.61 (Reference 3), the following equations and variables are to be used for calculating RT_{PTS} values at the clad/base metal interface of the vessel. RT_{PTS} is also referred to as the EOL RT_{NDT} (reference nil-ductility transition temperature).

$$\mathbf{RT_{PTS}\ (°F) = IRT_{NDT} + M + \Delta RT_{NDT}} \qquad \text{[Eqn. 8]}$$

Where,

IRT_{NDT} (°F) = Initial Unirradiated RT_{NDT} value

$$\mathbf{M = Margin\ (°F)} = 2\sqrt{\sigma_U{}^2 + \sigma_\Delta{}^2} \qquad \text{[Eqn. 9]}$$

Where,

σ_U = 0°F when Initial RT_{NDT} is a measured value

σ_U = 17°F when Initial RT_{NDT} is a generic value

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not credible or not used***

σ_Δ = 8.5°F when credible surveillance capsule data is used***

For welds:

σ_Δ = 28°F when surveillance capsule data is not credible or not used***

σ_Δ = 14°F when credible surveillance capsule data is used***

*** σ_Δ not to exceed 0.5*ΔRT_{NDT} per 10 CFR 50.61 (Reference 3)

ΔRT_{NDT} (°F) = CF * FF [Eqn. 10]

Where,

CF = chemistry factor (°F) calculated generically for copper (Cu) and nickel (Ni) content based on Tables 1 and 2 in Reference 3 for welds and plates, respectively (also referred to as Position 1.1). It can also be calculated using credible surveillance capsule data per Equation 5 of Reference 3 (also referred to as Position 2.1).

FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$, where the normalized neutron fluence at the clad/base metal interface on the inside surface of the vessel is $f = \Phi / (1.0 \times 10^{19})$. The units for Φ are n/cm^2, E > 1.0 MeV.

The RT_{PTS} screening criteria values are 270°F for plates, forgings and axial weld materials and 300°F for circumferential weld materials. All available surveillance data must be considered in the evaluation.

Surveillance Capsule Withdrawal Schedule

Per ASTM E185-82 (Reference 4), Section 4.15, the ΔRT_{NDT} or adjustment in reference temperature is "the difference in the 41 J (30 ft-lb_f) index temperatures from the average Charpy curves measured before and after irradiation."

Per ASTM E185-82, Section 4.18, the USE level is "the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper-shelf energy."

The surveillance capsule withdrawal schedule is generated based upon the guidelines specified in ASTM E185-82, Section 7.6. The minimum recommended number of surveillance capsules and their withdrawal times are identified in Table 1-1.

Table 1-1 Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule (Schedule in Terms of Effective Full Power Years of the Reactor Vessel)

Predicted Transition Temperature Shift at Vessel Inside Surface	**≤ 100°F**	**> 100°F & ≤ 200°F**	**> 200°F**
Minimum Number of Capsules	3	4	5
Withdrawal Sequence	**EFPY (Effective Full Power Years)**		
First	6[(a)]	3[(a)]	1.5[(a)]
Second	15[(b)]	6[(c)]	3[(d)]
Third	EOL[(e)]	15[(b)]	6[(c)]
Fourth	- -	EOL[(e)]	15[(b)]
Fifth	- -	- -	EOL[(e)]

Notes:

(a) Or at the time when the accumulated neutron fluence of the capsule exceeds 5 x 10^{18} n/cm^2, or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is approximately 50°F, whichever comes first

(b) Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first

(c) Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel 1/4T location, whichever comes first

(d) Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules

(e) Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates S_N transport analysis was performed for the McGuire Units 1 and 2 reactors to determine the neutron radiation environment within the reactor pressure vessels. In this analysis, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. An evaluation of the most recent dosimetry sensor set from Capsule W in McGuire Unit 1, withdrawn at the end of the eighteenth plant operating cycle, is provided in Reference 5. The dosimetry analysis documented in Reference 5 showed that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190 (Reference 6) is met. The results of this analysis are consistent with those of Reference 5 within the uncertainty of the methodology. Therefore, the acceptance criterion continues to be met for McGuire Unit 1. Because the axial power distributions for Cycles 13 through 18 are updated in the current analysis, dosimetry sets from Capsule W withdrawn from McGuire Unit 1 at end of Cycle 18 were re-analyzed to validate the neutron transport calculations for McGuire Unit 1 beyond Cycle 13. For McGuire Unit 2, the neutron dosimetry sensor sets removed from the six previously withdrawn surveillance capsules [V, X, U, Y, Z, and W] were re-analyzed using the current dosimetry evaluation methodology and updated neutron transport calculations. These dosimetry evaluations were used to validate the plant-specific neutron transport calculations applicable to McGuire Units 1 and 2 and are described in Appendix C. These validated calculations form the basis for providing projections of the neutron exposure of the reactor pressure vessels for operating periods extending to 60 EFPY.

All of the calculations described in this section were based on nuclear cross-section data derived from ENDF/B-VI.3 and made use of the latest available calculational tools. Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190 (Reference 6). 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 (Reference 2).

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the McGuire Units 1 and 2 reactor vessels, a series of fuel-cycle-specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

$$\varphi(\mathrm{r},\theta,\mathrm{z}) = \varphi(\mathrm{r},\theta) * \frac{\varphi(\mathrm{r},\mathrm{z})}{\varphi(\mathrm{r})}$$

where $\varphi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\varphi(r,\theta)$ is the transport solution in r,θ geometry, $\varphi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\varphi(r)$ 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 McGuire Units 1 and 2.

For the McGuire Unit 1 transport calculations, the r,θ models depicted in Figures 2-1 through 2-3 were utilized since, with the exception of the neutron pads, the reactor is octant symmetric. These r,θ models

include the core, the reactor internals, the neutron pads – including explicit representations of an octant not containing surveillance capsules and octants with surveillance capsules at 17.5° and 20.0° – the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. In developing these analytical models, nominal design dimensions were employed for the various structural components. Likewise, water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. The coolant densities were treated on a fuel-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, et cetera. The geometric mesh description of the r,θ reactor models consisted of 170 radial by 98 or 101 azimuthal intervals. 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. Due to the geometry similarity of McGuire Units 1 and 2, plan views of the McGuire Unit 2 geometry model are not presented here.

The r,z model used for the McGuire Unit 1 calculations is shown in Figure 2-4 and extends 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 to approximately five feet above the active fuel. As in the case of the r,θ models, 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 r,z geometric mesh description of these reactor models consisted of 153 radial by 188 axial intervals. As in the case of the r,θ calculations, 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,z calculations was also set at a value of 0.001. Again, a section view of the McGuire Unit 2 r,z model is not presented due to the similarity of these two units.

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 be determined on a meshwise basis throughout the entire geometry.

The data utilized for the core power distributions in the plant-specific transport analyses included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle-averaged neutron flux, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations supporting this analysis were carried out using the DORT discrete ordinates code, version 3.2 (Reference 7), and the BUGLE-96 cross-section library (Reference 8). The BUGLE-96 library provides a 67-group coupled neutron-gamma ray 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 angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

In Table 2-1, locations of the lower shell to lower vessel head circumferential weld, lower shell longitudinal welds, lower shell plates, intermediate shell to lower shell circumferential weld, intermediate shell longitudinal welds, intermediate shell plates, upper shell to intermediate shell circumferential weld, upper shell longitudinal welds, upper shell plates and outlet/inlet nozzle to upper shell welds are given for McGuire Unit 1. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the mid-plane of the active fuel stack. Similarly, in Table 2-2, locations of the lower shell B to lower vessel head circumferential weld, lower shell A to lower shell B circumferential weld, intermediate shell to lower shell A circumferential weld, and upper shell to intermediate shell circumferential weld and outlet/inlet nozzle to upper shell welds are given for McGuire Unit 2. Note that the McGuire Unit 2 reactor vessel does not have any longitudinal welds.

Selected results from the neutron transport analyses are provided in Tables 2-3 through 2-6. In Tables 2-3 and 2-5, calculated fast neutron (E > 1.0 MeV) fluence for reactor vessel materials, on the pressure vessel clad/base metal interface, is provided at future projections to 34, 40, 48 and 54 EFPY for both units. Cycle-specific calculations were performed for Cycles 1 to 20, where a core thermal power of 3411 MWt was used. The projections were based on the assumption that the core power distributions and associated plant operating characteristics from the averages of Cycles 18, 19, and 20 at each respective unit were representative of future plant operation, but with an uprated core power at 3479 MWt. In Table 2-4 and Table 2-6, calculated fast neutron (E > 1.0 MeV) fluence on the pressure vessel clad/base metal interface is provided for Cycles 1 through 20 and future projections for both McGuire Units 1 and 2, at various azimuthal locations.

2.3 CALCULATIONAL UNCERTAINTIES

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

1. Comparison of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL).
2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment.
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. Comparisons of the plant-specific calculations with all available dosimetry results from the McGuire Units 1 and 2 surveillance program.

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 or geometric variables that impact power reactor calculations. The second phase of the

qualification (H. B. Robinson comparisons) addressed uncertainties in these additional areas 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 input parameters. The overall calculational uncertainty applicable to the McGuire Units 1 and 2 analyses was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with McGuire Units 1 and 2 measurements) was used solely to demonstrate the validity 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 in any way to modify the calculated surveillance capsule and pressure vessel neutron exposures.

Table 2-7 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Reference 2. 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 Appendix A of Reference 5 support these uncertainty assessments for McGuire Unit 1; and the plant-specific measurement comparisons given in Appendix C of this report support these uncertainty assessments for McGuire Unit 2 and the Capsule W from McGuire Unit 1.

Table 2-1 Pressure Vessel Material Locations for McGuire Unit 1

Material	Axial Location (cm)	Azimuthal Location (°)	r-θ Neutron Pad Configuration used in Exposure Calculations
Lower shell to lower vessel head circumferential weld	-309.396	0 to 360	15.0° neutron pad
Lower shell longitudinal weld 1	-309.396 to -38.807	60	15.0° neutron pad
Lower shell longitudinal weld 2		180	15.0° neutron pad
Lower shell longitudinal weld 3		300	15.0° neutron pad
Lower shell plate 1	-309.396 to -38.807	60 to 180	15.0° neutron pad
Lower shell plate 2		180 to 300	15.0° neutron pad
Lower shell plate 3		300 to 60	15.0° neutron pad
Intermediate shell to lower shell circumferential weld	-38.807	0 to 360	15.0° neutron pad
Intermediate shell longitudinal weld 1	-38.807 to 237.656	0	15.0° neutron pad
Intermediate shell longitudinal weld 2		120	15.0° neutron pad
Intermediate shell longitudinal weld 3		240	15.0° neutron pad
Intermediate shell plate 1	-38.807 to 237.656	0 to 120	15.0° neutron pad
Intermediate shell plate 2		120 to 240	15.0° neutron pad
Intermediate shell plate 3		240 to 360	15.0° neutron pad
Upper shell to intermediate shell circumferential weld	237.656	0 to 360	15.0° neutron pad
Upper shell longitudinal weld 1	Above 237.656	60	15.0° neutron pad
Upper shell longitudinal weld 2		180	15.0° neutron pad
Upper shell longitudinal weld 3		300	15.0° neutron pad
Upper shell plate 1	Above 237.656	60 to 180	15.0° neutron pad
Upper shell plate 2		180 to 300	15.0° neutron pad
Upper shell plate 3		300 to 60	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 1	276.153	22	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 2		158	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 3		202	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 4		338	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 1	271.073	67	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 2		113	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 3		247	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 4		293	15.0° neutron pad

Note:

(a) Lowest extent.

Table 2-2 Pressure Vessel Material Locations for McGuire Unit 2

Material	Axial Location (cm)	Azimuthal Location (°)	r-θ Neutron Pad Configuration used in Exposure Calculations
Lower shell B to lower vessel head circumferential weld	-310.597	0 to 360	15.0° neutron pad
Lower shell A to lower shell B circumferential weld	-202.397	0 to 360	15.0° neutron pad
Intermediate shell to lower shell A circumferential weld	15.603	0 to 360	15.0° neutron pad
Upper shell to intermediate shell circumferential weld	233.103	0 to 360	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 1	275.713	22	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 2		158	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 3		202	15.0° neutron pad
Outlet Nozzle to Upper Shell Weld[(a)] 4		338	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 1	264.713	67	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 2		113	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 3		247	15.0° neutron pad
Inlet Nozzle to Upper Shell Weld[(a)] 4		293	15.0° neutron pad

Note:

(a) Lowest extent.

Table 2-3 McGuire Unit 1 Calculated Neutron Fluence Projections at the Reactor Vessel Clad/Base Metal Interface at 34, 40, 48, and 54 EFPY

Reactor Vessel Material	Fluence[a] (n/cm^2, E > 1.0 MeV)			
	34 EFPY	40 EFPY	48 EFPY	54 EFPY
Outlet Nozzle to Upper Shell Welds (Lowest Extent) 1, 2, 3, and 4	3.66E+16	4.23E+16	4.99E+16	5.56E+16
Inlet Nozzle to Upper Shell Welds (Lowest Extent) 1, 2, 3 and 4	5.06E+16	5.85E+16	6.90E+16	7.69E+16
Upper Shell Plates	3.60E+17	4.16E+17	4.91E+17	5.47E+17
Intermediate Shell Plates	1.68E+19	1.95E+19	2.30E+19	2.56E+19
Lower Shell Plates	1.68E+19	1.94E+19	2.30E+19	2.57E+19
Upper Shell Longitudinal Weld 1: 1-442 A[b]	2.97E+17	3.43E+17	4.05E+17	4.51E+17
Upper Shell Longitudinal Weld 2: 1-442 B[b]	2.06E+17	2.37E+17	2.78E+17	3.08E+17
Upper Shell Longitudinal Weld 3: 1-442 C[b]	2.97E+17	3.43E+17	4.05E+17	4.51E+17
Upper Shell to Intermediate Shell Circumferential Weld	3.60E+17	4.16E+17	4.91E+17	5.47E+17
Intermediate Shell Longitudinal Weld 1: 2-442 A[b]	9.46E+18	1.09E+19	1.28E+19	1.42E+19
Intermediate Shell Longitudinal Weld 2: 2-442 B[b]	1.40E+19	1.62E+19	1.91E+19	2.13E+19
Intermediate Shell Longitudinal Weld 3: 2-442 C[b]	1.40E+19	1.62E+19	1.91E+19	2.13E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.62E+19	1.87E+19	2.21E+19	2.47E+19
Lower Shell Longitudinal Weld 1: 3-442 A[b]	1.40E+19	1.62E+19	1.91E+19	2.13E+19
Lower Shell Longitudinal Weld 2: 3-442 B[b]	9.44E+18	1.09E+19	1.28E+19	1.42E+19
Lower Shell Longitudinal Weld 3: 3-442 C[b]	1.40E+19	1.62E+19	1.91E+19	2.13E+19
Lower Shell to Lower Vessel Head Circumferential Weld	1.49E+15	1.72E+15	2.03E+15	2.27E+15

Notes:

(a) Extended beltline materials are currently interpreted to be the reactor vessel materials that will be exposed to a neutron fluence greater than or equal to 1×10^{17} n/cm^2 (E > 1.0 MeV). Only the materials that are projected to experience a fluence value of at least 1×10^{17} n/cm^2 (E > 1.0 MeV) will be included in the subsequent evaluations contained within this report.

(b) The fluence value at each individual weld location (A, B, C) was reported here for documentation purposes; however, the maximum fluence value across all three longitudinal welds in each reactor vessel shell will be used as the bounding fluence value in the subsequent calculations contained within this report.

Table 2-4 McGuire Unit 1 Calculated Neutron Fluence at the Reactor Vessel Clad/Base Metal Interface for Cycles 1 through 20 and Future Projections

Cycle ID	Cycle Time (EFPY)	Cumulative Cycle Time (EFPY)	Fluence (n/cm^2, E > 1.0 MeV)					Maximum Fluence (n/cm^2, E > 1.0 MeV)
			0°	15°	22°	30°	45°	
1	1.09	1.09	4.21E+17	6.36E+17	7.53E+17	6.30E+17	7.71E+17	7.71E+17
2	0.73	1.82	7.80E+17	1.18E+18	1.40E+18	1.15E+18	1.37E+18	1.40E+18
3	0.79	2.61	1.08E+18	1.62E+18	1.90E+18	1.56E+18	1.83E+18	1.90E+18
4	0.83	3.44	1.37E+18	2.06E+18	2.40E+18	1.96E+18	2.29E+18	2.40E+18
5	0.86	4.30	1.70E+18	2.52E+18	2.90E+18	2.36E+18	2.72E+18	2.90E+18
6	0.82	5.12	1.99E+18	2.96E+18	3.40E+18	2.76E+18	3.16E+18	3.40E+18
7	1.11	6.23	2.36E+18	3.50E+18	4.04E+18	3.29E+18	3.76E+18	4.04E+18
8	1.01	7.24	2.71E+18	4.03E+18	4.65E+18	3.77E+18	4.29E+18	4.65E+18
9	0.91	8.15	3.00E+18	4.49E+18	5.18E+18	4.21E+18	4.79E+18	5.18E+18
10	1.07	9.22	3.36E+18	4.99E+18	5.77E+18	4.71E+18	5.37E+18	5.77E+18
11	0.99	10.21	3.69E+18	5.45E+18	6.31E+18	5.15E+18	5.84E+18	6.31E+18
12	0.99	11.20	3.95E+18	5.87E+18	6.81E+18	5.57E+18	6.32E+18	6.81E+18
13	1.19	12.39	4.31E+18	6.33E+18	7.35E+18	6.02E+18	6.78E+18	7.35E+18
14	1.31	13.70	4.71E+18	6.85E+18	7.92E+18	6.48E+18	7.26E+18	7.92E+18
15	1.39	15.09	5.04E+18	7.34E+18	8.52E+18	7.04E+18	7.96E+18	8.52E+18
16	1.40	16.49	5.36E+18	7.85E+18	9.14E+18	7.58E+18	8.60E+18	9.14E+18
17	1.35	17.84	5.68E+18	8.35E+18	9.75E+18	8.12E+18	9.24E+18	9.75E+18
18	1.38	19.22	5.99E+18	8.84E+18	1.03E+19	8.62E+18	9.77E+18	1.03E+19
19	1.29	20.51	6.30E+18	9.31E+18	1.09E+19	9.08E+18	1.02E+19	1.09E+19
20	1.32	21.83	6.60E+18	9.80E+18	1.15E+19	9.55E+18	1.07E+19	1.15E+19
- - -	- - -	26.00	7.58E+18	1.13E+19	1.33E+19	1.11E+19	1.22E+19	1.33E+19
- - -	- - -	30.00	8.52E+18	1.28E+19	1.51E+19	1.25E+19	1.37E+19	1.51E+19
- - -	- - -	34.00	9.46E+18	1.43E+19	1.68E+19	1.40E+19	1.52E+19	1.68E+19
- - -	- - -	40.00	1.09E+19	1.65E+19	1.95E+19	1.62E+19	1.74E+19	1.95E+19
- - -	- - -	44.00	1.18E+19	1.80E+19	2.12E+19	1.76E+19	1.88E+19	2.12E+19
- - -	- - -	48.00	1.28E+19	1.95E+19	2.30E+19	1.91E+19	2.03E+19	2.30E+19
- - -	- - -	54.00	1.42E+19	2.17E+19	2.57E+19	2.13E+19	2.26E+19	2.57E+19
- - -	- - -	60.00	1.56E+19	2.39E+19	2.83E+19	2.35E+19	2.48E+19	2.83E+19

Table 2-5 McGuire Unit 2 Calculated Neutron Fluence Projections at the Reactor Vessel Clad/Base Metal Interface at 34, 40, 48, and 54 EFPY

Reactor Vessel Material	Fluence[(a)] (n/cm^2, E > 1.0 MeV)			
	34 EFPY	**40 EFPY**	**48 EFPY**	**54 EFPY**
Outlet Nozzle to Upper Shell Welds (Lowest Extent) 1, 2, 3, and 4	3.69E+16	4.23E+16	4.94E+16	5.48E+16
Inlet Nozzle to Upper Shell Welds (Lowest Extent) 1, 2, 3, and 4	6.67E+16	7.64E+16	8.93E+16	9.90E+16
Upper Shell Plate	4.80E+17	5.49E+17	6.42E+17	7.11E+17
Intermediate Shell Plate	1.62E+19	1.86E+19	2.17E+19	2.41E+19
Lower Shell Plate A	1.66E+19	1.91E+19	2.24E+19	2.48E+19
Lower Shell Plate B	2.26E+18	2.59E+18	3.03E+18	3.36E+18
Upper Shell to Intermediate Shell Circumferential Weld	4.80E+17	5.49E+17	6.42E+17	7.11E+17
Intermediate Shell to Lower Shell A Circumferential Weld	1.57E+19	1.80E+19	2.11E+19	2.34E+19
Lower Shell A to Lower Shell B Circumferential Weld	2.26E+18	2.59E+18	3.03E+18	3.36E+18
Lower Shell B to Lower Vessel Head Circumferential Weld	1.36E+15	1.57E+15	1.84E+15	2.04E+15

Note:

(a) Extended beltline materials are currently interpreted to be the reactor vessel materials that will be exposed to a neutron fluence greater than or equal to 1×10^{17} n/cm^2 (E > 1.0 MeV). Only the materials that are projected to experience a fluence value of at least 1×10^{17} n/cm^2 (E > 1.0 MeV) will be included in the subsequent evaluations contained within this report.

Table 2-6 McGuire Unit 2 Calculated Neutron Fluence at the Reactor Vessel Clad/Base Metal Interface for Cycles 1 through 20 and Future Projections

Cycle ID	Cycle Time (EFPY)	Cumulative Cycle Time (EFPY)	Fluence (n/cm^2, E > 1.0 MeV)					Maximum Fluence (n/cm^2, E > 1.0 MeV)
			0°	15°	22°	30°	45°	
1	1.03	1.03	4.02E+17	6.10E+17	7.25E+17	6.01E+17	7.14E+17	7.25E+17
2	0.68	1.71	7.22E+17	1.10E+18	1.30E+18	1.07E+18	1.27E+18	1.30E+18
3	0.73	2.44	9.79E+17	1.50E+18	1.78E+18	1.48E+18	1.77E+18	1.78E+18
4	0.85	3.29	1.29E+18	1.97E+18	2.32E+18	1.91E+18	2.27E+18	2.32E+18
5	0.87	4.16	1.61E+18	2.46E+18	2.87E+18	2.35E+18	2.74E+18	2.87E+18
6	0.92	5.08	1.97E+18	2.99E+18	3.47E+18	2.80E+18	3.22E+18	3.47E+18
7	0.97	6.05	2.32E+18	3.54E+18	4.09E+18	3.29E+18	3.75E+18	4.09E+18
8	1.13	7.18	2.75E+18	4.17E+18	4.81E+18	3.85E+18	4.37E+18	4.81E+18
9	1.09	8.27	3.13E+18	4.73E+18	5.46E+18	4.39E+18	4.99E+18	5.46E+18
10	1.17	9.44	3.49E+18	5.25E+18	6.08E+18	4.90E+18	5.58E+18	6.08E+18
11	1.12	10.56	3.84E+18	5.76E+18	6.68E+18	5.40E+18	6.14E+18	6.68E+18
12	1.20	11.76	4.17E+18	6.26E+18	7.28E+18	5.92E+18	6.72E+18	7.28E+18
13	1.31	13.07	4.49E+18	6.76E+18	7.87E+18	6.43E+18	7.34E+18	7.87E+18
14	1.35	14.42	4.83E+18	7.28E+18	8.46E+18	6.92E+18	7.92E+18	8.46E+18
15	1.42	15.84	5.23E+18	7.85E+18	9.11E+18	7.45E+18	8.56E+18	9.11E+18
16	1.40	17.24	5.58E+18	8.37E+18	9.74E+18	8.01E+18	9.25E+18	9.74E+18
17	1.40	18.64	5.94E+18	8.88E+18	1.03E+19	8.52E+18	9.80E+18	1.03E+19
18	1.29	19.93	6.24E+18	9.33E+18	1.09E+19	8.97E+18	1.03E+19	1.09E+19
19	1.37	21.30	6.55E+18	9.84E+18	1.15E+19	9.46E+18	1.08E+19	1.15E+19
20	1.36	22.66	6.87E+18	1.03E+19	1.19E+19	9.88E+18	1.14E+19	1.19E+19
- - -	- - -	26.00	7.67E+18	1.14E+19	1.33E+19	1.10E+19	1.27E+19	1.33E+19
- - -	- - -	30.00	8.61E+18	1.28E+19	1.50E+19	1.24E+19	1.43E+19	1.50E+19
- - -	- - -	34.00	9.57E+18	1.43E+19	1.66E+19	1.38E+19	1.59E+19	1.66E+19
- - -	- - -	40.00	1.10E+19	1.64E+19	1.91E+19	1.59E+19	1.83E+19	1.91E+19
- - -	- - -	44.00	1.19E+19	1.78E+19	2.07E+19	1.73E+19	1.99E+19	2.07E+19
- - -	- - -	48.00	1.29E+19	1.92E+19	2.24E+19	1.87E+19	2.15E+19	2.24E+19
- - -	- - -	54.00	1.43E+19	2.13E+19	2.48E+19	2.08E+19	2.39E+19	2.48E+19
- - -	- - -	60.00	1.57E+19	2.34E+19	2.73E+19	2.29E+19	2.63E+19	2.73E+19

Table 2-7 Calculational Uncertainties

Description	Uncertainty	
	Capsule	Vessel IR
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%

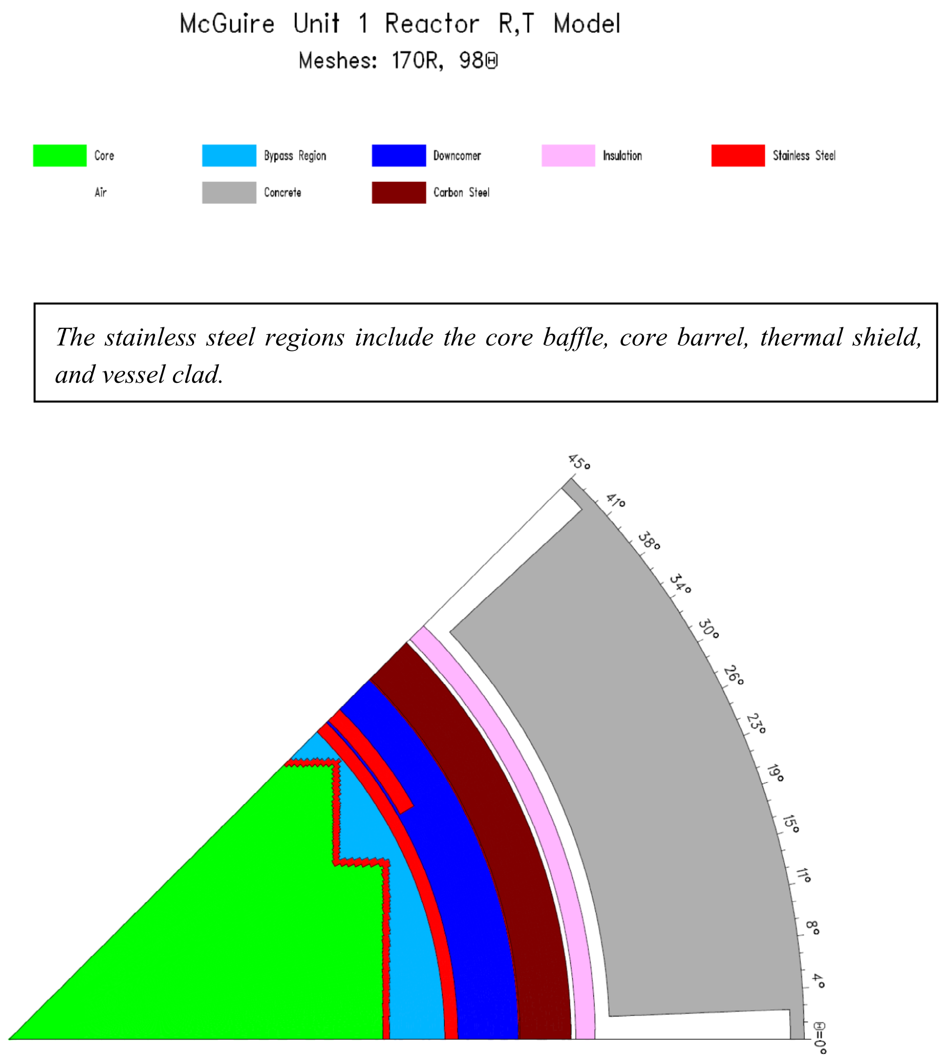

Figure 2-1 McGuire Unit 1 r-θ Reactor Geometry at the Core Mid-plane – 15° Neutron Pad Configuration

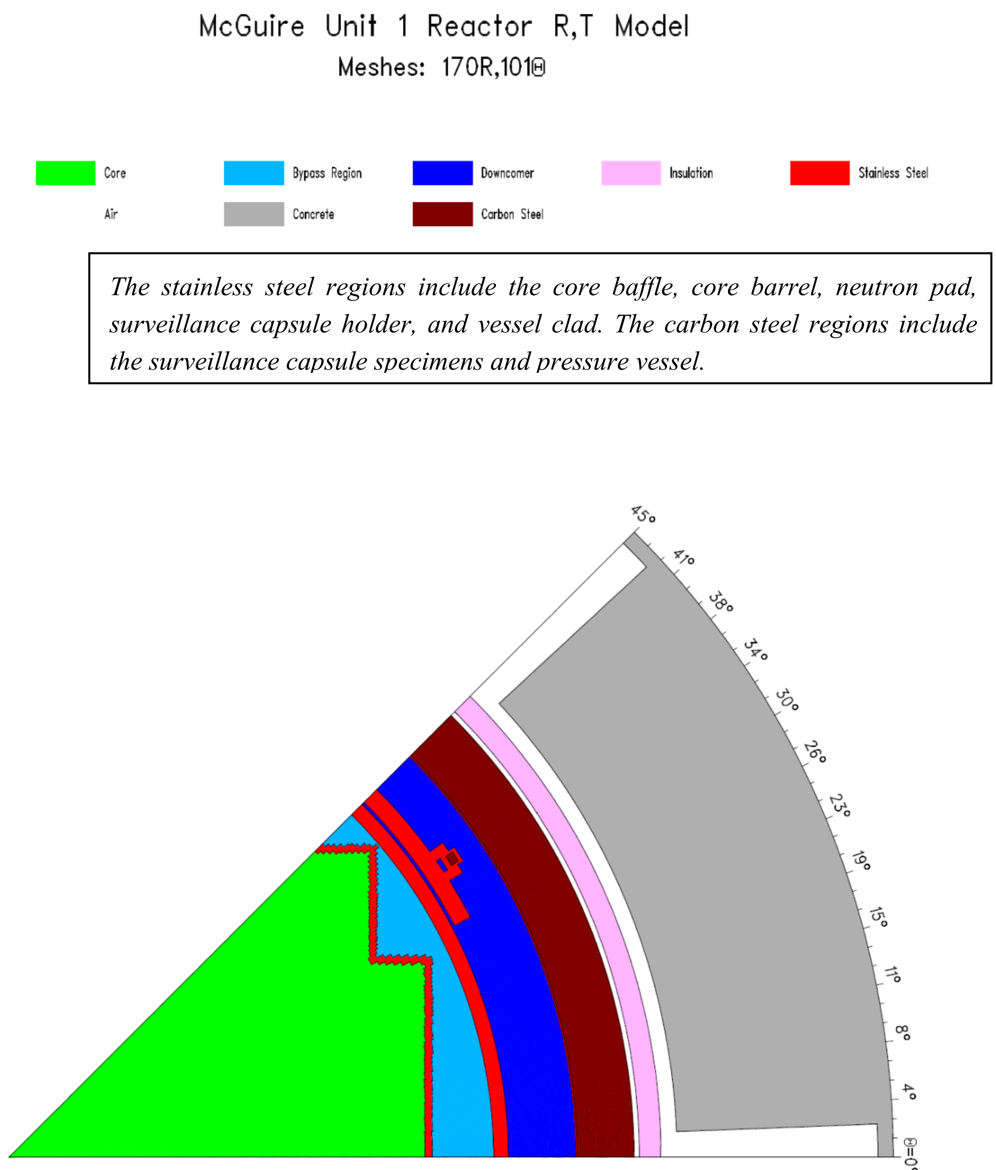

Figure 2-2 McGuire Unit 1 r-θ Reactor Geometry at the Core Mid-plane – 17.5° Neutron Pad Configuration

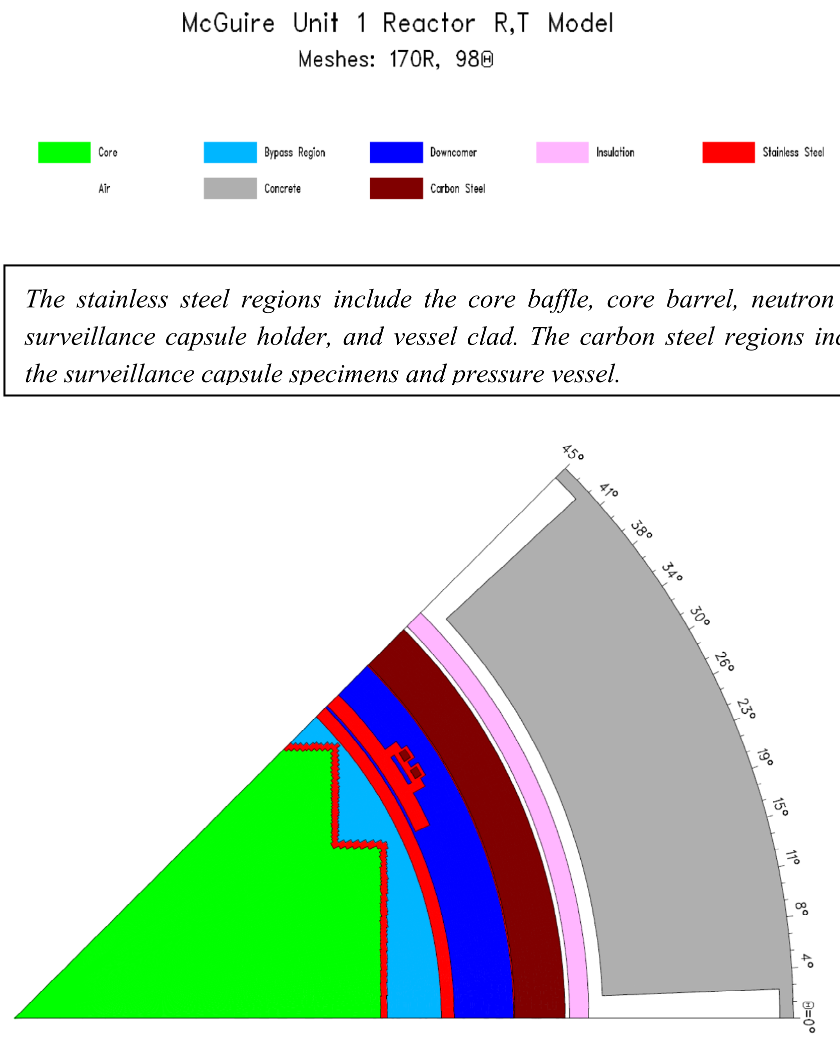

Figure 2-3 McGuire Unit 1 r-θ Reactor Geometry at the Core Mid-plane – 20.0° Neutron Pad Configuration

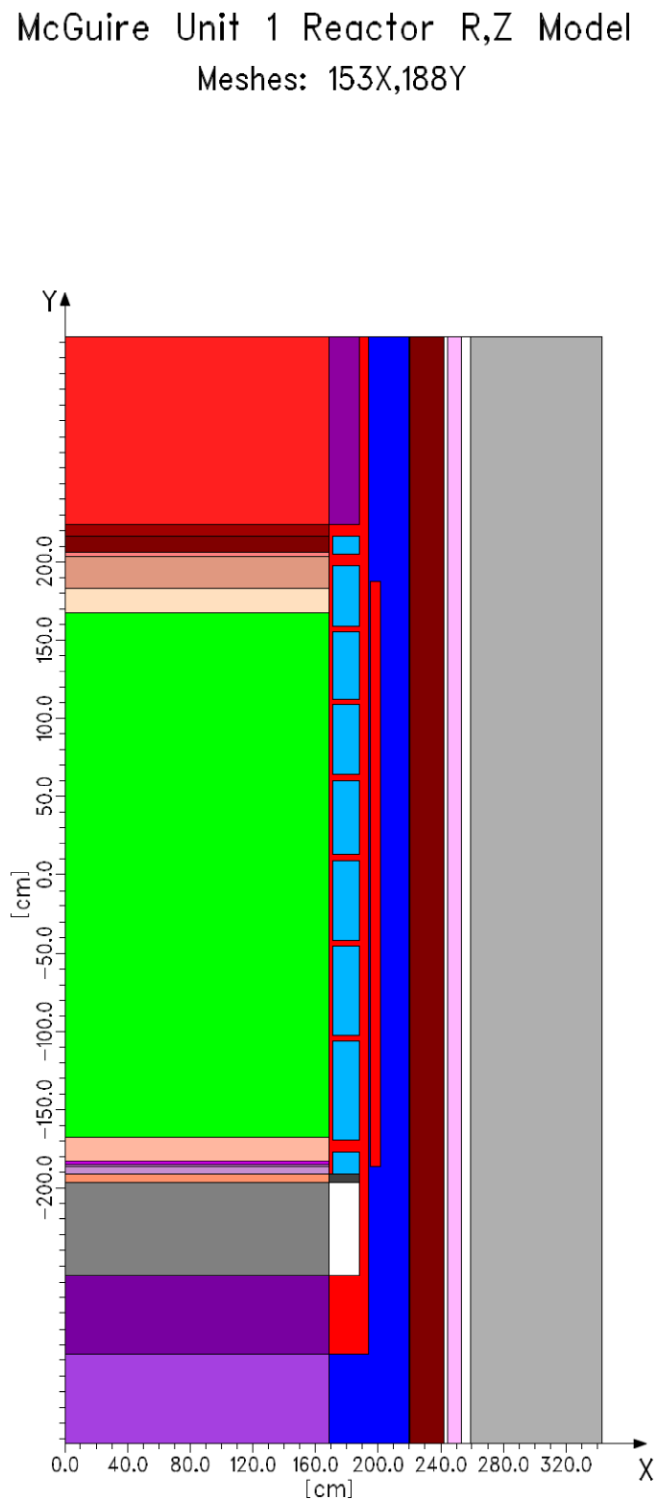

Figure 2-4 McGuire Unit 1 r-z Reactor Geometry

3 MATERIAL PROPERTY INPUT

The fracture toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan (Reference 9). The beltline region of a reactor vessel, per 10 CFR 50.61 (Reference 3), is defined as:

> *"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 beltline materials, as described in the paragraph above, must be considered in the RV integrity evaluations. Additionally, as described in Item IV.A2.R-84 of NUREG-1801, Revision 2 (Reference 10), any materials with an EOLE fluence value exceeding 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) must be considered in the RV integrity evaluations. The materials that exceed this threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met through EOLE.

Summaries of the best-estimate copper and nickel contents, $RT_{NDT(U)}$ values, and initial USE values for the reactor vessel materials are provided in Tables 3.1-1 and 3.2-1 for McGuire Units 1 and 2, respectively

3.1 MCGUIRE UNIT 1

Table 3.1-1 Material Properties for the McGuire Unit 1 Reactor Vessel[a]

Material Description	Chemical Composition		Fracture Toughness Properties	
Reactor Vessel Material and Identification Number	**Cu Wt. %**	**Ni Wt. %**	**Initial RT_{NDT}[b] (°F)**	**Initial USE (ft-lb)**
Upper Shell (US) Plate B5453-2	0.14	0.58	15	72.4[c]
US Plate B5011-2	0.10	0.54	27	68.3[c]
US Plate B5011-3	0.13	0.56	0	94.7[c]
Intermediate Shell (IS) Plate B5012-1	0.11	0.61	34	101
IS Plate B5012-2	0.14	0.61	0	105
IS Plate B5012-3	0.11	0.66	-13	112[c]
Lower Shell (LS) Plate B5013-1	0.14	0.58	0	95[c]
LS Plate B5013-2	0.10	0.51	30	115
LS Plate B5013-3	0.10	0.55	15	103[c]
US Longitudinal (Long.) Welds 1-442A,B,C (Heat # 20291/12008)[g]	0.199	0.846	-50	112[c]
US to IS Circumferential (Circ.) Weld 8-442 (Heat # 21935)	0.183	0.704	-56[d]	109[c]
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	0.199[e]	0.846[e]	-50	112[c]
IS to LS Circ. Weld 9-442 (Heat # 83640)	0.051[e]	0.096[e]	-70	143[c]
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	0.213[e]	0.867[e]	-50	124
McGuire Unit 1 Surveillance Weld (Heat # 20291/12008)	0.198[e]	0.874[e]	- - -	- - -
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	0.22[f]	0.87[f]	- - -	- - -

Notes:

(a) Values obtained from Tables 3.1.2-1 and 3.1.2-2 of WCAP-17174-P (Reference 11), unless otherwise noted.
(b) Per WCAP-16945-NP (Reference 12), the initial RT_{NDT} values are based on measured data, unless otherwise noted.
(c) Values obtained from Table 4.2-1 of the McGuire and Catawba License Renewal Application (LRA) (Reference 13).
(d) Initial RT_{NDT} is the generic value from 10 CFR 50.61 (Reference 3) for a weld made with Linde 1092 flux.
(e) Values obtained from Table 1 of WCAP-15192, Revision 2 (Reference 14).
(f) Values obtained from Page 4-6 of WCAP-15423 (Reference 15).
(g) The weld identification number and heat number for the US Long. Welds were previously reported incorrectly as 1-422 and 21935/12008, respectively. These are corrected herein to be 1-442 and 20291/12008. The information reported herein should be used in any future McGuire Unit 1 evaluations.

3.2 MCGUIRE UNIT 2

Table 3.2-1 Material Properties for the McGuire Unit 2 Reactor Vessel[a]

Material Description	Chemical Composition		Fracture Toughness Properties	
Reactor Vessel Material and Identification Number	**Cu Wt. %**	**Ni Wt. %**	**Initial RT_{NDT}[b] (°F)**	**Initial USE (ft-lb)**
US Forging 06	0.16[c]	0.89	25	98[e]
IS Forging 05	0.153	0.793	-4	94
LS Forging 04	0.15	0.88	-30	141
Bottom Head Ring 03	0.06	0.77	15	>71[d]
US to IS Circ. Weld W06 (Heat # 1725)	0.11[e]	0.29[f]	10[g]	>71[e]
IS to LS Circ. Weld W05 (Heat # 895075)	0.039	0.724	-68	132
LS to Bottom Head Ring Weld W04 (Heat # 899680)	0.03[i]	0.75[i]	10[i]	99[i]
McGuire Unit 2 Surveillance Weld (Heat # 895075)	0.04[h]	0.74[h]	- - -	- - -
Catawba Unit 1 Surveillance Weld (Heat # 895075)	0.05[h]	0.73[h]	- - -	- - -
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	0.03[h]	0.75[h]	- - -	- - -

Notes:

(a) Values obtained from Tables 3.2.2-1 and 3.2.2-2 of WCAP-17174-P (Reference 11), unless otherwise noted.

(b) All initial RT_{NDT} values are based on measured data.

(c) No weight percent copper value was reported in the Certified Material Test Report (CMTR). Therefore, the maximum copper weight percent value for A508 Class 2 forging materials is conservatively applied based on the generic data provided in Appendix G of Oak Ridge National Laboratory document ORNL/TM-2006/530 (Reference 16).

(d) According to WCAP-17174-P (Reference 11), the value is 109 ft-lb. However, according to the CMTR, the test direction was tangential (strong). Therefore, per Section B.1.2 of NUREG-0800 Branch Technical Position 5-3 (Reference 9), the value should be reduced to 65% of the strong direction value to approximate the weak direction.

(e) Values obtained from Table 4.2-2 of the McGuire and Catawba LRA (Reference 13).

(f) This nickel weight percent is the maximum value from the Rotterdam weld certification records and is a conservative value for weld Heat # 1725, Flux 89, Lot # 2275.

(g) Value is based on measured data from the Rotterdam weld certification records for weld Heat # 1725. The initial RT_{NDT} was determined using the measured data and the method described in Section B.1.1(4) of NUREG-0800 Branch Technical Position 5-3 (Reference 9).

(h) Information for the surveillance welds is taken from Table 4 of WCAP-15203, Revision 1 (Reference 17).

(i) The weld certification records for weld Heat # 899680 reports only three impact energy values at a single test temperature (-12°C or 10.4°F) that did not reach greater than 55% shear. No other information is available for this weld heat. The initial RT_{NDT} was determined using the measured data and the method described in Section B.1.1(4) of NUREG-0800 Branch Technical Position 5-3 (Reference 9). Furthermore, weld Heat # 895075 does have USE data and is a Rotterdam weld of the same type (Grau L.O., LW 320). Therefore, the weld Heat # 895075 test results from the first surveillance capsule were used in accordance with Section B.1.2 of NUREG-0800 Branch Technical Position 5-3 (Reference 9) to conservatively estimate the initial USE value for weld Heat # 899680.

4 SURVEILLANCE DATA

Per 10 CFR 50.61, calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. Furthermore, Regulatory Guide 1.99, Revision 2 allows the use of data from the plant-specific surveillance program in determining the decrease in USE.

In addition to the plant-specific surveillance data, 10 CFR 50.61 also requires the use of data from surveillance programs at other plants which include the same limiting beltline material when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister plant' data.

Section 4.1 summarizes the McGuire Unit 1 surveillance data as well as surveillance data from Diablo Canyon Unit 2. The Diablo Canyon Unit 2 surveillance program includes weld Heat # 21935/12008, which is the same weld heat as the McGuire Unit 1 Lower Shell longitudinal welds. Thus, the Diablo Canyon Unit 2 data will be used in calculation of the Position 2.1 chemistry factor value for McGuire Unit 1 weld Heat # 21935/12008.

Section 4.2 summarizes the McGuire Unit 2 surveillance data as well as surveillance data from Catawba Unit 1 and Watts Bar Unit 1. The Catawba Unit 1 and Watts Bar Unit 1 surveillance programs include weld Heat # 895075, which is the same weld heat as the McGuire Unit 2 Intermediate Shell to Lower Shell circumferential weld. Thus, the Catawba Unit 1 and Watts Bar Unit 1 data will be used in calculation of the Position 2.1 chemistry factor value for McGuire Unit 2 weld Heat # 895075.

4.1 MCGUIRE UNIT 1 AND DIABLO CANYON UNIT 2 SURVEILLANCE DATA

Table 4.1-1 McGuire Unit 1 Surveillance Capsule Data

Material	Capsule	Withdrawal EFPY	Lead Factor[a]	Capsule Fluence[a] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)	Measured USE Decrease[b] (%)
IS Plate B5012-1 (Longitudinal)	U	1.09	4.96	0.382	30.95	5.0
	X	4.30	4.83	1.40	33.51	5.0
	V	7.24	4.15	1.93	81.01	11.0
	Z	7.24	4.75	2.21	- - -	- - -
	Y	10.21	4.20	2.65	93.10	9.0
	W	19.22	4.92	5.08	- - -	- - -
IS Plate B5012-1 (Transverse)	U	1.09	4.96	0.382	48.44	1.0
	X	4.30	4.83	1.40	60.69	0.0
	V	7.24	4.15	1.93	74.60	7.0
	Z	7.24	4.75	2.21	- - -	- - -
	Y	10.21	4.20	2.65	108.58	0.0
	W	19.22	4.92	5.08	- - -	- - -
Surveillance Weld Material (Heat # 20291/12008)	U	1.09	4.96	0.382	161.1[c]	34[c]
	X	4.30	4.83	1.40	170.6[c]	26[c]
	V	7.24	4.15	1.93	179.8[c]	38[c]
	Z	7.24	4.75	2.21	- - -	- - -
	Y	10.21	4.20	2.65	190.2[c]	41[c]
	W	19.22	4.92	5.08	208.0[c]	36[c]

Notes:

(a) The calculated fluence values and lead factors were updated as part of the MUR power uprate analysis.
(b) Values obtained from Table 5-10 of WCAP-14993 (Reference 18), unless otherwise noted.
(c) Values obtained from Table 5-4 of WCAP-17014-NP (Reference 5).

Table 4.1-2 Diablo Canyon Unit 2 Surveillance Capsule Data for Weld Heat # 21935/12008[a]

Material	Capsule	Capsule Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift (°F)	Inlet Temperature (°F)	Temperature Adjustment (°F)
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	U	0.330	173.0	545	-10[b]
	X	0.906	203.2		
	Y	1.53	211.4		
	V	2.38	224.5		

Notes:

(a) Data pertaining to the Diablo Canyon Unit 2 surveillance weld was taken from Table 4.2-1 of WCAP-17315-NP (Reference 19), unless otherwise noted.

(b) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 555°F for McGuire Unit 1 (applied to the weld ΔRT_{NDT} data for each of the Diablo Canyon Unit 2 capsules in the Position 2.1 Chemistry Factor calculation).

4.2 MCGUIRE UNIT 2, CATAWBA UNIT 1, AND WATTS BAR UNIT 1 SURVEILLANCE DATA

Table 4.2-1 McGuire Unit 2 Surveillance Capsule Data

Material	Capsule	Withdrawal EFPY	Lead Factor[a]	Capsule Fluence[a] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)	Measured USE Decrease[b] (%)
IS Forging 05 (Axial)	V	1.03	4.16	0.302	58.64	10
	X	4.16	4.81	1.38	91.12	19
	U	6.05	4.66	1.90	84.14	11
	Y	7.18	4.03	1.94	- - -	- - -
	Z	7.18	4.60	2.21	- - -	- - -
	W	9.44	4.64	2.82	130.33	21
IS Forging 05 (Tangential)	V	1.03	4.16	0.302	68.97	13
	X	4.16	4.81	1.38	98.28	14
	U	6.05	4.66	1.90	91.18	21
	Y	7.18	4.03	1.94	- - -	- - -
	Z	7.18	4.60	2.21	- - -	- - -
	W	9.44	4.64	2.82	102.03	27
Surveillance Weld Material (Heat # 895075)	V	1.03	4.16	0.302	38.51	0
	X	4.16	4.81	1.38	35.93	0
	U	6.05	4.66	1.90	23.81	3
	Y	7.18	4.03	1.94	- - -	- - -
	Z	7.18	4.60	2.21	- - -	- - -
	W	9.44	4.64	2.82	43.76	4

Notes:

(a) The calculated fluence values and lead factors were updated as part of the MUR power uprate analysis.
(b) Values obtained from Table 5-10 of WCAP-14799 (Reference 20).

Table 4.2-2 Catawba Unit 1 and Watts Bar Unit 1 Surveillance Capsule Data for Weld Heat # 895075

Material	Capsule	Capsule Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift (°F)	Inlet Temperature (°F)	Temperature Adjustment (°F)
Catawba Unit 1 Surveillance Weld[a] (Heat # 895075)	Z	0.284[b]	1.91	562[c]	+7[f]
	Y	1.27[b]	17.79		
	V	2.23[b]	26.5		
Watts Bar Unit 1 Surveillance Weld[a] (Heat # 895075)	U	0.447	0.0[e]	560[d]	+5[f]
	W	1.08	30.5		
	X	1.71	25.8		
	Z	2.40	13.9		

Notes:

(a) Data pertaining to the Catawba Unit 1 and Watts Bar Unit 1 surveillance welds was taken from Table 2-3 of WCAP-16761-NP (Reference 21), unless otherwise noted.
(b) Capsule fluence values for Catawba Unit 1 were taken from Table 2.3-2 of WCAP-16869-NP, Revision 1 (Reference 22).
(c) Catawba Unit 1 inlet temperature was updated as part of the latest ex-vessel neutron dosimetry evaluation performed in WCAP-16869-NP, Revision 1 (Reference 22).
(d) Watts Bar Unit 1 inlet temperature was taken from footnote (4) of Table 4-8 of WCAP-15118 (Reference 23).
(e) Original value was -6.4°F, but physically a reduction should not occur; therefore, a value of zero will be used.
(f) Temperature adjustment = $1.0*(T_{capsule} - T_{plant})$, where T_{plant} = 555°F for McGuire Unit 2 (applied to the weld ΔRT_{NDT} data for each of the Catawba Unit 1 and Watts Bar Unit 1 capsules in the Position 2.1 Chemistry Factor calculation).

5 CHEMISTRY FACTORS

As described in Section 1 of this report, Position 1.1 chemistry factors 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 10 CFR 50.61. The best-estimate copper and nickel weight percents for the McGuire Units 1 and 2 reactor vessel materials were provided in Tables 3.1-1 and 3.2-1 of this report, respectively.

The Position 2.1 chemistry factors are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in 10 CFR 50.61, which is also summarized in Section 1 of this report. The McGuire Units 1 and 2 surveillance data as well as any applicable sister plant data was summarized in Section 4 of this report, and will be utilized in the Position 2.1 chemistry factor calculations in this Section.

The Position 2.1 chemistry factor calculations are presented in Tables 5.1-1 and 5.2-1 for the McGuire Units 1 and 2 reactor vessel materials contained in their respective radiation surveillance programs. These values were calculated using the surveillance data summarized in Section 4 of this report. Additionally, surveillance data from Catawba Unit 1 and Watts Bar Unit 1 is utilized in Table 5.2-1 as it is applicable to the McGuire Unit 2 Intermediate Shell to Lower Shell circumferential weld (Heat # 895075).

The Position 2.1 chemistry factor calculations for weld Heat # 21935/12008 are presented in Table 5.1-2. This weld heat is not contained in the McGuire Unit 1 surveillance program, but is contained in the Diablo Canyon Unit 2 surveillance program. Therefore, the sister plant data as presented in Section 4 of this report is used to calculate the Position 2.1 chemistry factor for weld Heat # 21935/12008.

All of the surveillance data is adjusted for irradiation temperature and chemical composition differences in accordance with the guidance presented at an industry meeting held by the NRC on February 12 and 13, 1998 (Reference 24).

The Position 1.1 chemistry factors are summarized along with the Position 2.1 chemistry factors in Tables 5.1-3 and 5.2-2 for McGuire Units 1 and 2, respectively.

Appendix A contains the credibility evaluation for each of the surveillance materials for which a chemistry factor is calculated in this Section, with exception to the Diablo Canyon Unit 2 surveillance weld. The credibility evaluation for Diablo Canyon Unit 2 surveillance weld Heat # 21935/12008 is contained in Appendix A.2 of WCAP-17315-NP (Reference 19) and concludes that the data is credible. Margin will be applied to the calculations of ART and RT_{PTS} according to the conclusions of the credibility evaluation for each of the surveillance materials.

5.1 MCGUIRE UNIT 1

Table 5.1-1 Calculation of McGuire Unit 1 Chemistry Factors Using McGuire Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
IS Plate B5012-1 (Longitudinal)	U	0.382	0.734	30.95	22.71	0.538
	X	1.40	1.093	33.51	36.64	1.196
	V	1.93	1.180	81.01	95.57	1.392
	Y	2.65	1.261	93.10	117.37	1.589
IS Plate B5012-1 (Transverse)	U	0.382	0.734	48.44	35.54	0.538
	X	1.40	1.093	60.69	66.36	1.196
	V	1.93	1.180	74.60	88.01	1.392
	Y	2.65	1.261	108.58	136.88	1.589
				SUM:	599.08	9.430
	$CF_{IS\ Plate} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (599.08) \div (9.430) = \mathbf{63.5°F}$					
Surveillance Weld (Heat # 20291/12008)	U	0.382	0.734	159.49 (161.1)	117.01	0.538
	X	1.40	1.093	168.89 (170.6)	184.67	1.196
	V	1.93	1.180	178.00 (179.8)	210.00	1.392
	Y	2.65	1.261	188.30 (190.2)	237.38	1.589
	W	5.08	1.405	205.92 (208.0)	289.41	1.975
				SUM:	1038.48	6.690
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1038.48) \div (6.690) = \mathbf{155.2°F}$					

Notes:

(a) f = fluence.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Table 4.1-1 of this report. The ΔRT_{NDT} values for the surveillance weld data have been adjusted by a ratio of 0.99 (pre-adjusted values are listed in parentheses).
Ratio = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 201.3°F / 204.2°F = 0.99.

Table 5.1-2 Calculation of McGuire Unit 1 Weld Heat # 21935/12008 Chemistry Factor Using Surveillance Capsule Data from Diablo Canyon Unit 2

Material	Capsule	Capsule f(a) ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	U	0.330	0.695	161.37 (173.0)	112.16	0.483
	X	0.906	0.972	191.27 (203.2)	185.97	0.945
	Y	1.53	1.118	199.39 (211.4)	222.84	1.249
	V	2.38	1.234	212.36 (224.5)	262.01	1.522
				SUM:	782.99	4.200
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (782.99) \div (4.200) =$ **186.4°F**					

Notes:

(a) f = fluence.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Table 4.1-2 of this report. The Diablo Canyon Unit 2 ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the beltline weld chemistry (pre-adjusted values are listed in parentheses). The temperature adjustment is -10°F per Table 4.1-2 of this report. Ratio = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 208.2°F / 211.2°F = 0.99.

Table 5.1-3 Summary of McGuire Unit 1 Positions 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material and Identification Number	Chemistry Factor (˚F)	
	Position 1.1[a]	Position 2.1
US Plate B5453-2	99.1	- - -
US Plate B5011-2	65	- - -
US Plate B5011-3	89.8	- - -
IS Plate B5012-1	74.2	63.5[b]
IS Plate B5012-2	100.3	- - -
IS Plate B5012-3	74.9	- - -
LS Plate B5013-1	99.1	- - -
LS Plate B5013-2	65	- - -
LS Plate B5013-3	65	- - -
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	201.3	155.2[b]
US to IS Circ. Weld 8-442 (Heat # 21935)	170.5	- - -
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	201.3	155.2[b]
IS to LS Circ. Weld 9-442 (Heat # 83640)	37.5	- - -
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	208.2	186.4[c]
McGuire Unit 1 Surveillance Weld (Heat # 20291/12008)	204.2	- - -
Diablo Canyon Unit 2 Surveillance Weld (Heat # 21935/12008)	211.2	- - -

Notes:

(a) Position 1.1 Chemistry Factors were calculated using the copper and nickel weight percents presented in Table 3.1-1 of this report and Tables 1 and 2 of 10 CFR 50.61.

(b) Position 2.1 Chemistry Factors taken from Table 5.1-1 of this report. Per Appendix A.1, the McGuire Unit 1 surveillance plate and weld data is credible.

(c) Position 2.1 Chemistry Factor taken from Table 5.1-2 of this report. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld data is credible.

5.2 MCGUIRE UNIT 2

Table 5.2-1 Calculation of McGuire Unit 2 Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
IS Forging 05 (Axial)	V	0.302	0.672	58.64	39.41	0.452
	X	1.38	1.089	91.12	99.27	1.187
	U	1.90	1.176	84.14	98.92	1.382
	W	2.82	1.276	130.33	166.28	1.628
IS Forging 05 (Tangential)	V	0.302	0.672	68.97	46.35	0.452
	X	1.38	1.089	98.28	107.07	1.187
	U	1.90	1.176	91.18	107.20	1.382
	W	2.82	1.276	102.03	130.17	1.628
				SUM:	794.67	9.297
	$CF_{IS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (794.67) \div (9.297) =$ **85.5˚F**					
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	0.302	0.672	37.74 (38.51)	25.36	0.452
	X	1.38	1.089	35.21 (35.93)	38.36	1.187
	U	1.90	1.176	23.33 (23.81)	27.43	1.382
	W	2.82	1.276	42.88 (43.76)	54.71	1.628
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	0.284	0.656	6.95 (1.91)	4.56	0.431
	Y	1.27	1.067	19.34 (17.79)	20.62	1.138
	V	2.23	1.217	26.13 (26.5)	31.81	1.482
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	0.447	0.776	6.45 (0.0[(d)])	5.01	0.602
	W	1.08	1.022	45.80 (30.5)	46.78	1.044
	X	1.71	1.148	39.73 (25.8)	45.60	1.317
	Z	2.40	1.236	24.38 (13.9)	30.13	1.528
				SUM:	330.38	12.189
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (330.38) \div (12.189) =$ **27.1˚F**					

Notes:

(a) f = fluence.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Tables 4.2-1 and 4.2-2 of this report. The Catawba Unit 1 and Watts Bar Unit 1 surveillance weld ΔRT_{NDT} values have been adjusted according to the temperature adjustments summarized in Table 4.2-2 of this report then by using the ratio procedure to account for differences in the surveillance weld chemistry and the beltline weld chemistry (pre-adjusted values are listed in parentheses). For Catawba Unit 1, Ratio = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 52.7°F / 68°F = 0.78. For Watts Bar Unit 1, Ratio = 52.7°F / 41°F = 1.29. The McGuire Unit 2 surveillance weld ΔRT_{NDT} values are not adjusted for temperature differences, but are adjusted by a ratio for chemistry differences (pre-adjusted values are listed in parentheses). For McGuire Unit 2, Ratio = 52.7°F / 54°F = 0.98.

(d) This ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur; therefore, a value of zero is used.

Table 5.2-2 Summary of McGuire Unit 2 Positions 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material and Identification Number	Chemistry Factor (˚F)	
	Position 1.1[a]	Position 2.1[b]
US Forging 06	123.9	- - -
IS Forging 05	117.2	85.5
LS Forging 04	115.8	- - -
Bottom Head Ring 03	37	- - -
US to IS Circ. Weld W06 (Heat # 1725)	82.9	- - -
IS to LS Circ. Weld W05 (Heat # 895075)	52.7	27.1
LS to Bottom Head Ring Weld W04 (Heat # 899680)	41	- - -
McGuire Unit 2 Surveillance Weld (Heat # 895075)	54	- - -
Catawba Unit 1 Surveillance Weld (Heat # 895075)	68	- - -
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	41	- - -

Notes:

(a) Position 1.1 Chemistry Factors were calculated using the copper and nickel weight percents presented in Table 3.2-1 of this report and Tables 1 and 2 of 10 CFR 50.61.

(b) Position 2.1 Chemistry Factors taken from Table 5.2-1 of this report. Per Appendix A.2, the McGuire Unit 2 surveillance forging and weld data is credible.

6 PRESSURIZED THERMAL SHOCK CALCULATIONS

A limiting condition on reactor vessel integrity known as 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 vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization,
- 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 (10 CFR 50.61) on PTS (Reference 3) that established screening criteria on reactor vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end-of-license, termed reference temperature for pressurized thermal shock (RT_{PTS}). 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 RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2 (Reference 1).

These accepted methods were used with the surface fluence of Section 2 to calculate the following RT_{PTS} values for the McGuire Units 1 and 2 reactor vessel materials at 54 EFPY (EOLE). The EOLE RT_{PTS} calculations are summarized below in Table 6.1-1 for McGuire Unit 1 and in Table 6.2-1 for McGuire Unit 2.

PTS Conclusion

For McGuire Unit 1, the limiting RT_{PTS} value at 54 EFPY is 203°F (see Table 6.1-1); this value corresponds to LS Longitudinal Welds 3-442A,B,C (Position 2.1). For McGuire Unit 2, the limiting RT_{PTS} value at 54 EFPY is 148°F (see Table 6.2-1); this value corresponds to LS Forging 04. Therefore, all of the beltline and extended beltline materials in the McGuire Units 1 and 2 reactor vessels are below the RT_{PTS} screening criteria values of 270°F for axially oriented welds and plates / forgings, and 300°F for circumferentially oriented welds through EOLE (54 EFPY).

The Alternate PTS Rule (10 CFR 50.61a (Reference 25)) was published in the *Federal Register* by the NRC in 2010. This alternate rule is less restrictive than the Mandatory PTS Rule (10 CFR 50.61) and is intended to be used for situations where the 10 CFR 50.61 criteria cannot be met. McGuire Units 1 and 2 currently meet the criteria for the Mandatory PTS Rule through EOLE and therefore do not need to utilize the Alternate PTS Rule at this time.

6.1 MCGUIRE UNIT 1

Table 6.1-1 RT_{PTS} Calculations for the McGuire Unit 1 Reactor Vessel Materials at 54 EFPY[a]

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[b] (°F)	Fluence[c] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT} [d] (°F)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[f] (°F)	Margin (°F)	RT_{PTS} (°F)
US Plate B5453-2	1.1	99.1	0.0547	0.3072	15	30.4	0	15.2	30.4	76
US Plate B5011-2	1.1	65	0.0547	0.3072	27	20.0	0	10.0	20.0	67
US Plate B5011-3	1.1	89.8	0.0547	0.3072	0	27.6	0	13.8	27.6	55
IS Plate B5012-1	1.1	74.2	2.56	1.2521	34	92.9	0	17	34	161
Using credible surveillance data	2.1	63.5	2.56	1.2521	34	79.5	0	8.5	17	131
IS Plate B5012-2	1.1	100.3	2.56	1.2521	0	125.6	0	17	34	160
IS Plate B5012-3	1.1	74.9	2.56	1.2521	-13	93.8	0	17	34	115
LS Plate B5013-1	1.1	99.1	2.57	1.2531	0	124.2	0	17	34	158
LS Plate B5013-2	1.1	65	2.57	1.2531	30	81.4	0	17	34	145
LS Plate B5013-3	1.1	65	2.57	1.2531	15	81.4	0	17	34	130
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.0451	0.2767	-50	55.7	0	27.9	55.7	61
Using credible surveillance data	2.1	155.2	0.0451	0.2767	-50	43.0	0	14	28	21
US to IS Circ. Weld 8-442 (Heat # 21935)	1.1	170.5	0.0547	0.3072	-56[e]	52.4	17[e]	26.2	62.4	59

Table 6.1-1 RT_{PTS} Calculations for the McGuire Unit 1 Reactor Vessel Materials at 54 EFPY[(a)]

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(b)] (°F)	Fluence[(c)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT} [(d)] (°F)	ΔRT_{NDT} (°F)	σ_U [(d)] (°F)	σ_Δ [(f)] (°F)	Margin (°F)	RT_{PTS} (°F)
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	1.1	201.3	2.13	1.2055	-50	242.7	0	28	56	249
Using credible surveillance data	2.1	155.2	2.13	1.2055	-50	187.1	0	14	28	165
IS to LS Circ. Weld 9-442 (Heat # 83640)	1.1	37.5	2.47	1.2432	-70	46.6	0	23.3	46.6	23
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	2.13	1.2055	-50	251.0	0	28	56	257
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	2.13	1.2055	-50	224.7	0	14	28	203

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values. See Section 1 of this report for details.

(b) Taken from Table 5.1-3 of this report.

(c) Taken from Table 2-3 of this report.

(d) Initial RT_{NDT} values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.

(e) Initial RT_{NDT} value is generic; therefore, $\sigma_U = 17°F$.

(f) Per Appendix A.1 of this report, the McGuire Unit 1 plate and weld surveillance data were deemed credible. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld surveillance data was also deemed credible. Per the guidance of 10 CFR 50.61, the base metal $\sigma_\Delta = 17°F$ for Position 1.1 and, with credible surveillance data, $\sigma_\Delta = 8.5°F$ for Position 2.1; the weld metal $\sigma_\Delta = 28°F$ for Position 1.1 and, with credible surveillance data, $\sigma_\Delta = 14°F$ for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

6.2 MCGUIRE UNIT 2

Table 6.2-1 RT_{PTS} Calculations for the McGuire Unit 2 Reactor Vessel Materials at 54 EFPY[a]

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[b] (°F)	Fluence[c] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	IRT_{NDT}[d] (°F)	ΔRT_{NDT} (°F)	σ_U[d] (°F)	σ_Δ[e] (°F)	Margin (°F)	RT_{PTS} (°F)
US Forging 06	1.1	123.9	0.0711	0.3521	25	43.6	0	17	34	103
IS Forging 05	1.1	117.2	2.41	1.2370	-4	145.0	0	17	34	175
Using <u>credible</u> surveillance data	2.1	85.5	2.41	1.2370	-4	105.7	0	8.5	17	119
LS Forging 04	1.1	115.8	2.48	1.2442	-30	144.1	0	17	34	148
Bottom Head Ring 03	1.1	37	0.336	0.6997	15	25.9	0	12.9	25.9	67
US to IS Circ. Weld W06 (Heat # 1725)	1.1	82.9	0.0711	0.3521	10	29.2	0	14.6	29.2	68
IS to LS Circ. Weld W05 (Heat # 895075)	1.1	52.7	2.34	1.2296	-68	64.8	0	28	56	53
Using <u>credible</u> surveillance data	2.1	27.1	2.34	1.2296	-68	33.3	0	14	28	-7
LS to Bottom Head Ring Weld W04 (Heat # 899680)	1.1	41	0.336	0.6997	10	28.7	0	14.3	28.7	67

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values. See Section 1 of this report for details.
(b) Taken from Table 5.2-2 of this report.
(c) Taken from Table 2-5 of this report.
(d) Initial RT_{NDT} values are measured and are taken from Table 3.2-1 of this report.
(e) Per Appendix A.2 of this report, the surveillance data of the forging and weld were deemed <u>credible</u>. Per the guidance of 10 CFR 50.61, the base metal σ_Δ = 17°F for Position 1.1 and, with <u>credible</u> surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with <u>credible</u> surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

7 UPPER-SHELF ENERGY CALCULATIONS

The requirements for upper-shelf energy (USE) are contained in 10 CFR 50, Appendix G (Reference 26). 10 CFR 50, Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the USE of any reactor vessel material is predicted to drop below 50 ft-lb.

Regulatory Guide 1.99, Revision 2 defines two methods that can be used to predict the decrease in USE due to irradiation. The method to be used depends on the availability of credible surveillance capsule data. For reactor vessel beltline materials that are not in the surveillance program or are not credible, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2 (Reference 1).

When two or more credible surveillance data sets become available from the reactor vessel, they may be used to determine the Charpy USE of the surveillance materials. The surveillance data are then used in conjunction with Figure 2 of the Regulatory Guide to predict the decrease in USE (Position 2.2) of the reactor vessel materials due to irradiation. If the EOLE USE values calculated using Position 2.2 are most limiting, then they must be used regardless of the credibility of the surveillance data.

The 54 EFPY (EOLE) Position 1.2 USE values of the reactor vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content, and Figure 2 in Regulatory Guide 1.99, Revision 2.

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The reduced plant surveillance data was obtained from Table 5-10 of WCAP-14993 (Reference 18) and Table 5-4 of WCAP-17014-NP (Reference 5) for McGuire Unit 1. The reduced plant surveillance data was obtained from Table 5-10 of WCAP-14799 (Reference 20) for McGuire Unit 2. The surveillance data was plotted on Regulatory Guide 1.99, Revision 2, Figure 2 (see Figures 7.1-1 and 7.2-1 of this report) using the updated surveillance capsule fluence values documented in Tables 4.1-1 and 4.2-1 of this report for McGuire Units 1 and 2, respectively. This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 EOLE USE values.

The projected USE values were calculated to determine if the McGuire Units 1 and 2 reactor vessel materials remain above the 50 ft-lb limit at EOLE. These calculations are summarized in Table 7.1-1 for McGuire Unit 1 and in Table 7.2-1 for McGuire Unit 2.

USE Conclusion

For McGuire Unit 1, the limiting USE value at 54 EFPY is 60.5 ft-lb (see Table 7.1-1); this value corresponds to IS Longitudinal Welds 2-442A,B,C (Position 2.2). For McGuire Unit 2, the limiting USE value at 54 EFPY is greater than 61.8 ft-lb (see Table 7.2-1); this value corresponds to Bottom Head Ring 03. Therefore, all of the beltline and extended beltline materials in the McGuire Units 1 and 2 reactor vessels are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOLE (54 EFPY).

7.1 MCGUIRE UNIT 1

Table 7.1-1 McGuire Unit 1 Predicted Positions 1.2 and 2.2 USE Values at 54 EFPY

Reactor Vessel Material and Identification Number	Wt. % Cu[(a)]	1/4T EOLE Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease (%)	Projected EOLE USE (ft-lb)
US Plate B5453-2	0.14	0.033	72.4	11	64.4
US Plate B5011-2	0.10	0.033	68.3	8.6	62.4
US Plate B5011-3	0.13	0.033	94.7	9.8	85.4
IS Plate B5012-1	0.11	1.525	101	23	77.8
Using surveillance data	0.11	1.525	101	11[(c)]	89.9
IS Plate B5012-2	0.14	1.525	105	26	77.7
IS Plate B5012-3	0.11	1.525	112	23	86.2
LS Plate B5013-1	0.14	1.531	95	26	70.3
LS Plate B5013-2	0.10	1.531	115	21	90.9
LS Plate B5013-3	0.10	1.531	103	21	81.4
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	0.199	0.027	112	15	95.2
Using surveillance data	0.199	0.027	112	19[(c)]	90.7
US to IS Circ. Weld 8-442 (Heat # 21935)	0.183	0.033	109	15	92.7
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	0.199	1.269	112	36	71.7
Using surveillance data	0.199	1.269	112	46[(c)]	60.5
IS to LS Circ. Weld 9-442 (Heat # 83640)	0.051	1.472	143	21	113.0
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	0.213	1.269	124	38	76.9

Notes:

(a) From Table 3.1-1 of this report.

(b) 1/4T fluence was calculated using Equation (3) of Regulatory Guide 1.99, Revision 2, and the McGuire Unit 1 reactor vessel beltline wall thickness of 8.63 inches.

(c) Percentage USE decrease is based on Position 2.2 of Regulatory Guide 1.99, Revision 2 using data from Table 4.1-1. Credibility Criterion 3 in the Discussion section of Regulatory Guide 1.99, Revision 2, indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "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 E 185-82." Regulatory Guide 1.99, Revision 2, Position 2.2 indicates that an upper-bound line drawn parallel to the existing lines (in Figure 2 of the Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE.

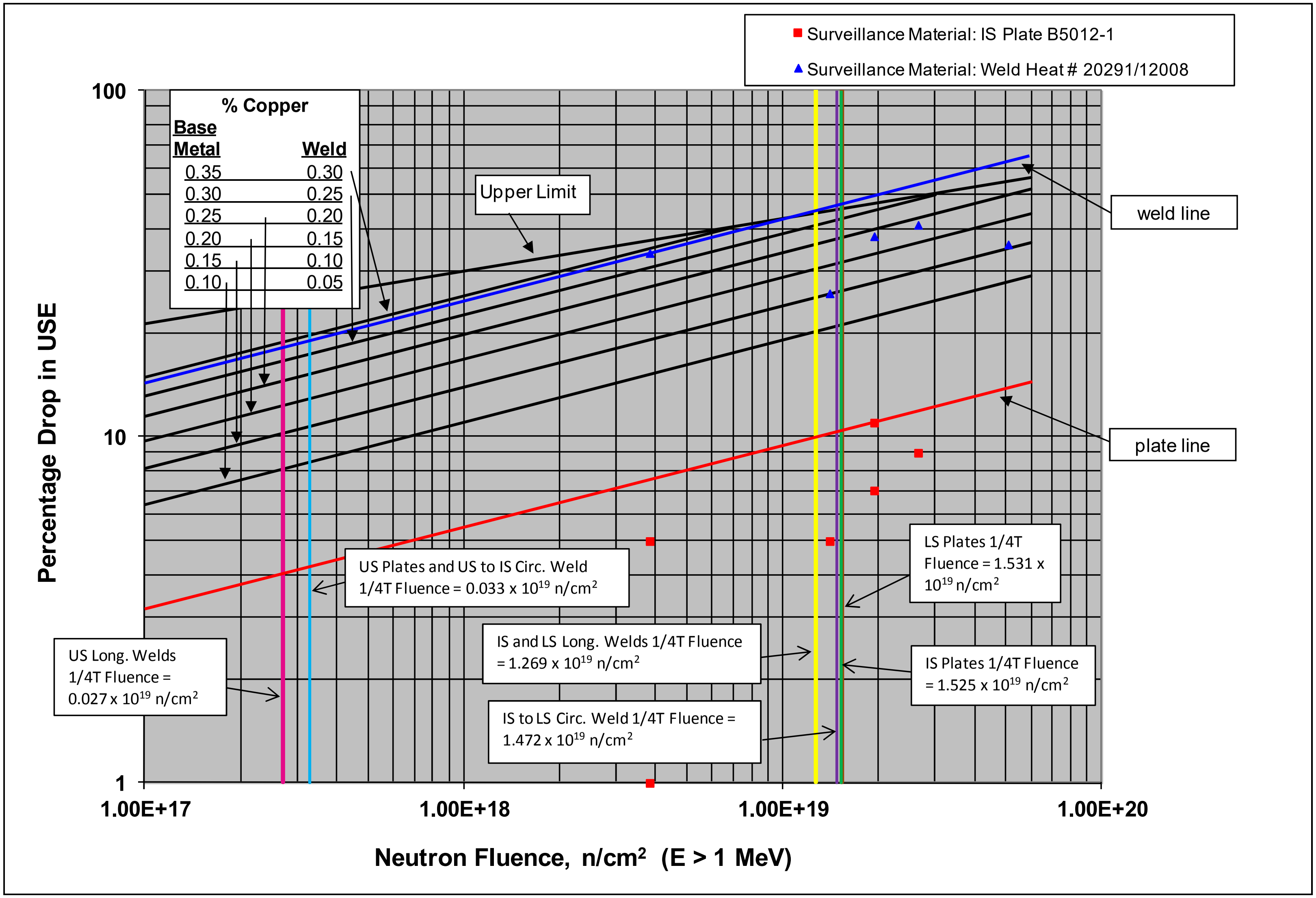

Figure 7.1-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE as a Function of Copper and Fluence for McGuire Unit 1

7.2 MCGUIRE UNIT 2

Table 7.2-1 McGuire Unit 2 Predicted Positions 1.2 and 2.2 USE Values at 54 EFPY

Reactor Vessel Material and Identification Number	Wt. % Cu[a]	1/4T EOLE Fluence[b] (x10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[a] (ft-lb)	Projected USE Decrease (%)	Projected EOLE USE (ft-lb)
US Forging 06	0.16	0.043	98	12	86.2
IS Forging 05	0.153	1.450	94	27	68.6
Using surveillance data	0.153	1.450	94	23[d]	72.4
LS Forging 04	0.15	1.492	141	27	102.9
Bottom Head Ring 03	0.06	0.202	>71	13[c]	>61.8
US to IS Circ. Weld W06 (Heat # 1725)	0.11	0.043	>71	12	>62.5
IS to LS Circ. Weld W05 (Heat # 895075)	0.039	1.408	132	21[c]	104.3
Using surveillance data	0.039	1.408	132	3.4[d]	127.5
LS to Bottom Head Ring Weld W04 (Heat # 899680)	0.03	0.202	99	13[c]	86.1

Notes:

(a) From Table 3.2-1 of this report.

(b) 1/4T fluence was calculated using Equation (3) of Regulatory Guide 1.99, Revision 2, and the McGuire Unit 2 reactor vessel beltline wall thickness of 8.465 inches.

(c) Percentage USE decrease is conservatively based on lowest Cu Wt. % chemistry line (0.05% for weld and 0.10% for base metal) delineated in Figure 2 of Regulatory Guide 1.99, Revision 2.

(d) Percentage USE decrease is based on Position 2.2 of Regulatory Guide 1.99, Revision 2 using data from Table 4.2-1. Credibility Criterion 3 in the Discussion section of Regulatory Guide 1.99, Revision 2, indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "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 E 185-82." Regulatory Guide 1.99, Revision 2, Position 2.2 indicates that an upper-bound line drawn parallel to the existing lines (in Figure 2 of the Guide) through the surveillance data points should be used in preference to the existing graph lines for determining the decrease in USE.

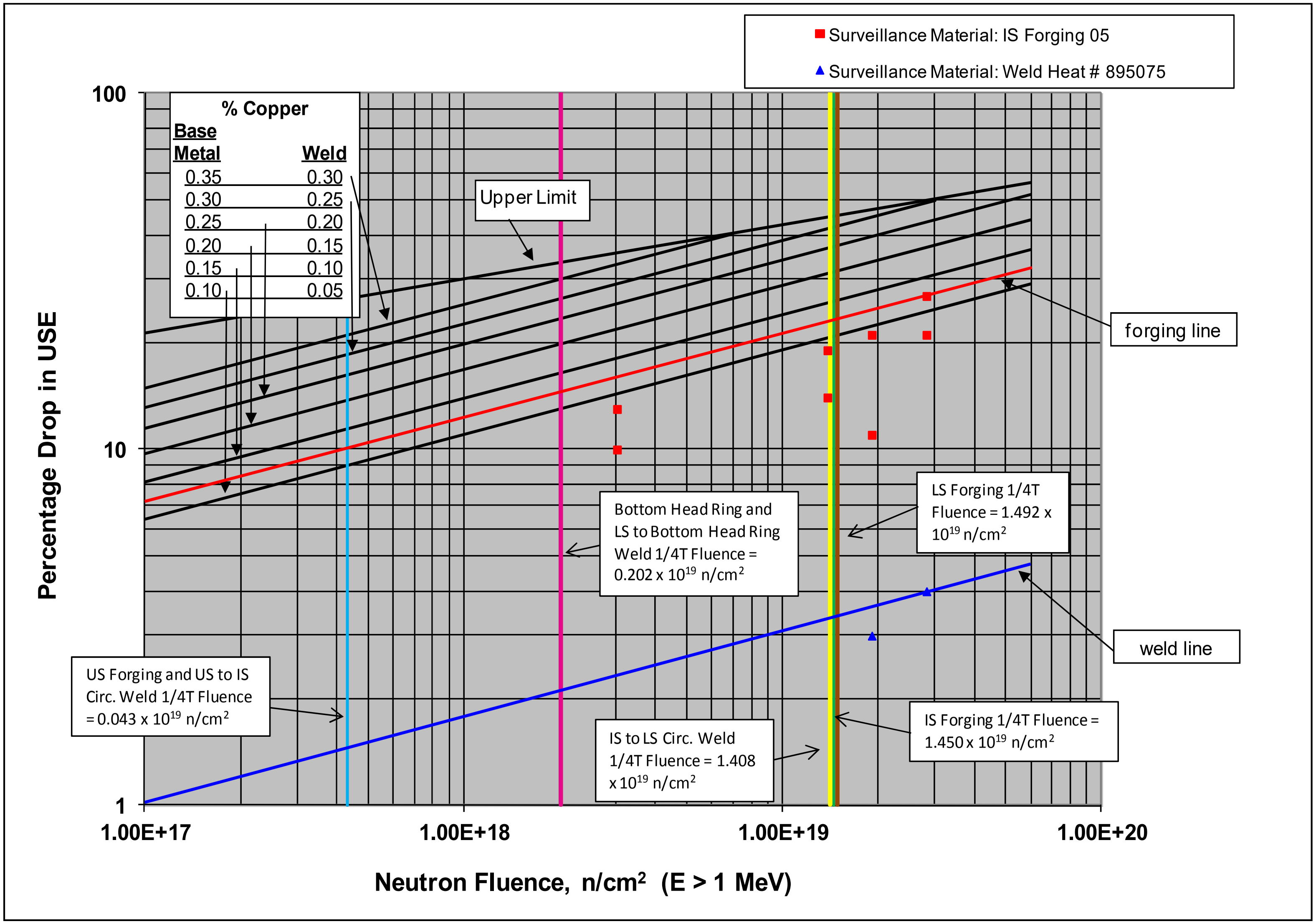

Figure 7.2-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE as a Function of Copper and Fluence for McGuire Unit 2

8 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES APPLICABILITY

Heatup and cooldown limit curves are calculated using the most limiting values of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting reactor vessel material. The most limiting reactor vessel material RT_{NDT} values are determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}).

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; 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 original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature (P-T) limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G (Reference 26), as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Reference 27).

The P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for McGuire Units 1 and 2 were previously developed in WCAP-15192, Revision 2 (Reference 14) and WCAP-15201, Revision 2 (Reference 28) for 34 EFPY. The existing 34 EFPY P-T limit curves are based on the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material. The McGuire Units 1 and 2 P-T limit curves were developed by calculating ART values utilizing the clad/base metal interface fluence that corresponded to each reactor vessel beltline material.

To confirm the applicability of the P-T limit curves developed in WCAP-15192, Revision 2 (Reference 14) for McGuire Unit 1 and in WCAP-15201, Revision 2 (Reference 28) for McGuire Unit 2, the limiting reactor vessel material ART values with consideration of the MUR power uprate must be shown to be less than the limiting beltline material ART values used in development of the existing 34 EFPY P-T limit curves contained in References 14 and 28. The Regulatory Guide 1.99, Revision 2 (Reference 1) methodology was used along with the surface fluence of Section 2 to calculate ART values for the McGuire Units 1 and 2 reactor vessel materials at 34 EFPY and 54 EFPY. The ART calculations with consideration of the MUR power uprate are summarized in Tables 8.1-1 through 8.1-4 for McGuire Unit 1 and in Tables 8.2-1 through 8.2-4 for McGuire Unit 2.

Existing P-T Limit Curves Applicability Conclusions

Comparisons of the limiting MUR power uprate ART values to those used in calculation of the existing P-T limit curves are contained in Tables 8.1-5 and 8.2-5 for McGuire Units 1 and 2, respectively. With a re-evaluation of surveillance data credibility, a recalculation of chemistry factors, and the consideration of MUR power uprate fluence projections, the applicability of the McGuire Units 1 and 2 P-T limit curves may either remain unchanged or be extended. For more detailed conclusions, refer to Sections 8.1.2 and 8.2.2 below for McGuire Units 1 and 2, respectively.

8.1 MCGUIRE UNIT 1

8.1.1 MUR Power Uprate ART Calculations

Table 8.1-1 Calculation of the McGuire Unit 1 ART Values at the 1/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	IRT_{NDT} [(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
US Plate B5453-2	1.1	99.1	0.021	0.1796	15	17.8	0	8.9	17.8	51
US Plate B5011-2	1.1	65	0.021	0.1796	27	11.7	0	5.8	11.7	50
US Plate B5011-3	1.1	89.8	0.021	0.1796	0	16.1	0	8.1	16.1	32
IS Plate B5012-1	1.1	74.2	1.001	1.0003	34	74.2	0	17	34	142
Using credible surveillance data	2.1	63.5	1.001	1.0003	34	63.5	0	8.5	17	115
IS Plate B5012-2	1.1	100.3	1.001	1.0003	0	100.3	0	17	34	134
IS Plate B5012-3	1.1	74.9	1.001	1.0003	-13	74.9	0	17	34	96
LS Plate B5013-1	1.1	99.1	1.001	1.0003	0	99.1	0	17	34	133
LS Plate B5013-2	1.1	65	1.001	1.0003	30	65.0	0	17	34	129
LS Plate B5013-3	1.1	65	1.001	1.0003	15	65.0	0	17	34	114
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.018	0.1594	-50	32.1	0	16.0	32.1	14
Using credible surveillance data	2.1	155.2	0.018	0.1594	-50	24.7	0	12.4	24.7	-1
US to IS Circ. Weld 8-442 (Heat # 21935)	1.1	170.5	0.021	0.1796	-56[(d)]	30.6	17[(d)]	15.3	45.8	20
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.834	0.9491	-50	191.1	0	28	56	197
Using credible surveillance data	2.1	155.2	0.834	0.9491	-50	147.3	0	14	28	125

Table 8.1-1 Calculation of the McGuire Unit 1 ART Values at the 1/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	IRT_{NDT} [(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
IS to LS Circ. Weld 9-442 (Heat # 83640)	1.1	37.5	0.965	0.9901	-70	37.1	0	18.6	37.1	4
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	0.834	0.9491	-50	197.6	0	28	56	204
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	0.834	0.9491	-50	176.9	0	14	28	155

Notes:

(a) Taken from Table 5.1-3 of this report.
(b) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 1 reactor vessel beltline wall thickness of 8.63 inches.
(c) Initial RT_{NDT} values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.
(d) Initial RT_{NDT} value is generic; therefore, σ_I = 17°F.
(e) Per Appendix A.1 of this report, the McGuire Unit 1 plate and weld surveillance data were deemed credible. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld surveillance data was also deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.1-2 Calculation of the McGuire Unit 1 ART Values at the 3/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	3/4T Fluence[b] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF[b]	IRT_{NDT} [c] (°F)	ΔRT_{NDT} (°F)	σ_I [c] (°F)	σ_Δ [e] (°F)	Margin (°F)	ART (°F)
US Plate B5453-2	1.1	99.1	0.008	0.0908	15	9.0	0	4.5	9.0	33
US Plate B5011-2	1.1	65	0.008	0.0908	27	5.9	0	3.0	5.9	39
US Plate B5011-3	1.1	89.8	0.008	0.0908	0	8.2	0	4.1	8.2	16
IS Plate B5012-1	1.1	74.2	0.355	0.7145	34	53.0	0	17	34	121
Using <u>credible</u> surveillance data	2.1	63.5	0.355	0.7145	34	45.4	0	8.5	17	96
IS Plate B5012-2	1.1	100.3	0.355	0.7145	0	71.7	0	17	34	106
IS Plate B5012-3	1.1	74.9	0.355	0.7145	-13	53.5	0	17	34	75
LS Plate B5013-1	1.1	99.1	0.355	0.7145	0	70.8	0	17	34	105
LS Plate B5013-2	1.1	65	0.355	0.7145	30	46.4	0	17	34	110
LS Plate B5013-3	1.1	65	0.355	0.7145	15	46.4	0	17	34	95
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.006	0.0792	-50	15.9	0	8.0	15.9	-18
Using <u>credible</u> surveillance data	2.1	155.2	0.006	0.0792	-50	12.3	0	6.1	12.3	-25
US to IS Circ. Weld 8-442 (Heat # 21935)	1.1	170.5	0.008	0.0908	-56[d]	15.5	17[d]	7.7	37.4	-3
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.296	0.6669	-50	134.3	0	28	56	140
Using <u>credible</u> surveillance data	2.1	155.2	0.296	0.6669	-50	103.5	0	14	28	82
IS to LS Circ. Weld 9-442 (Heat # 83640)	1.1	37.5	0.343	0.7049	-70	26.4	0	13.2	26.4	-17

Table 8.1-2 Calculation of the McGuire Unit 1 ART Values at the 3/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	$CF^{(a)}$ (°F)	3/4T $Fluence^{(b)}$ ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T $FF^{(b)}$	IRT_{NDT} $^{(c)}$ (°F)	ΔRT_{NDT} (°F)	σ_I $^{(c)}$ (°F)	σ_Δ $^{(e)}$ (°F)	Margin (°F)	ART (°F)
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	0.296	0.6669	-50	138.9	0	28	56	145
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	0.296	0.6669	-50	124.3	0	14	28	102

Notes:

(a) Taken from Table 5.1-3 of this report.
(b) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 1 reactor vessel beltline wall thickness of 8.63 inches.
(c) Initial RT_{NDT} values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.
(d) Initial RT_{NDT} value is generic; therefore, $\sigma_I = 17$°F.
(e) Per Appendix A.1 of this report, the McGuire Unit 1 plate and weld surveillance data were deemed credible. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld surveillance data was also deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal $\sigma_\Delta = 17$°F for Position 1.1 and, with credible surveillance data, $\sigma_\Delta = 8.5$°F for Position 2.1; the weld metal $\sigma_\Delta = 28$°F for Position 1.1 and, with credible surveillance data, $\sigma_\Delta = 14$°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.1-3 Calculation of the McGuire Unit 1 ART Values at the 1/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	1/4T Fluence[b] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[b]	IRT_{NDT} [c] (°F)	ΔRT_{NDT} (°F)	σ_I[c] (°F)	σ_Δ[e] (°F)	Margin (°F)	ART (°F)
US Plate B5453-2	1.1	99.1	0.033	0.2305	15	22.8	0	11.4	22.8	61
US Plate B5011-2	1.1	65	0.033	0.2305	27	15.0	0	7.5	15.0	57
US Plate B5011-3	1.1	89.8	0.033	0.2305	0	20.7	0	10.3	20.7	41
IS Plate B5012-1	1.1	74.2	1.525	1.1168	34	82.9	0	17	34	151
Using credible surveillance data	2.1	63.5	1.525	1.1168	34	71.0	0	8.5	17	122
IS Plate B5012-2	1.1	100.3	1.525	1.1168	0	112.0	0	17	34	146
IS Plate B5012-3	1.1	74.9	1.525	1.1168	-13	83.6	0	17	34	105
LS Plate B5013-1	1.1	99.1	1.531	1.1179	0	110.8	0	17	34	145
LS Plate B5013-2	1.1	65	1.531	1.1179	30	72.7	0	17	34	137
LS Plate B5013-3	1.1	65	1.531	1.1179	15	72.7	0	17	34	122
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.027	0.2058	-50	41.4	0	20.7	41.4	33
Using credible surveillance data	2.1	155.2	0.027	0.2058	-50	31.9	0	14	28	10
US to IS Circ. Weld 8-442 (Heat # 21935)	1.1	170.5	0.033	0.2305	-56[d]	39.3	17[d]	19.6	52.0	35
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	1.1	201.3	1.269	1.0664	-50	214.7	0	28	56	221
Using credible surveillance data	2.1	155.2	1.269	1.0664	-50	165.5	0	14	28	144
IS to LS Circ. Weld 9-442 (Heat # 83640)	1.1	37.5	1.472	1.1071	-70	41.5	0	20.8	41.5	13

Table 8.1-3 Calculation of the McGuire Unit 1 ART Values at the 1/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	IRT_{NDT} [(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	1.269	1.0664	-50	222.0	0	28	56	228
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	1.269	1.0664	-50	198.8	0	14	28	177

Notes:

(a) Taken from Table 5.1-3 of this report.

(b) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 1 reactor vessel beltline wall thickness of 8.63 inches.

(c) Initial RT_{NDT} values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.

(d) Initial RT_{NDT} value is generic; therefore, σ_I = 17°F.

(e) Per Appendix A.1 of this report, the McGuire Unit 1 plate and weld surveillance data were deemed credible. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld surveillance data was also deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.1-4 Calculation of the McGuire Unit 1 ART Values at the 3/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF(a) (°F)	3/4T Fluence(b) ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF(b)	IRT_{NDT} (c) (°F)	ΔRT_{NDT} (°F)	σ_I(c) (°F)	σ_Δ(e) (°F)	Margin (°F)	ART (°F)
US Plate B5453-2	1.1	99.1	0.012	0.1210	15	12.0	0	6.0	12.0	39
US Plate B5011-2	1.1	65	0.012	0.1210	27	7.9	0	3.9	7.9	43
US Plate B5011-3	1.1	89.8	0.012	0.1210	0	10.9	0	5.4	10.9	22
IS Plate B5012-1	1.1	74.2	0.542	0.8285	34	61.5	0	17	34	129
Using <u>credible</u> surveillance data	2.1	63.5	0.542	0.8285	34	52.6	0	8.5	17	104
IS Plate B5012-2	1.1	100.3	0.542	0.8285	0	83.1	0	17	34	117
IS Plate B5012-3	1.1	74.9	0.542	0.8285	-13	62.1	0	17	34	83
LS Plate B5013-1	1.1	99.1	0.544	0.8296	0	82.2	0	17	34	116
LS Plate B5013-2	1.1	65	0.544	0.8296	30	53.9	0	17	34	118
LS Plate B5013-3	1.1	65	0.544	0.8296	15	53.9	0	17	34	103
US Long. Welds 1-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.010	0.1062	-50	21.4	0	10.7	21.4	-7
Using <u>credible</u> surveillance data	2.1	155.2	0.010	0.1062	-50	16.5	0	8.2	16.5	-17
US to IS Circ. Weld 8-442 (Heat # 21935)	1.1	170.5	0.012	0.1210	-56(d)	20.6	17(d)	10.3	39.8	4
IS Long. Welds 2-442A,B,C (Heat # 20291/12008)	1.1	201.3	0.451	0.7781	-50	156.6	0	28	56	163
Using <u>credible</u> surveillance data	2.1	155.2	0.451	0.7781	-50	120.8	0	14	28	99
IS to LS Circ. Weld 9-442 (Heat # 83640)	1.1	37.5	0.522	0.8187	-70	30.7	0	15.4	30.7	-9

Table 8.1-4 Calculation of the McGuire Unit 1 ART Values at the 3/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	3/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	IRT_{NDT} [(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
LS Long. Welds 3-442A,B,C (Heat # 21935/12008)	1.1	208.2	0.451	0.7781	-50	162.0	0	28	56	168
Using credible Diablo Canyon Unit 2 surveillance data	2.1	186.4	0.451	0.7781	-50	145.1	0	14	28	123

Notes:

(a) Taken from Table 5.1-3 of this report.

(b) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 1 reactor vessel beltline wall thickness of 8.63 inches.

(c) Initial RT_{NDT} values are taken from Table 3.1-1 of this report and are measured values, unless otherwise noted.

(d) Initial RT_{NDT} value is generic; therefore, σ_I = 17°F.

(e) Per Appendix A.1 of this report, the McGuire Unit 1 plate and weld surveillance data were deemed credible. Per Appendix A.2 of WCAP-17315-NP (Reference 19), the Diablo Canyon Unit 2 weld surveillance data was also deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

8.1.2 P-T Limit Curves Applicability Evaluation

Tables 8.1-1 through 8.1-4 above summarize the 1/4T and 3/4T ART calculations for McGuire Unit 1. The limiting 34 EFPY 1/4T ART value for McGuire Unit 1 corresponds to LS Long. Welds 3-442A,B,C (Heat # 21935/12008) using credible Diablo Canyon Unit 2 surveillance data (Position 2.1). The limiting 34 EFPY 3/4T ART value for McGuire Unit 1 corresponds to LS Plate B5013-2. The limiting 1/4T and 3/4T ART values at 54 EFPY for McGuire Unit 1 correspond to LS Long. Welds 3-442A,B,C (Heat # 21935/12008) using credible Diablo Canyon Unit 2 surveillance data (Position 2.1).

The applicability of the existing 34 EFPY P-T limit curves, contained in WCAP-15192, Revision 2 (Reference 14) for McGuire Unit 1, is evaluated by comparing the updated ART values contained in Section 8.1.1 with those used in the Reference 14 calculations. The existing 34 EFPY P-T limit curves for McGuire Unit 1 are based on the limiting beltline material ART values, which are influenced by both fluence and initial material properties of that material. Using the MUR power uprate fluence projections, the 1/4T and 3/4T ART values were recalculated in Tables 8.1-1 through 8.1-4 as part of this applicability evaluation for McGuire Unit 1. Since the capsule fluence values were also revised as part of the MUR power uprate, the Position 2.1 chemistry factor values were updated in Section 5 of this report. The comparison of limiting ART values is contained in Table 8.1-5 for McGuire Unit 1.

Table 8.1-5 Summary of the McGuire Unit 1 Limiting ART Values used in the Applicability Evaluation of the Current Reactor Vessel Heatup and Cooldown Curves

Limiting Material	1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
	Existing 34 EFPY Curves documented in WCAP-15192, Revision 2	MUR Uprate Evaluation at 54 EFPY (Table 8.1-3)	Existing 34 EFPY Curves documented in WCAP-15192, Revision 2	MUR Uprate Evaluation at 54 EFPY (Table 8.1-4)
LS Long. Welds 3-442A,B,C (Heat # 21935/12008) Using Diablo Canyon Unit 2 Surveillance Data	202	177	146	123

Table 8.1-5 above compares the MUR power uprate limiting ART values at 54 EFPY to the limiting ART values used in development of the existing 34 EFPY P-T limit curves that are documented in WCAP-15192, Revision 2 (Reference 14). The limiting ART values used to develop the existing P-T limit curves are documented in Table 10 of Reference 14.

The MUR power uprate limiting ART values at 54 EFPY are bounded by the limiting ART values used to develop the existing 34 EFPY P-T limit curves. Even at 54 EFPY, considerable margin remains. This is primarily due to the re-evaluation of the Diablo Canyon Unit 2 surveillance weld data that was performed in WCAP-17315-NP (Reference 19). Therefore, the existing McGuire Unit 1 P-T limit curves may be deemed applicable through 54 EFPY.

For McGuire Unit 1, it is concluded that the existing 34 EFPY P-T limit curves do not require a reduction of the applicability date. Furthermore, based on the MUR power uprate evaluation, Duke Energy may instead choose to extend the applicability of the existing McGuire Unit 1 P-T limit curves. The new applicability date with consideration of the MUR power uprate fluence evaluations is 54 EFPY.

8.2 MCGUIRE UNIT 2

8.2.1 MUR Power Uprate ART Calculations

Table 8.2-1 Calculation of the McGuire Unit 2 ART Values at the 1/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	1/4T Fluence[b] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[b]	IRT_{NDT}[c] (°F)	ΔRT_{NDT} (°F)	σ_I[c] (°F)	σ_Δ[d] (°F)	Margin (°F)	ART (°F)
US Forging 06	1.1	123.9	0.029	0.2148	25	26.6	0	13.3	26.6	78
IS Forging 05	1.1	117.2	0.975	0.9929	-4	116.4	0	17	34	146
Using credible surveillance data	2.1	85.5	0.975	0.9929	-4	84.9	0	8.5	17	98
LS Forging 04	1.1	115.8	0.999	0.9997	-30	115.8	0	17	34	120
Bottom Head Ring 03	1.1	37	0.136	0.4812	15	17.8	0	8.9	17.8	51
US to IS Circ. Weld W06 (Heat # 1725)	1.1	82.9	0.029	0.2148	10	17.8	0	8.9	17.8	46
IS to LS Circ. Weld W05 (Heat # 895075)	1.1	52.7	0.945	0.9841	-68	51.9	0	25.9	51.9	36
Using credible surveillance data	2.1	27.1	0.945	0.9841	-68	26.7	0	13.3	26.7	-15
LS to Bottom Head Ring Weld W04 (Heat # 899680)	1.1	41	0.136	0.4812	10	19.7	0	9.9	19.7	49

Notes:

(a) Taken from Table 5.2-2 of this report.

(b) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 2 reactor vessel beltline wall thickness of 8.465 inches.

(c) Initial RT_{NDT} values are measured and are taken from Table 3.2-1 of this report.

(d) Per Appendix A.2 of this report, the surveillance data of the forging and weld were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.2-2 Calculation of the McGuire Unit 2 ART Values at the 3/4T Location for 34 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	3/4T Fluence[b] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF[b]	IRT_{NDT} [c] (°F)	ΔRT_{NDT} (°F)	σ_I[c] (°F)	σ_Δ[d] (°F)	Margin (°F)	ART (°F)
US Forging 06	1.1	123.9	0.010	0.1130	25	14.0	0	7.0	14.0	53
IS Forging 05	1.1	117.2	0.353	0.7127	-4	83.5	0	17	34	114
Using credible surveillance data	2.1	85.5	0.353	0.7127	-4	60.9	0	8.5	17	74
LS Forging 04	1.1	115.8	0.362	0.7192	-30	83.3	0	17	34	87
Bottom Head Ring 03	1.1	37	0.049	0.2903	15	10.7	0	5.4	10.7	36
US to IS Circ. Weld W06 (Heat # 1725)	1.1	82.9	0.010	0.1130	10	9.4	0	4.7	9.4	29
IS to LS Circ. Weld W05 (Heat # 895075)	1.1	52.7	0.342	0.7045	-68	37.1	0	18.6	37.1	6
Using credible surveillance data	2.1	27.1	0.342	0.7045	-68	19.1	0	9.5	19.1	-30
LS to Bottom Head Ring Weld W04 (Heat # 899680)	1.1	41	0.049	0.2903	10	11.9	0	6.0	11.9	34

Notes:

(a) Taken from Table 5.2-2 of this report.
(b) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 2 reactor vessel beltline wall thickness of 8.465 inches.
(c) Initial RT_{NDT} values are measured and are taken from Table 3.2-1 of this report.
(d) Per Appendix A.2 of this report, the surveillance data of the forging and weld were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.2-3 Calculation of the McGuire Unit 2 ART Values at the 1/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	1/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	IRT_{NDT}[(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART (°F)
US Forging 06	1.1	123.9	0.043	0.2688	25	33.3	0	16.7	33.3	92
IS Forging 05	1.1	117.2	1.450	1.1031	-4	129.3	0	17	34	159
Using credible surveillance data	2.1	85.5	1.450	1.1031	-4	94.3	0	8.5	17	107
LS Forging 04	1.1	115.8	1.492	1.1109	-30	128.6	0	17	34	133
Bottom Head Ring 03	1.1	37	0.202	0.5720	15	21.2	0	10.6	21.2	57
US to IS Circ. Weld W06 (Heat # 1725)	1.1	82.9	0.043	0.2688	10	22.3	0	11.1	22.3	55
IS to LS Circ. Weld W05 (Heat # 895075)	1.1	52.7	1.408	1.0950	-68	57.7	0	28.0	56.0	46
Using credible surveillance data	2.1	27.1	1.408	1.0950	-68	29.7	0	14.0	28.0	-10
LS to Bottom Head Ring Weld W04 (Heat # 899680)	1.1	41	0.202	0.5720	10	23.5	0	11.7	23.5	57

Notes:

(a) Taken from Table 5.2-2 of this report.
(b) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 2 reactor vessel beltline wall thickness of 8.465 inches.
(c) Initial RT_{NDT} values are measured and are taken from Table 3.2-1 of this report.
(d) Per Appendix A.2 of this report, the surveillance data of the forging and weld were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 8.2-4 Calculation of the McGuire Unit 2 ART Values at the 3/4T Location for 54 EFPY

Reactor Vessel Material and Identification Number	R.G. 1.99, Rev. 2 Position	CF[(a)] (°F)	3/4T Fluence[(b)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	IRT_{NDT} [(c)] (°F)	ΔRT_{NDT} (°F)	σ_I[(c)] (°F)	σ_Δ[(d)] (°F)	Margin (°F)	ART (°F)
US Forging 06	1.1	123.9	0.015	0.1464	25	18.1	0	9.1	18.1	61
IS Forging 05	1.1	117.2	0.525	0.8201	-4	96.1	0	17	34	126
Using credible surveillance data	2.1	85.5	0.525	0.8201	-4	70.1	0	8.5	17	83
LS Forging 04	1.1	115.8	0.540	0.8280	-30	95.9	0	17	34	100
Bottom Head Ring 03	1.1	37	0.073	0.3574	15	13.2	0	6.6	13.2	41
US to IS Circ. Weld W06 (Heat # 1725)	1.1	82.9	0.015	0.1464	10	12.1	0	6.1	12.1	34
IS to LS Circ. Weld W05 (Heat # 895075)	1.1	52.7	0.510	0.8120	-68	42.8	0	21.4	42.8	18
Using credible surveillance data	2.1	27.1	0.510	0.8120	-68	22.0	0	11.0	22.0	-24
LS to Bottom Head Ring Weld W04 (Heat # 899680)	1.1	41	0.073	0.3574	10	14.7	0	7.3	14.7	39

Notes:

(a) Taken from Table 5.2-2 of this report.
(b) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the McGuire Unit 2 reactor vessel beltline wall thickness of 8.465 inches.
(c) Initial RT_{NDT} values are measured and are taken from Table 3.2-1 of this report.
(d) Per Appendix A.2 of this report, the surveillance data of the forging and weld were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and, with credible surveillance data, σ_Δ = 8.5°F for Position 2.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

8.2.2 P-T Limit Curves Applicability Evaluation

Tables 8.2-1 through 8.2-4 above summarize the 1/4T and 3/4T ART calculations for McGuire Unit 2. The limiting 1/4T and 3/4T ART values at 34 EFPY and 54 EFPY for McGuire Unit 2 correspond to LS Forging 04.

The applicability of the existing 34 EFPY P-T limit curves, contained in WCAP-15201, Revision 2 (Reference 28) for McGuire Unit 2, is evaluated by comparing the updated ART values contained in Section 8.2.1 with those used in the Reference 28 calculations. The existing 34 EFPY P-T limit curves for McGuire Unit 2 are based on the limiting beltline material ART values, which are influenced by both fluence and initial material properties of that material. Using the MUR power uprate fluence projections, the 1/4T and 3/4T ART values were recalculated in Tables 8.2-1 through 8.2-4 as part of this applicability evaluation for McGuire Unit 2. Since the capsule fluence values were also revised as part of the MUR power uprate, the Position 2.1 chemistry factor values were updated in Section 5 of this report. The comparison of limiting ART values is contained in Table 8.2-5 for McGuire Unit 2.

Table 8.2-5 Summary of the McGuire Unit 2 Limiting ART Values used in the Applicability Evaluation of the Current Reactor Vessel Heatup and Cooldown Curves

Limiting Material	1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
	Existing 34 EFPY Curves documented in WCAP-15201, Revision 2	MUR Uprate Evaluation at 34 EFPY (Table 8.2-1)	Existing 34 EFPY Curves documented in WCAP-15201, Revision 2	MUR Uprate Evaluation at 34 EFPY (Table 8.2-2)
LS Forging 04	123	120	91	87

Table 8.2-5 above compares the MUR power uprate limiting ART values at 34 EFPY to the limiting ART values used in development of the existing 34 EFPY P-T limit curves that are documented in WCAP-15201, Revision 2 (Reference 28). The limiting ART values used to develop the existing P-T limit curves are documented in Table 10 of Reference 28.

The MUR power uprate limiting ART values at 34 EFPY are bounded by the limiting ART values used to develop the existing 34 EFPY P-T limit curves. Therefore, the existing McGuire Unit 2 P-T limit curves remain valid through 34 EFPY.

Since some margin remains, the extended applicability of the existing P-T limit curves is determined with consideration of MUR power uprate conditions. LS Forging 04 is proven to be the limiting material based on the calculations presented in Tables 8.2-1 through 8.2-4. Surveillance data is not used for this material, and the initial RT_{NDT} and chemistry factor have not changed from the values documented in Table 8 of WCAP-15201, Revision 2 (Reference 28). Therefore, only the surface fluence value must be

considered in this assessment. The surface fluence for LS Forging 04 per Table 6 of Reference 28 is 1.85×10^{19} n/cm^2 (E > 1.0 MeV). Using the updated MUR power uprate fluence projections documented in Table 2-5 of this report, the cycle time that corresponds to 1.85×10^{19} n/cm^2 is approximately 38.6 EFPY for the LS Forging 04 material. Therefore, the existing McGuire Unit 2 P-T limit curves may be deemed applicable through 38.6 EFPY.

For McGuire Unit 2, it is concluded that the existing 34 EFPY P-T limit curves do not require a reduction of the applicability date. Furthermore, based on the MUR power uprate evaluation, Duke Energy may instead choose to extend the applicability of the existing McGuire Unit 2 P-T limit curves. The new applicability date with consideration of the MUR power uprate fluence evaluations is 38.6 EFPY.

9 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES

9.1 MCGUIRE UNIT 1

Table 9.1-1 summarizes the removal of the six surveillance capsules from the McGuire Unit 1 reactor vessel, meeting the requirements of ASTM E185-82 (Reference 4), as required by 10 CFR 50, Appendix H (Reference 29).

Table 9.1-1 McGuire Unit 1 Surveillance Capsule Withdrawal Summary

Capsule	Capsule Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)
U	56°	4.96	1.09	0.382
X	236°	4.83	4.30	1.40
V	58.5°	4.15	7.24	1.93
Z[(c)]	304°	4.75	7.24	2.21
Y	238.5°	4.20	10.21	2.65
W[(d)]	124°	4.92	19.22	5.08

Notes:

(a) Updated as part of the MUR power uprate fluence evaluation.
(b) EFPY from plant startup.
(c) Capsule Z was removed from the vessel at 7.24 EFPY and the dosimeters were tested. The material specimens were not tested and are being stored for potential future testing or further irradiation.
(d) Capsule W was removed from the vessel at 19.22 EFPY. The weld material specimens were tested. The remaining specimens were not tested and are being stored for potential future testing or further irradiation.

Based on the limiting ΔRT_{NDT} value (224.7°F) for McGuire Unit 1 documented in Table 6.1-1 of this report, McGuire Unit 1 is required to withdraw five surveillance capsules, with the fifth capsule able to be held without testing following withdrawal. To date, four capsules were withdrawn and tested per ASTM E185-82. The fifth capsule was withdrawn and the weld specimens were tested. The sixth capsule was withdrawn and the dosimeters were tested, but the material specimens were not tested and are being stored for potential future testing or further irradiation.

The withdrawal schedule for these surveillance capsules meets the current recommendations of ASTM E185-82 as required by 10 CFR Part 50, Appendix H for a license extension through 60 years of operation, with consideration of the MUR power uprate. Furthermore, although McGuire Unit 1 does not have any capsules remaining in the reactor vessel, an Ex-Vessel Neutron Dosimetry (EVND) program is in place, meeting the requirements of Section XI.M31 of the Generic Aging Lessons Learned (GALL) Report (Reference 10). If necessary in the future, the removed and untested specimens may either be tested or re-inserted into the reactor vessel to be further irradiated. Decisions regarding the stored specimens should be made when Duke Energy seeks to obtain a license extension to 80 years of operation.

9.2 MCGUIRE UNIT 2

Table 9.2-1 summarizes the removal of the six surveillance capsules from the McGuire Unit 2 reactor vessel, meeting the requirements of ASTM E185-82 (Reference 4), as required by 10 CFR 50, Appendix H (Reference 29).

Table 9.2-1 McGuire Unit 2 Surveillance Capsule Withdrawal Summary

Capsule	Capsule Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)
V	58.5°	4.16	1.03	0.302
X	236°	4.81	4.16	1.38
U	56°	4.66	6.05	1.90
Y[(c)]	238.5°	4.03	7.18	1.94
Z[(c)]	304°	4.60	7.18	2.21
W	124°	4.64	9.44	2.82

Notes:

(a) Updated as part of the MUR power uprate fluence evaluation.
(b) EFPY from plant startup.
(c) Capsules Y and Z were removed from the vessel at 7.18 EFPY and the dosimeters were tested. The material specimens were not tested and are being stored for potential future testing or further irradiation.

Based on the limiting ΔRT_{NDT} value (144.1°F) for McGuire Unit 2 documented in Table 6.2-1 of this report, McGuire Unit 2 is required to withdraw four surveillance capsules, with the fourth capsule able to be held without testing following withdrawal. To date, four capsules were withdrawn and tested per ASTM E185-82. The other two capsules were withdrawn and the dosimeters were tested, but the material specimens were not tested and are being stored for potential future testing or further irradiation.

The withdrawal schedule for these surveillance capsules meets the current recommendations of ASTM E185-82 as required by 10 CFR Part 50, Appendix H for a license extension through 60 years of operation, with consideration of the MUR power uprate. Furthermore, although McGuire Unit 2 does not have any capsules remaining in the reactor vessel, an EVND program is in place, meeting the requirements of Section XI.M31 of the GALL Report (Reference 10). If necessary in the future, the removed and untested specimens may either be tested or re-inserted into the reactor vessel to be further irradiated. Decisions regarding the stored specimens should be made when Duke Energy seeks to obtain a license extension to 80 years of operation.

10 REFERENCES

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APPENDIX A
SURVEILLANCE DATA CREDIBILITY EVALUATION

A.1 MCGUIRE UNIT 1

Introduction

Regulatory Guide 1.99, Revision 2 (Reference A.1-1) 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 C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been five surveillance capsules removed and tested from the McGuire Unit 1 reactor vessel. Note that only the weld specimens were tested from the fifth capsule removed (Capsule W). To use these surveillance data sets, they must be shown to be credible. In accordance with Regulatory Guide 1.99, Revision 2, 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 Regulatory Guide 1.99, Revision 2, to the McGuire Unit 1 reactor vessel surveillance data and determine if that surveillance data is credible.

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," (Reference A.1-2) 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 McGuire Unit 1 reactor vessel beltline region consists of the following materials:

1. Intermediate Shell Plates B5012-1, B5012-2, and B5012-3
2. Lower Shell Plates B5013-1, B5013-2, and B5013-3
3. Intermediate Shell Longitudinal Weld Seams 2-442A, B, and C were fabricated with weld wire Heat # 20291/12008, Flux Type 1092, Lot # 3854
4. Lower Shell Longitudinal Weld Seams 3-442A, B, and C were fabricated with weld wire Heat # 21935/12008, Flux Type 1092, Lot # 3889

5. Intermediate to Lower Shell Circumferential Weld Seam 9-442 was fabricated with weld wire Heat # 83640, Flux Type 0091, Lot # 3490

The McGuire Unit 1 surveillance program utilizes longitudinal and transverse test specimens from Intermediate Shell Plate B5012-1. The surveillance weld metal was fabricated with weld wire Heat # 20291/12008, Flux Type 1092, Lot # 3854.

At the time when the surveillance program was selected it was believed that copper and phosphorus were the elements most important to embrittlement of reactor vessel steels. Intermediate Shell Plate B5012-1 had the highest initial RT_{NDT} and one of the lowest initial upper-shelf energy values of all plate materials in the beltline region. Weld seams 2-442A, B, and C had the highest initial RT_{NDT} and the lowest initial upper-shelf energy value of all weld materials in the beltline region. In addition, Intermediate Shell Plate B5012-1 and weld seams 2-442A, B, and C had approximately the same copper and phosphorus content of the other beltline materials. Based on their initial RT_{NDT} values and initial upper-shelf energy values, Intermediate Shell Plate B5012-1 and weld seams 2-442A, B, and C were chosen for the surveillance program.

Therefore, the materials selected for use in the McGuire 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.

<u>Based on the above discussion, Criterion 1 is met for the McGuire Unit 1 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 for all of the McGuire Unit 1 surveillance materials are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-14993 (Reference A.1-3). Additionally, more recent plots of Charpy energy versus temperature for the unirradiated and irradiated conditions for the weld metal are presented in Section 5 and Appendix C of WCAP-17014-NP (Reference A.1-4).

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 McGuire Unit 1 surveillance materials unambiguously.

<u>Hence, Criterion 2 is met for the McGuire Unit 1 surveillance program.</u>

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 (Reference A.1-5).

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 welds and less than 17°F for the plate.

The McGuire Unit 1 Intermediate Shell Plate B5012-1 and surveillance weld will be evaluated for credibility. The weld is made from weld wire Heat # 20291/12008. This weld metal is not in any other surveillance program.

Table A.1-1 contains the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 (Reference A.1-6). At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation listed above for the McGuire Unit 1 surveillance weld metal and plate materials.

Table A.1-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation for McGuire Unit 1

Material	Capsule	Capsule $f^{(a)}$ ($x10^{19}$ n/cm^2, E > 1.0 MeV)	$FF^{(b)}$	$\Delta RT_{NDT}^{(c)}$ (°F)	$FF*\Delta RT_{NDT}$ (°F)	FF^2
IS Plate B5012-1 (Longitudinal)	U	0.382	0.734	30.95	22.71	0.538
	X	1.40	1.093	33.51	36.64	1.196
	V	1.93	1.180	81.01	95.57	1.392
	Y	2.65	1.261	93.10	117.37	1.589
IS Plate B5012-1 (Transverse)	U	0.382	0.734	48.44	35.54	0.538
	X	1.40	1.093	60.69	66.36	1.196
	V	1.93	1.180	74.60	88.01	1.392
	Y	2.65	1.261	108.58	136.88	1.589
				SUM:	599.08	9.430
	$CF_{IS\ Plate} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (599.08) \div (9.430) = \mathbf{63.5°F}$					
Surveillance Weld Material (Heat # 20291/12008)	U	0.382	0.734	161.1	118.20	0.538
	X	1.40	1.093	170.6	186.53	1.196
	V	1.93	1.180	179.8	212.12	1.392
	Y	2.65	1.261	190.2	239.78	1.589
	W	5.08	1.405	208.0	292.34	1.975
				SUM:	1048.97	6.690
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1048.97) \div (6.690) = \mathbf{156.8°F}$					

Notes:

(a) f = capsule fluence taken from Table 4.1-1 of this report.

(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 Table 4.1-1 of this report. These measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Reg. 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 McGuire Unit 1 data is being considered; therefore, no temperature adjustment is required.

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 A.1-2.

Table A.1-2 Best-Fit Evaluation for McGuire Unit 1 Surveillance Materials Only

Material	Capsule	CF (Slope$_{best-fit}$) (°F)	Capsule f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT}[(a)] (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
IS Plate B5012-1 (Longitudinal)	U	63.5	0.382	0.734	30.95	46.6	15.7	Yes
	X	63.5	1.40	1.093	33.51	69.5	36.0	No
	V	63.5	1.93	1.180	81.01	75.0	6.1	Yes
	Y	63.5	2.65	1.261	93.10	80.1	13.0	Yes
IS Plate B5012-1 (Transverse)	U	63.5	0.382	0.734	48.44	46.6	1.8	Yes
	X	63.5	1.40	1.093	60.69	69.5	8.8	Yes
	V	63.5	1.93	1.180	74.60	75.0	0.4	Yes
	Y	63.5	2.65	1.261	108.58	80.1	28.5	No
Surveillance Weld Material (Heat # 20291/12008)	U	156.8	0.382	0.734	161.1	115.0	46.1	No
	X	156.8	1.40	1.093	170.6	171.4	0.8	Yes
	V	156.8	1.93	1.180	179.8	185.0	5.2	Yes
	Y	156.8	2.65	1.261	190.2	197.7	7.5	Yes
	W	156.8	5.08	1.405	208.0	220.7	12.7	Yes

Notes:

(a) Predicted $\Delta RT_{NDT} = CF_{best-fit} * FF$.
(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 Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal. Table A.1-2 indicates that six of the eight surveillance data points fall within the +/- 1σ of 17°F scatter band for surveillance base metals; therefore, the plate data is deemed "credible" per the third criterion.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 28°F for weld metal. Table A.1-2 indicates that four of the five surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material is deemed "credible" per the third criterion.

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 McGuire 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 such that the temperatures will not differ by more than 25˚F.

<u>Hence, Criterion 4 is met for the McGuire Unit 1 surveillance program.</u>

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

<u>Conclusion</u>

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2, Section B, and the application of engineering judgment, the McGuire Unit 1 surveillance plate and weld materials are deemed <u>credible</u>.

Appendix A.1 References

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

A.1-2 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

A.1-3 WCAP-14993, Revision 0, "Analysis of Capsule Y from the Duke Power Company McGuire Unit 1 Reactor Vessel Radiation Surveillance Program," T. J. Laubham et al., December 1998.

A.1-4 WCAP-17014-NP, Revision 0, "Analysis of Capsule W from the McGuire Unit No. 1 Reactor Vessel Radiation Surveillance Program for the Calvert Cliffs Unit 1 Vessel Weld Metal," C. C. Heinecke et al., December 2008.

A.1-5 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society of Testing and Materials, 1982.

A.1-6 K. Wichman, M. Mitchell, and A. Hiser, USNRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues,* February 12, 1998.

A.2 MCGUIRE UNIT 2

Introduction

Regulatory Guide 1.99, Revision 2 (Reference A.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 C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 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 the McGuire Unit 2 reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with Regulatory Guide 1.99, Revision 2, 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 Regulatory Guide 1.99, Revision 2, to the McGuire Unit 2 reactor vessel surveillance data and determine if that surveillance data is credible.

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," (Reference A.2-2) 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 McGuire Unit 2 reactor vessel beltline region consists of the following materials:

1. Intermediate Shell Forging 05
2. Lower Shell Forging 04
3. Intermediate Shell Forging to Lower Shell Forging Circumferential Weld Seam W05 (Weld Wire Heat # 895075, Flux Type Grau L.O. # LW320, Flux Lot # P46)

The McGuire Unit 2 surveillance program utilizes tangential and axial test specimens from Intermediate Shell Forging 05. The surveillance weld metal was fabricated with weld wire Heat # 895075, Flux Type Grau L.O., Flux Lot # P46.

Intermediate Shell Forging 05 had the highest initial RT_{NDT} and lowest initial upper-shelf energy out of the two beltline forgings in the McGuire Unit 2 reactor vessel. Thus, it was selected as the surveillance base metal.

The weld material in the McGuire Unit 2 surveillance program was made of the same material as the reactor vessel beltline circumferential weld. In accordance with the definition of the reactor vessel beltline at that time, this was the only weld in the beltline region.

Therefore, the materials selected for use in the McGuire 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, Criterion 1 is met for the McGuire 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-14799 (Reference A.2-3).

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 McGuire Unit 2 surveillance materials unambiguously.

<u>Hence, Criterion 2 is met for the McGuire Unit 2 surveillance program.</u>

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 (Reference A.2-4).

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 Regulatory Guide 1.99, Revision 2. In addition, 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 (Reference A.2-5). At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation for the McGuire Unit 2 surveillance forging material. However, McGuire Unit 2 has a weld that will be evaluated for credibility using the guidance for the appropriate case as explained in Reference A.2-5. Weld Heat # 895075 pertains to IS to LS circumferential weld W05 in the McGuire Unit 2 reactor vessel. This weld heat is contained in the McGuire Unit 2 surveillance program as well as the Catawba Unit 1 and Watts Bar Unit 1 surveillance programs. NRC Case 4 per Reference A.2-5 is entitled "Surveillance Data from Plant and Other Sources" and most closely represents the situation for McGuire Unit 2 weld Heat # 895075.

Case 1: IS Forging 05 and

Case 4: Weld Heat # 895075 (McGuire Unit 2 data only)

Following the NRC Case 4 guidelines, the McGuire Unit 2 surveillance weld metal (Heat # 895075) will be evaluated first using McGuire Unit 2 data only. The McGuire Unit 2 surveillance forging data will also be evaluated here since only McGuire Unit 2 data is considered. Table A.2-1 contains these evaluations.

Table A.2-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using McGuire Unit 2 Surveillance Capsule Data Only

Material	Capsule	Capsule f[(a)] (x10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
IS Forging 05 (Axial)	V	0.302	0.672	58.64	39.41	0.452
	X	1.38	1.089	91.12	99.27	1.187
	U	1.90	1.176	84.14	98.92	1.382
	W	2.82	1.276	130.33	166.28	1.628
IS Forging 05 (Tangential)	V	0.302	0.672	68.97	46.35	0.452
	X	1.38	1.089	98.28	107.07	1.187
	U	1.90	1.176	91.18	107.20	1.382
	W	2.82	1.276	102.03	130.17	1.628
				SUM:	794.67	9.297
	$CF_{IS\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (794.67) \div (9.297) =$ **85.5°F**					
Surveillance Weld Material (Heat # 895075)	V	0.302	0.672	38.51	25.88	0.452
	X	1.38	1.089	35.93	39.14	1.187
	U	1.90	1.176	23.81	27.99	1.382
	W	2.82	1.276	43.76	55.83	1.628
				SUM:	148.85	4.648
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (148.85) \div (4.648) =$ **32.0°F**					

Notes:

(a) f = capsule fluence taken from Table 4.2-1 of this report.
(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 Table 4.2-1 of this report. These measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Reg. 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 McGuire Unit 2 data is being considered; therefore, no temperature adjustment is required.

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 A.2-2.

Table A.2-2 Best-Fit Evaluation for McGuire Unit 2 Surveillance Materials Only

Material	Capsule	CF (Slope$_{best-fit}$) (°F)	Capsule f (x10^{19} n/cm^2, E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT}[(a)] (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
IS Forging 05 (Axial)	V	85.5	0.302	0.672	58.64	57.4	1.2	Yes
	X	85.5	1.38	1.089	91.12	93.1	2.0	Yes
	U	85.5	1.90	1.176	84.14	100.5	16.4	Yes
	W	85.5	2.82	1.276	130.33	109.1	21.3	No
IS Forging 05 (Tangential)	V	85.5	0.302	0.672	68.97	57.4	11.5	Yes
	X	85.5	1.38	1.089	98.28	93.1	5.2	Yes
	U	85.5	1.90	1.176	91.18	100.5	9.3	Yes
	W	85.5	2.82	1.276	102.03	109.1	7.0	Yes
Surveillance Weld Material (Heat # 895075)	V	32.0	0.302	0.672	38.51	21.5	17.0	Yes
	X	32.0	1.38	1.089	35.93	34.9	1.0	Yes
	U	32.0	1.90	1.176	23.81	37.6	13.8	Yes
	W	32.0	2.82	1.276	43.76	40.9	2.9	Yes

Notes:

(a) Predicted $\Delta RT_{NDT} = CF_{best-fit} * FF$.
(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 Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal. Table A.2-2 indicates that seven of the eight surveillance data points fall within the +/- 1σ of 17°F scatter band for surveillance base metals; <u>therefore, the IS Forging 05 data is deemed "credible" per the third criterion</u>.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 28°F for weld metal. Table A.2-2 indicates that four of the four surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; <u>therefore, the weld metal (Heat # 895075) is deemed "credible" per the third criterion when only the McGuire Unit 2 data is considered.</u>

Case 4: Weld Heat # 895075 (All data)

In accordance with the NRC Case 4 guidelines, the data from McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 will now be analyzed together. Since the data are from multiple sources, the data is adjusted to the mean chemical composition and operating temperature of the surveillance capsules. This is performed in Table A.2-3 below.

Table A.2-3 Mean Chemical Composition and Operating Temperature for McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1

Material	Capsule	Cu Wt. % (a)	Ni Wt. % (a)	Plant Inlet Temperature(b) (°F)
Weld Metal Heat # 895075 (McGuire Unit 2 data)	V	0.04	0.74	555
	X	0.04	0.74	555
	U	0.04	0.74	555
	W	0.04	0.74	555
Weld Metal Heat # 895075 (Catawba Unit 1 data)	Z	0.05	0.73	562
	Y	0.05	0.73	562
	V	0.05	0.73	562
Weld Metal Heat # 895075 (Watts Bar Unit 1 data)	U	0.03	0.75	560
	W	0.03	0.75	560
	X	0.03	0.75	560
	Z	0.03	0.75	560
MEAN		**0.04**	**0.74**	**558.7**

Notes:

(a) All copper and nickel weight percent values are documented in Table 3.2-1 of this report, except for the calculated mean values.

(b) All inlet temperature values are documented in Table 4.2-2 of this report, except for the calculated mean value.

Therefore, the McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 surveillance capsule data will be adjusted to the mean chemical composition and operating temperature calculated in Table A.2-3.

McGuire Unit 2 data

CF_{Mean} = 54°F (calculated per Table 1 of Regulatory Guide 1.99, Revision 2 using Cu Wt. % = 0.04 and Ni Wt. % = 0.74 per Table A.2-3)

$CF_{Surv.\ Weld\ (McGuire\ Unit\ 2)}$ = 54°F (per Table 5.2-2 of this report)

Ratio = 54 ÷ 54 = 1 (no ratio is applied to McGuire Unit 2 surveillance data for weld Heat # 895075 in the credibility evaluation since ratio = 1)

Catawba Unit 1 data

CF_{Mean} = 54°F

$CF_{Surv.\ Weld\ (Catawba\ Unit\ 1)}$ = 68°F (per Table 5.2-2 of this report)

Ratio = 54 ÷ 68 = 0.79 (applied to Catawba Unit 1 surveillance data for weld Heat # 895075 in the credibility evaluation)

Watts Bar Unit 1 data

CF_{Mean} = 54°F

$CF_{Surv.\ Weld\ (Watts\ Bar\ Unit\ 1)}$ = 41°F (per Table 5.2-2 of this report)

Ratio = 54 ÷ 41 = 1.32 (applied to Watts Bar Unit 1 surveillance data for weld Heat # 895075 in the credibility evaluation)

The capsule-specific temperature adjustments are as shown in Table A.2-4 below:

Table A.2-4 Operating Temperature Adjustments for the McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 Surveillance Capsule Data

Material	Plant Inlet Temperature (°F)	Mean Operating Temperature (°F)	Temperature Adjustment (°F)
Weld Metal Heat # 895075 (McGuire Unit 2 data)	555	558.7	-3.7
Weld Metal Heat # 895075 (Catawba Unit 1 data)	562	558.7	+3.3
Weld Metal Heat # 895075 (Watts Bar Unit 1 data)	560	558.7	+1.3

Using the chemical composition and operating temperature adjustments described and calculated above, an interim chemistry factor is calculated for weld Heat # 895075 using the McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 data. This calculation is shown in Table A.2-5 below.

Table A.2-5 Calculation of Weld Heat # 895075 Interim Chemistry Factor for the Credibility Evaluation Using McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[(a)] ($x10^{19}$ n/cm², E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	0.302	0.672	34.81 (38.51)	23.39	0.452
	X	1.38	1.089	32.23 (35.93)	35.11	1.187
	U	1.90	1.176	20.11 (23.81)	23.64	1.382
	W	2.82	1.276	40.06 (43.76)	51.11	1.628
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	0.284	0.656	4.12 (1.91)	2.70	0.431
	Y	1.27	1.067	16.66 (17.79)	17.77	1.138
	V	2.23	1.217	23.54 (26.5)	28.66	1.482
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	0.447	0.776	1.72 (0.0[(d)])	1.33	0.602
	W	1.08	1.022	41.98 (30.5)	42.88	1.044
	X	1.71	1.148	35.77 (25.8)	41.05	1.317
	Z	2.40	1.236	20.06 (13.9)	24.80	1.528
				SUM :	292.45	12.189
	$CF_{Heat\ \#\ 895075} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (292.45) \div (12.189) =$ **24.0°F**					

Notes:

(a) f = capsule fluence taken from Tables 4.2-1 and 4.2-2 of this report.
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.
(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. Pre-adjusted values are taken from Tables 4.2-1 and 4.2-2 of this report. ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences between each surveillance weld's chemistry and the mean surveillance weld chemistry for Heat # 895075 (pre-adjusted values are listed in parentheses). The temperature adjustments are shown in Table A.2-4 of this report. The ratios applied are 0.79 for Catawba Unit 1 and 1.32 for Watts Bar Unit 1. No ratio is applied to the McGuire Unit 2 data since the ratio is equal to 1.
(d) This ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur; therefore, a value of zero is used.

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 A.2-6.

Table A.2-6 Best-Fit Evaluation for Surveillance Weld Metal Heat # 895075 Using McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 Data

Material	Capsule	CF ($Slope_{best-fit}$) (°F)	Capsule f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT}[(a)] (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<28°F (Weld)
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	24.0	0.302	0.672	34.81	16.1	18.7	Yes
	X	24.0	1.38	1.089	32.23	26.1	6.1	Yes
	U	24.0	1.90	1.176	20.11	28.2	8.1	Yes
	W	24.0	2.82	1.276	40.06	30.6	9.4	Yes
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	24.0	0.284	0.656	4.12	15.7	11.6	Yes
	Y	24.0	1.27	1.067	16.66	25.6	8.9	Yes
	V	24.0	2.23	1.217	23.54	29.2	5.7	Yes
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	24.0	0.447	0.776	1.72	18.6	16.9	Yes
	W	24.0	1.08	1.022	41.98	24.5	17.5	Yes
	X	24.0	1.71	1.148	35.77	27.5	8.2	Yes
	Z	24.0	2.40	1.236	20.06	29.7	9.6	Yes

Notes:

(a) Predicted $\Delta RT_{NDT} = CF_{best-fit} * FF$.

(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 Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 28°F for weld metal. Table A.2-6 indicates that eleven of the eleven surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material (Heat # 895075) is deemed "credible" per the third criterion when all available data is considered.

In conclusion, the combined surveillance data from McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1 for weld Heat # 895075 may be applied to the McGuire Unit 2 reactor vessel weld. The chemistry factor calculation as applicable to the McGuire Unit 2 reactor vessel weld is contained in Table 5.2-1 of this report.

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

Conclusion

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2, Section B, and the application of engineering judgment, the McGuire Unit 2 surveillance forging data is deemed credible and the surveillance weld data is deemed credible when considering McGuire Unit 2 data only and also when considering all available data (McGuire Unit 2, Catawba Unit 1, and Watts Bar Unit 1).

Appendix A.2 References

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

A.2-2 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

A.2-3 WCAP-14799, Revision 0, "Analysis of Capsule W from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," E. Terek et al., March 1997.

A.2-4 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society of Testing and Materials, 1982.

A.2-5 K. Wichman, M. Mitchell, and A. Hiser, USNRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues,* February 12, 1998.

APPENDIX B
EMERGENCY RESPONSE GUIDELINE LIMITS

The Emergency Response Guideline (ERG) limits (Reference B-1) were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values are calculated in Section 6 of this report. The material with the highest RT_{NDT} defines the limiting material, which for McGuire Unit 1 is LS Longitudinal Welds 3-442A,B,C (Position 2.1) and for McGuire Unit 2 is LS Forging 04. Table B-1 identifies ERG category limits and the limiting material RT_{NDT} values at 54 EFPY for McGuire Units 1 and 2.

Table B-1 Evaluation of McGuire Units 1 and 2 ERG Limit Category

ERG Pressure-Temperature Limits (Reference B-1)		
Applicable RT_{NDT} Value[(a)]		**ERG P-T Limit Category**
RT_{NDT} < 200°F		Category I
200°F < RT_{NDT} < 250°F		Category II
250°F < RT_{NDT} < 300°F		Category III b
Limiting RT_{NDT} Values		
Reactor Vessel Material		**RT_{NDT} Value @ 54 EFPY**
Unit 1	LS Longitudinal Welds 3-442A,B,C (Position 2.1)	203°F [(b)]
Unit 2	LS Forging 04	148°F [(c)]

Notes:
(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.
(b) Unit 1 value taken from Table 6.1-1 of this report.
(c) Unit 2 value taken from Table 6.2-1 of this report.

Unit 1

Per Table B-1, the limiting material RT_{NDT} for McGuire Unit 1 is 203°F at 54 EFPY. This value pertains to Lower Shell Longitudinal Welds 3-442A,B,C (Position 2.1). Thus, the limiting material RT_{NDT} value exceeds the ERG Category I criterion (RT_{NDT} < 200°F) prior to 54 EFPY. The transition occurs when RT_{NDT} = 200°F. The EFPY at which this transition will occur is 50.8 EFPY. Therefore, McGuire Unit 1 transitions from ERG Category I to Category II at about 50.8 EFPY.

Unit 2

Per Table B-1, the limiting material for McGuire Unit 2 (Lower Shell Forging 04) has an RT_{NDT} less than 200°F through 54 EFPY. Therefore, McGuire Unit 2 remains in ERG Category I through EOLE (54 EFPY).

Emergency Response Guideline Limits Conclusion

As summarized above, McGuire Unit 1 is currently in ERG Category I and will transition to ERG Category II at about 50.8 EFPY. McGuire Unit 2 is currently in ERG Category I and remains in ERG Category I through EOLE (54 EFPY).

To maintain consistency between the McGuire Units in terms of ERG Category, it is suggested that McGuire Units 1 and 2 both transition from Category I to Category II at 50.8 EFPY. This is consistent with the recommendations contained in Section 4.1.5 of WCAP-17174-P (Reference B-2).

Appendix B References

B-1 HF04BG, "Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 2," Westinghouse Owners Group, April 30, 2005.

B-2 WCAP-17174-P, Revision 0, "McGuire Units 1 and 2 Reactor Vessel Integrity Program Plans," B. A. Rosier and P. R. Sotherland, August 2010.

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

C.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 McGuire Unit 2 are described herein. Similarly, comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for Capsule W withdrawn from McGuire Unit 1 are also described in this section. 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" (Reference C-1). 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.

C.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the six neutron sensor sets analyzed to date as part of the McGuire Unit 2 Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

Capsule ID	Azimuthal Location	Withdrawal Time	Irradiation Time [EFPY]
V	31.5° Dual	End of Cycle 1	1.03
X	34.0° Dual	End of Cycle 5	4.16
U	34.0° Dual	End of Cycle 7	6.05
Y	31.5° Dual	End of Cycle 8	7.18
Z	34.0° Single	End of Cycle 8	7.18
W	34.0° Single	End of Cycle 10	9.44

For McGuire Unit 1, only the Capsule W dosimetry sensor sets were re-analyzed. Nevertheless, the capsule designation, location within the reactor, and time of withdrawal of all six surveillance capsules are provided here for completeness. The azimuthal locations included in the above and below tabulations represent the first octant equivalent azimuthal angle of the geometric center of the respective surveillance capsules.

Capsule ID	Azimuthal Location	Withdrawal Time	Irradiation Time [EFPY]
U	34.0° Dual	End of Cycle 1	1.09
X	34.0° Dual	End of Cycle 5	4.30
V	31.5° Dual	End of Cycle 8	7.24
Z	34.0° Single	End of Cycle 8	7.24
Y	31.5° Dual	End of Cycle 11	10.21
W	34.0° Single	End of Cycle 18	19.22

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

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

Similarly, the passive neutron sensors included in the evaluations of Surveillance Capsules U, X, V, Z, Y, and W for McGuire Unit 1 are summarized as follows:

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

Since all of the dosimetry monitors 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 passive neutron sensors are listed in Table C-1.

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 Capsules U, X, V, Z, Y, and W are documented in References C-2 through C-5, and C-14, respectively for McGuire Unit 1. Results from the radiometric counting of the neutron sensors from Capsules V, X, U, Y, Z, and W are documented in References C-8 through C-12, respectively, and summarized in Tables A-2 through A-7 of Reference C-13 for McGuire Unit 2. 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 V, X, U, Y, Z and W withdrawn from McGuire Unit 2 was based on the monthly power generation of McGuire Unit 2 from initial reactor criticality through the end of the dosimetry evaluation period. The irradiation history of the reactor over the irradiation periods experienced by Capsule W withdrawn from McGuire Unit 1 was based on the monthly power generation of McGuire 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 irradiation history applicable to Capsules V, X, U, Y, Z and W withdrawn from McGuire Unit 2 is given in Table A-1 of Reference C-13. The irradiation history applicable to Capsule W withdrawn from McGuire Unit 1 is given in Table A-2 of Reference C-14.

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 along with the computed values for C_j are listed in Tables C-2 and C-3 for McGuire Units 1 and 2, respectively. These flux values represent the cycle-dependent results 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, 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 McGuire Unit 2 fission sensor reaction rates are summarized as follows:

Correction	Capsule V	Capsule X	Capsule U	Capsule Y	Capsule Z	Capsule W
^{235}U Impurity/Pu Build-in	0.8724	0.8300	0.8124	0.8114	0.8023	0.7824
^{238}U(γ,f)	0.9640	0.9634	0.9635	0.9644	0.9664	0.9664
Net ^{238}U Correction	0.8410	0.7996	0.7827	0.7825	0.7753	0.7561
^{237}Np(γ,f)	0.9896	0.9897	0.9897	0.9897	0.9907	0.9907

Similarly, the correction factors applied to the McGuire Unit 1 fission sensor reaction rates are summarized below:

Correction	Capsule W
^{235}U Impurity/Pu Build-in	0.7096
^{238}U(γ,f)	0.9668
Net ^{238}U Correction	0.6860
^{237}Np(γ,f)	0.9907

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 Capsules V, X, U, Y, Z, and W from McGuire Unit 2 and Capsule W from McGuire Unit 1 are given in Table C-4. In Table C-4, 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.

C.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 McGuire Unit 1 Capsule W dosimetry and McGuire Unit 2 all surveillance capsule dosimetry, the FERRET code (Reference C-6) 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 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 McGuire Units 1 and 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 C.1.1. The dosimetry reaction cross sections and uncertainties were obtained from the SNLRML dosimetry cross-section library (Reference C-7). 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 E 706 (IIB)."

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 E 944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The following provides a summary of the uncertainties associated with the least-squares evaluation of the McGuire Units 1 and 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%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	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 McGuire Units 1 and 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%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{238}U(n,f)^{137}Cs$	0.54-0.64%
$^{237}Np(n,f)^{137}Cs$	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 McGuire Units 1 and 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

C.1.3 Comparisons of Measurements and Calculations

Results of the least-squares evaluations of the dosimetry from the McGuire Unit 2 surveillance capsules withdrawn to date and from Capsule W withdrawn from McGuire Unit 1 are provided in Tables C-5, C-6, and C-9. In Table C-5, measured, calculated, and best-estimate values for sensor reaction rates are given for each capsule. Also provided in this tabulation are ratios of the measured reaction rates to both the calculated and least-squares adjusted reaction rates. These ratios of M/C and M/BE illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. In Tables C-6 and C-9, comparison of the calculated and best-estimate values of neutron flux (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the BE/C ratios observed for each of the capsules.

The data comparisons provided in Tables C-5, C-6, and C-9 show that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross sections. Further, these results indicate that the use of the least-squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 2 of this report, it may be noted that the uncertainty associated with the unadjusted calculation of neutron fluence (E > 1.0 MeV) and iron atom displacements at the surveillance capsule locations is specified as 12% at the 1σ level. From Tables C-6 and C-9, it is noted that the corresponding uncertainties associated with the least-squares adjusted exposure parameters have

been reduced to 6-8% for neutron flux (E > 1.0 MeV) and 8-10% for iron atom displacement rate. Again, the uncertainties from the least-squares evaluation are at the 1σ level.

Further comparisons of the measurement results (from Tables C-5, C-6, and C-9) with calculations are given in Tables C-7 and C-8 for McGuire Unit 2 and Tables C-10 and C-11 for McGuire Unit 1. These comparisons are given on two levels. In Table C-7 and Table C-10, calculations of individual threshold sensor reaction rates are compared directly with the corresponding measurements. These threshold reaction rate comparisons provide a good evaluation of the accuracy of the fast neutron portion of the calculated energy spectra. In Table C-8 and Table C-11, calculations of fast neutron exposure rates in terms of ϕ(E > 1.0 MeV) and dpa/s are compared with the best-estimate results obtained from the least-squares evaluation of the capsule dosimetry results. These two levels of comparison yield consistent and similar results with all measurement-to-calculation comparisons falling well within the 20% limits specified as the acceptance criteria in Regulatory Guide 1.190.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.87 to 1.21 for the 30 samples included in the McGuire Unit 2 data set. The overall average M/C ratio for the entire set of McGuire Unit 2 data is 1.03 with an associated standard deviation of 9.2%. Similarly for McGuire Unit 1, the M/C comparisons for fast neutron reactions range from 0.82 to 1.22 for the 27 samples included in the data set. The overall average M/C ratio for the entire set of McGuire Unit 1 data is 1.03 with an associated standard deviation of 10.5%.

In the comparisons of best estimate and calculated fast neutron exposure parameters for McGuire Unit 2, the corresponding BE/C comparisons for the capsule data sets range from 0.96 to 1.02 for neutron flux (E > 1.0 MeV) and from 0.98 to 1.04 for iron atom displacement rate. The overall average BE/C ratios for neutron flux (E > 1.0 MeV) and iron atom displacement rate are 0.99 with a standard deviation of 2.1% and 1.01 with a standard deviation of 2.1%, respectively. Similarly, for McGuire Unit 1, the corresponding BE/C comparisons for the capsule data sets range from 0.92 to 1.11 for neutron flux (E > 1.0 MeV) and from 0.93 to 1.12 for iron atom displacement rate. The overall average BE/C ratios for neutron flux (E > 1.0 MeV) and iron atom displacement rate are 0.99 with a standard deviation of 7.0% and 0.99 with a standard deviation of 6.9%, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 2 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region and extended beltline region of the McGuire Units 1 and 2 reactor pressure vessels.

Table C-1 Nuclear Parameters Used in the Evaluation of Neutron Sensors

Monitor Material	Reaction of Interest	Target Atom Fraction	90% Response Range (MeV)	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	0.6917	4.9 – 11.9	5.271 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	5.271 y	

Note:

The 90% response range is defined such that, in the neutron spectrum characteristic of the McGuire Units 1 and 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 C-2 Calculated C_j Factors at McGuire Unit 2 Surveillance Capsule Center Core Midplane Elevation

Fuel Cycle	Cycle Length [EFPS]	φ(E > 1.0 MeV) [n/cm²-s]					
		Capsule V	**Capsule X**	**Capsule U**	**Capsule Y**	**Capsule Z**	**Capsule W**
1	3.25E+07	9.32E+10	1.09E+11	1.09E+11	9.32E+10	1.08E+11	1.08E+11
2	2.15E+07		1.29E+11	1.29E+11	1.11E+11	1.28E+11	1.28E+11
3	2.30E+07		1.09E+11	1.09E+11	9.30E+10	1.09E+11	1.09E+11
4	2.68E+07		9.44E+10	9.44E+10	8.23E+10	9.40E+10	9.40E+10
5	2.75E+07		8.94E+10	8.94E+10	7.88E+10	8.89E+10	8.89E+10
6	2.90E+07			8.61E+10	7.65E+10	8.57E+10	8.57E+10
7	3.06E+07			8.93E+10	7.89E+10	8.88E+10	8.88E+10
8	3.57E+07				7.89E+10	8.93E+10	8.93E+10
9	3.44E+07						8.97E+10
10	3.69E+07						8.00E+10
Average		9.32E+10	1.05E+11	9.97E+10	8.55E+10	9.77E+10	9.46E+10

Table C-2 (cont.) Calculated C_j Factors at McGuire Unit 2 Surveillance Capsule Center Core Midplane Elevation

Fuel Cycle	Cycle Length [EFPS]	C_j					
		Capsule V	Capsule X	Capsule U	Capsule Y	Capsule Z	Capsule W
1	3.25E+07	1.00	1.04	1.09	1.09	1.11	1.14
2	2.15E+07		1.23	1.29	1.30	1.31	1.35
3	2.30E+07		1.04	1.09	1.09	1.12	1.15
4	2.68E+07		0.90	0.95	0.96	0.96	0.99
5	2.75E+07		0.85	0.90	0.92	0.91	0.94
6	2.90E+07			0.86	0.89	0.88	0.91
7	3.06E+07			0.90	0.92	0.91	0.94
8	3.57E+07				0.92	0.91	0.94
9	3.44E+07						0.95
10	3.69E+07						0.85
Average		1.00	1.00	1.00	1.00	1.00	1.00

Table C-3 Calculated C_j Factors at McGuire Unit 1 Surveillance Capsule Center Core Midplane Elevation

Fuel Cycle	Cycle Length [EFPS]	φ(E > 1.0 MeV) [n/cm^2-s]
		Capsule W
1	3.44E+07	1.11E+11
2	2.30E+07	1.31E+11
3	2.49E+07	9.70E+10
4	2.62E+07	9.24E+10
5	2.71E+07	8.36E+10
6	2.59E+07	9.01E+10
7	3.50E+07	8.82E+10
8	3.19E+07	8.72E+10
9	2.87E+07	9.12E+10
10	3.38E+07	8.82E+10
11	3.12E+07	8.18E+10
12	3.12E+07	7.98E+10
13	3.76E+07	6.69E+10
14	4.13E+07	6.21E+10
15	4.39E+07	7.91E+10
16	4.42E+07	7.42E+10
17	4.26E+07	7.66E+10
18	4.35E+07	6.59E+10
Average		8.37E+10

Table C-3 (cont.) Calculated C_j Factors at McGuire Unit 1 Surveillance Capsule Center Core Midplane Elevation

Fuel Cycle	Cycle Length [EFPS]	C_j
		Capsule W
1	3.44E+07	1.33
2	2.30E+07	1.57
3	2.49E+07	1.16
4	2.62E+07	1.10
5	2.71E+07	1.00
6	2.59E+07	1.08
7	3.50E+07	1.05
8	3.19E+07	1.04
9	2.87E+07	1.09
10	3.38E+07	1.05
11	3.12E+07	0.98
12	3.12E+07	0.95
13	3.76E+07	0.80
14	4.13E+07	0.74
15	4.39E+07	0.95
16	4.42E+07	0.89
17	4.26E+07	0.92
18	4.35E+07	0.79
Average		1.00

Table C-4 Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule V

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	3.84E+04	3.38E+05	5.16E-17
	Middle	4.03E+04	3.55E+05	5.42E-17
	Bottom	3.76E+04	3.31E+05	5.05E-17
	Average			5.21E-17
^{54}Fe (n,p) ^{54}Mn	Top	9.69E+05	3.23E+06	5.12E-15
	Middle	1.01E+06	3.37E+06	5.34E-15
	Bottom	9.87E+05	3.29E+06	5.22E-15
	Average			5.23E-15
^{58}Ni (n,p) ^{58}Co	Top	3.96E+06	5.05E+07	7.23E-15
	Middle	3.58E+06	4.56E+07	6.53E-15
	Bottom	4.04E+06	5.15E+07	7.37E-15
	Average			7.04E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.36E+05	5.93E+06	3.89E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.27E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	1.20E+06	5.23E+07	3.34E-13
	Including γ fission corrections:			3.30E-13
^{59}Co (n,γ) ^{60}Co	Top	1.01E+07	8.90E+07	5.81E-12
	Middle	8.83E+06	7.78E+07	5.08E-12
	Middle	7.83E+06	6.90E+07	4.50E-12
	Bottom	9.49E+06	8.36E+07	5.45E-12
	Bottom	8.75E+06	7.71E+07	5.03E-12
	Bottom	8.34E+06	7.35E+07	4.79E-12
	Average			5.11E-12
^{59}Co (n, γ) ^{60}Co (Cd)	Top	5.04E+06	4.44E+07	2.90E-12
	Middle	4.72E+06	4.16E+07	2.71E-12
	Bottom	4.74E+06	4.18E+07	2.72E-12
	Average			2.78E-12

Notes:

1. Measured specific activities are indexed to a counting date of September 16, 1985.

2. The average ^{238}U (n,f) reaction rate of 3.27E-14 includes a correction factor of 0.8724 to account for plutonium build-in and an additional factor of 0.9640 to account for photo-fission effects in the sensor.

3. The average ^{237}Np (n,f) reaction rate of 3.30E-13 includes a correction factor of 0.9896 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule X

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	Top	1.23E+05	3.43E+05	5.24E-17
	Middle	1.32E+05	3.69E+05	5.62E-17
	Bottom	1.28E+05	3.57E+05	5.45E-17
	Average			5.44E-17
$^{54}Fe\ (n,p)\ ^{54}Mn$	Top	1.58E+06	3.33E+06	5.29E-15
	Middle	1.71E+06	3.61E+06	5.72E-15
	Bottom	1.67E+06	3.52E+06	5.59E-15
	Average			5.53E-15
$^{58}Ni\ (n,p)\ ^{58}Co$	Top	6.20E+06	5.28E+07	7.56E-15
	Middle	6.55E+06	5.58E+07	7.98E-15
	Bottom	6.44E+06	5.48E+07	7.85E-15
	Average			7.80E-15
$^{238}U\ (n,f)\ ^{137}Cs$ (Cd)	Middle	6.14E+05	6.91E+06	4.54E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.63E-14
$^{237}Np\ (n,f)\ ^{137}Cs$ (Cd)	Middle	4.60E+06	5.18E+07	3.30E-13
	Including γ fission corrections:			3.27E-13
$^{59}Co\ (n,\gamma)\ ^{60}Co$	Top	2.86E+07	7.98E+07	5.21E-12
	Top	3.33E+07	9.30E+07	6.07E-12
	Middle	2.65E+07	7.40E+07	4.83E-12
	Middle	3.19E+07	8.91E+07	5.81E-12
	Bottom	2.77E+07	7.73E+07	5.05E-12
	Bottom	3.20E+07	8.93E+07	5.83E-12
	Average			5.46E-12
$^{59}Co\ (n,\gamma)\ ^{60}Co$ (Cd)	Top	1.78E+07	4.97E+07	3.24E-12
	Middle	1.67E+07	4.66E+07	3.04E-12
	Bottom	1.69E+07	4.72E+07	3.08E-12
	Average			3.12E-12

Notes:

1. Measured specific activities are indexed to a counting date of January 12, 1990.
2. The average ^{238}U (n,f) reaction rate of 3.63E-14 includes a correction factor of 0.8300 to account for plutonium build-in and an additional factor of 0.9634 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 3.27E-13 includes a correction factor of 0.9897 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule U

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.54E+05	3.36E+05	5.13E-17
	Middle	1.63E+05	3.56E+05	5.43E-17
	Bottom	1.56E+05	3.41E+05	5.20E-17
	Average			5.25E-17
^{54}Fe (n,p) ^{54}Mn	Top	1.75E+06	3.20E+06	5.08E-15
	Middle	1.86E+06	3.40E+06	5.40E-15
	Bottom	1.70E+06	3.11E+06	4.93E-15
	Average			5.14E-15
^{58}Ni (n,p) ^{58}Co	Top	1.16E+07	4.93E+07	7.05E-15
	Middle	1.19E+07	5.05E+07	7.24E-15
	Bottom	1.18E+07	5.01E+07	7.18E-15
	Average			7.16E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	8.57E+05	6.82E+06	4.48E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.50E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	6.48E+06	5.15E+07	3.29E-13
	Including γ fission corrections:			3.25E-13
^{59}Co (n,γ) ^{60}Co	Top	3.38E+07	7.38E+07	4.81E-12
	Top	3.75E+07	8.19E+07	5.34E-12
	Middle	3.38E+07	7.38E+07	4.81E-12
	Middle	3.15E+07	6.88E+07	4.49E-12
	Bottom	3.04E+07	6.64E+07	4.33E-12
	Bottom	3.57E+07	7.79E+07	5.09E-12
	Average			4.81E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Top	2.11E+07	4.61E+07	3.01E-12
	Middle	1.95E+07	4.26E+07	2.78E-12
	Bottom	2.05E+07	4.48E+07	2.92E-12
	Average			2.90E-12

Notes:

1. Measured specific activities are indexed to a counting date of May 15, 1992.
2. The average ^{238}U (n,f) reaction rate of 3.50E-14 includes a correction factor of 0.8124 to account for plutonium build-in and an additional factor of 0.9635 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 3.25E-13 includes a correction factor of 0.9897 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule Y

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.45E+05	3.07E+05	4.68E-17
	Middle	1.54E+05	3.26E+05	4.97E-17
	Bottom	1.48E+05	3.13E+05	4.78E-17
	Average			4.81E-17
^{54}Fe (n,p) ^{54}Mn	Top	1.00E+06	2.62E+06	4.16E-15
	Middle	1.07E+06	2.80E+06	4.45E-15
	Bottom	1.02E+06	2.67E+06	4.24E-15
	Average			4.28E-15
^{58}Ni (n,p) ^{58}Co	Top	1.75E+06	4.33E+07	6.20E-15
	Middle	1.83E+06	4.53E+07	6.49E-15
	Bottom	1.76E+06	4.36E+07	6.24E-15
	Average			6.31E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	8.52E+05	5.88E+06	3.86E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.02E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	6.41E+06	4.42E+07	2.82E-13
	Including γ fission corrections:			2.79E-13
^{59}Co (n,γ) ^{60}Co	Top	3.25E+07	6.88E+07	4.49E-12
	Top	2.77E+07	5.86E+07	3.83E-12
	Middle	2.46E+07	5.21E+07	3.40E-12
	Middle	2.96E+07	6.27E+07	4.09E-12
	Bottom	2.64E+07	5.59E+07	3.65E-12
	Bottom	N/A	N/A	N/A
	Average			3.89E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Top	1.68E+07	3.56E+07	2.32E-12
	Middle	1.58E+07	3.34E+07	2.18E-12
	Bottom	1.60E+07	3.39E+07	2.21E-12
	Average			2.24E-12

Notes:

1. Measured specific activities are indexed to a counting date of May 6, 1994.
2. The average ^{238}U (n,f) reaction rate of 3.02E-14 includes a correction factor of 0.8114 to account for plutonium build-in and an additional factor of 0.9644 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 2.79E-13 includes a correction factor of 0.9897 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule Z

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.53E+05	3.25E+05	4.96E-17
	Middle	1.55E+05	3.30E+05	5.03E-17
	Bottom	1.55E+05	3.30E+05	5.03E-17
	Average			5.01E-17
^{54}Fe (n,p) ^{54}Mn	Top	1.12E+06	2.97E+06	4.71E-15
	Middle	1.19E+06	3.15E+06	5.00E-15
	Bottom	1.14E+06	3.02E+06	4.79E-15
	Average			4.83E-15
^{58}Ni (n,p) ^{58}Co	Top	1.92E+06	4.80E+07	6.87E-15
	Middle	1.97E+06	4.92E+07	7.05E-15
	Bottom	1.90E+06	4.75E+07	6.80E-15
	Average			6.91E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.04E+06	7.18E+06	4.71E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.66E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	7.53E+06	5.20E+07	3.32E-13
	Including γ fission corrections:			3.28E-13
^{59}Co (n,γ) ^{60}Co	Top	3.30E+07	7.02E+07	4.58E-12
	Top	3.79E+07	8.06E+07	5.26E-12
	Middle	3.64E+07	7.74E+07	5.05E-12
	Middle	3.01E+07	6.40E+07	4.18E-12
	Bottom	3.17E+07	6.74E+07	4.40E-12
	Bottom	N/A	N/A	N/A
	Average			4.69E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Top	1.99E+07	4.23E+07	2.76E-12
	Middle	1.89E+07	4.02E+07	2.62E-12
	Bottom	1.94E+07	4.12E+07	2.69E-12
	Average			2.69E-12

Notes:

1. Measured specific activities are indexed to a counting date of May 6, 1994.
2. The average ^{238}U (n,f) reaction rate of 3.66E-14 includes a correction factor of 0.8023 to account for plutonium build-in and an additional factor of 0.9664 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 3.28E-13 includes a correction factor of 0.9907 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 2 Surveillance Capsule W

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.84E+05	3.24E+05	4.95E-17
	Middle	1.89E+05	3.33E+05	5.08E-17
	Bottom	1.84E+05	3.24E+05	4.95E-17
	Average			4.99E-17
^{54}Fe (n,p) ^{54}Mn	Top	1.59E+06	3.03E+06	4.81E-15
	Middle	1.65E+06	3.14E+06	4.99E-15
	Bottom	1.60E+06	3.05E+06	4.84E-15
	Average			4.88E-15
^{58}Ni (n,p) ^{58}Co	Top	7.15E+06	4.93E+07	7.05E-15
	Middle	7.36E+06	5.07E+07	7.26E-15
	Bottom	7.20E+06	4.96E+07	7.10E-15
	Average			7.14E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.29E+06	6.92E+06	4.54E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			3.43E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	9.25E+06	4.96E+07	3.16E-13
	Including γ fission corrections:			3.13E-13
^{59}Co (n,γ) ^{60}Co	Top	4.47E+07	7.88E+07	5.14E-12
	Top	4.04E+07	7.12E+07	4.65E-12
	Middle	4.16E+07	7.34E+07	4.79E-12
	Middle	3.51E+07	6.19E+07	4.04E-12
	Bottom	3.89E+07	6.86E+07	4.48E-12
	Bottom	4.45E+07	7.85E+07	5.12E-12
	Average			4.70E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Top	2.34E+07	4.13E+07	2.69E-12
	Middle	2.23E+07	3.93E+07	2.57E-12
	Bottom	2.35E+07	4.14E+07	2.70E-12
	Average			2.65E-12

Notes:

1. Measured specific activities are indexed to a counting date of September 26, 1996.
2. The average ^{238}U (n,f) reaction rate of 3.43E-14 includes a correction factor of 0.7824 to account for plutonium build-in and an additional factor of 0.9664 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 3.13E-13 includes a correction factor of 0.9907 to account for photo-fission effects in the sensor.

Table C-4 (cont.) Measured Sensor Activities and Reaction Rates for McGuire Unit 1 Surveillance Capsule W

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.86E+05	2.88E+05	4.39E-17
	Middle	1.95E+05	3.02E+05	4.61E-17
	Bottom	1.88E+05	2.91E+05	4.44E-17
	Average			4.48E-17
^{54}Fe (n,p) ^{54}Mn	Top	8.87E+05	2.89E+06	4.59E-15
	Middle	9.32E+05	3.04E+06	4.82E-15
	Bottom	8.97E+05	2.93E+06	4.64E-15
	Average			4.68E-15
^{58}Ni (n,p) ^{58}Co	Top	6.09E+05	4.86E+07	6.96E-15
	Middle	6.38E+05	5.10E+07	7.29E-15
	Bottom	6.29E+05	5.02E+07	7.19E-15
	Average			7.15E-15
^{238}U (n,f) ^{137}Cs (Cd)	Middle	2.05E+06	6.24E+06	4.10E-14
	Including ^{235}U, ^{239}Pu, and γ fission corrections:			2.81E-14
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	1.26E+07	3.83E+07	2.45E-13
	Including γ fission corrections:			2.42E-13
^{59}Co (n,γ) ^{60}Co	Top	3.87E+07	5.99E+07	3.91E-12
	Top	3.82E+07	5.92E+07	3.86E-12
	Middle	2.96E+07	4.58E+07	2.99E-12
	Middle	3.30E+07	5.11E+07	3.33E-12
	Bottom	3.22E+07	4.99E+07	3.25E-12
	Bottom	3.95E+07	6.12E+07	3.99E-12
	Average			3.56E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Top	2.07E+07	3.21E+07	2.09E-12
	Middle	2.15E+07	3.33E+07	2.17E-12
	Bottom	2.18E+07	3.38E+07	2.20E-12
	Average			2.16E-12

Notes:

1. Measured specific activities are indexed to a counting date of May 5, 2008.
2. The average ^{238}U (n,f) reaction rate of 2.81E-14 includes a correction factor of 0.7096 to account for plutonium build-in and an additional factor of 0.9668 to account for photo-fission effects in the sensor.
3. The average ^{237}Np (n,f) reaction rate of 2.42E-13 includes a correction factor of 0.9907 to account for photo-fission effects in the sensor.

Table C-5 Comparison of Measured, Calculated, and Best-Estimate Reaction Rates at the Surveillance Capsule Center

McGuire Unit 2 Capsule V					
	Reaction Rate [rps/atom]				
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	5.21E-17	4.66E-17	4.99E-17	1.12	1.04
$^{54}Fe(n,p)^{54}Mn$	5.23E-15	5.32E-15	5.35E-15	0.98	0.98
$^{58}Ni(n,p)^{58}Co$	7.04E-15	7.50E-15	7.43E-15	0.94	0.94
$^{59}Co(n,\gamma)^{60}Co$	5.11E-12	4.71E-12	5.07E-12	1.08	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.78E-12	3.03E-12	2.80E-12	0.92	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.27E-14	2.93E-14	2.97E-14	1.12	1.10
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.30E-13	2.92E-13	3.15E-13	1.13	1.05

McGuire Unit 2 Capsule X					
	Reaction Rate [rps/atom]				
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	5.44E-17	4.93E-17	5.23E-17	1.10	1.04
$^{54}Fe(n,p)^{54}Mn$	5.53E-15	5.78E-15	5.72E-15	0.96	0.96
$^{58}Ni(n,p)^{58}Co$	7.80E-15	8.19E-15	8.06E-15	0.95	0.97
$^{59}Co(n,\gamma)^{60}Co$	5.46E-12	5.43E-12	5.44E-12	1.01	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	3.12E-12	3.51E-12	3.15E-12	0.89	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.63E-14	3.26E-14	3.23E-14	1.11	1.12
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.27E-13	3.35E-13	3.29E-13	0.98	0.99

McGuire Unit 2 Capsule U					
	Reaction Rate [rps/atom]				
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	5.25E-17	4.72E-17	4.99E-17	1.11	1.05
$^{54}Fe(n,p)^{54}Mn$	5.14E-15	5.51E-15	5.36E-15	0.93	0.96
$^{58}Ni(n,p)^{58}Co$	7.15E-15	7.79E-15	7.52E-15	0.92	0.95
$^{59}Co(n,\gamma)^{60}Co$	4.81E-12	5.14E-12	4.81E-12	0.94	1.00
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.90E-12	3.31E-12	2.92E-12	0.88	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.50E-14	3.10E-14	3.03E-14	1.13	1.15
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.25E-13	3.17E-13	3.19E-13	1.03	1.02

Note:

See Section C.1.2 for details describing the Best-Estimate (BE) reaction rates.

Table C-5 (cont.) Comparison of Measured, Calculated, and Best-Estimate Reaction Rates at the Surveillance Capsule Center

McGuire Unit 2 Capsule Y

Reaction	Reaction Rate [rps/atom]			M/C	M/BE
	Measured	Calculated	Best Estimate		
$^{63}Cu(n,\alpha)^{60}Co$	4.81E-17	4.38E-17	4.53E-17	1.10	1.06
$^{54}Fe(n,p)^{54}Mn$	4.28E-15	4.93E-15	4.62E-15	0.87	0.93
$^{58}Ni(n,p)^{58}Co$	6.31E-15	6.94E-15	6.54E-15	0.91	0.96
$^{59}Co(n,\gamma)^{60}Co$	3.89E-12	4.28E-12	3.88E-12	0.91	1.00
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.24E-12	2.76E-12	2.26E-12	0.81	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.02E-14	2.70E-14	2.58E-14	1.12	1.18
$^{237}Np(n,f)^{137}Cs$ (Cd)	2.79E-13	2.67E-13	2.69E-13	1.04	1.04

McGuire Unit 2 Capsule Z

Reaction	Reaction Rate [rps/atom]			M/C	M/BE
	Measured	Calculated	Best Estimate		
$^{63}Cu(n,\alpha)^{60}Co$	5.00E-17	4.59E-17	4.75E-17	1.09	1.05
$^{54}Fe(n,p)^{54}Mn$	4.83E-15	5.35E-15	5.14E-15	0.90	0.94
$^{58}Ni(n,p)^{58}Co$	6.91E-15	7.58E-15	7.27E-15	0.91	0.95
$^{59}Co(n,\gamma)^{60}Co$	4.69E-12	4.62E-12	4.66E-12	1.02	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.69E-12	3.03E-12	2.72E-12	0.89	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.65E-14	3.02E-14	2.98E-14	1.21	1.22
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.28E-13	3.13E-13	3.22E-13	1.05	1.02

McGuire Unit 2 Capsule W

Reaction	Reaction Rate [rps/atom]			M/C	M/BE
	Measured	Calculated	Best Estimate		
$^{63}Cu(n,\alpha)^{60}Co$	4.99E-17	4.46E-17	4.77E-17	1.12	1.04
$^{54}Fe(n,p)^{54}Mn$	4.88E-15	5.19E-15	5.16E-15	0.94	0.94
$^{58}Ni(n,p)^{58}Co$	7.14E-15	7.35E-15	7.34E-15	0.97	0.97
$^{59}Co(n,\gamma)^{60}Co$	4.70E-12	4.47E-12	4.67E-12	1.05	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.65E-12	2.93E-12	2.68E-12	0.91	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	3.43E-14	2.92E-14	2.95E-14	1.17	1.16
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.13E-13	3.02E-13	3.11E-13	1.04	1.01

Note:

See Section C.1.2 for details describing the Best-Estimate (BE) reaction rates.

Table C-5 (cont.) Comparison of Measured, Calculated, and Best-Estimate Reaction Rates at the Surveillance Capsule Center

McGuire Unit 1 Capsule W					
	Reaction Rate [rps/atom]				
Reaction	**Measured**	**Calculated**	**Best Estimate**	**M/C**	**M/BE**
^{63}Cu(n,α)^{60}Co	4.48E-17	4.03E-17	4.40E-17	1.11	1.02
^{54}Fe(n,p)^{54}Mn	4.68E-15	4.64E-15	4.86E-15	1.01	0.96
^{58}Ni(n,p)^{58}Co	7.15E-15	6.56E-15	6.98E-15	1.09	1.02
^{59}Co(n,γ)^{60}Co	3.56E-12	3.91E-12	3.55E-12	0.91	1.00
^{59}Co(n,γ)^{60}Co (Cd)	2.16E-12	2.57E-12	2.17E-12	0.84	0.99
^{238}U(n,f)^{137}Cs (Cd)	2.81E-14	2.59E-14	2.68E-14	1.08	1.05
^{237}Np(n,f)^{137}Cs (Cd)	2.42E-13	2.67E-13	2.55E-13	0.91	0.95

Note:

See Section C.1.2 for details describing the Best-Estimate (BE) reaction rates.

Table C-6 Comparison of Calculated and Best-Estimate Exposure Rates at the Surveillance Capsule Center from McGuire Unit 2

Capsule ID	φ(E > 1.0 MeV) [n/cm^2-s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
V	9.37E+10	9.52E+10	6%	1.02
X	1.06E+11	1.04E+11	6%	0.98
U	1.00E+11	9.81E+10	6%	0.98
Y	8.59E+10	8.25E+10	6%	0.96
Z	9.82E+10	9.78E+10	6%	1.00
W	9.51E+10	9.62E+10	6%	1.01

Note:
Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period and are the average neutron exposure over the irradiation period for each capsule. See Section C.1.2 for details describing the Best-Estimate (BE) exposure rates.

Capsule ID	Iron Atom Displacement Rate [dpa/s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
V	1.82E-10	1.89E-10	8%	1.04
X	2.09E-10	2.07E-10	8%	0.99
U	1.98E-10	1.97E-10	8%	0.99
Y	1.67E-10	1.64E-10	8%	0.98
Z	1.95E-10	1.98E-10	8%	1.02
W	1.88E-10	1.93E-10	8%	1.03

Note:
Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period and are the average neutron exposure over the irradiation period for each capsule. See Section C.1.2 for details describing the Best-Estimate (BE) exposure rates.

Table C-7 Comparison of Measured/Calculated (M/C) Sensor Reaction Rate Ratios Including all Fast Neutron Threshold Reactions from McGuire Unit 2

Reaction	M/C Ratio					
	Capsule V	Capsule X	Capsule U	Capsule Y	Capsule Z	Capsule W
^{63}Cu(n, α)^{60}Co	1.12	1.10	1.11	1.10	1.09	1.12
^{54}Fe(n,p)^{54}Mn	0.98	0.96	0.93	0.87	0.90	0.94
^{58}Ni(n,p)^{58}Co	0.94	0.95	0.92	0.91	0.91	0.97
^{238}U(n,f)^{137}Cs (Cd)	1.12	1.11	1.13	1.12	1.21	1.17
^{237}Np(n,f)^{137}Cs (Cd)	1.13	0.98	1.03	1.04	1.05	1.04
Average	1.06	1.02	1.02	1.01	1.03	1.05
% Standard Deviation	8.6%	7.7%	9.6%	11.2%	12.6%	9.3%

Note:

The overall average M/C ratio for the set of 30 sensor measurements is 1.03 with an associated standard deviation of 9.2%.

Table C-8 Comparison of Best-Estimate/Calculated (BE/C) Exposure Rate Ratios from McGuire Unit 2

Capsule ID	BE/C Ratio	
	ϕ(E > 1.0 MeV)	dpa/s
V	1.02	1.04
X	0.98	0.99
U	0.98	0.99
Y	0.96	0.98
Z	1.00	1.02
W	1.01	1.03
Average	0.99	1.01
% Standard Deviation	2.1%	2.1%

Table C-9 Comparison of Calculated and Best-Estimate Exposure Rates at the Surveillance Capsule Center from McGuire Unit 1

Capsule ID	φ(E > 1.0 MeV) [n/cm^2-s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
U	1.11E+11	1.23E+11	6%	1.11
X	1.04E+11	9.75E+10	6%	0.94
V	8.49E+10	8.21E+10	7%	0.97
Z	9.73E+10	9.30E+10	6%	0.96
Y	8.24E+10	7.61E+10	8%	0.92
W	8.41E+10	8.56E+10	6%	1.02

Note:
Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period and are the average neutron exposure over the irradiation period for each capsule. See Section C.1.2 for details describing the Best Estimate (BE) exposure rates.

For Capsules U, X, V, Z, and Y, the calculated, best estimate values, uncertainties of best estimate, and BE/C ratios were taken from Reference C-14.

Capsule ID	Iron Atom Displacement Rate [dpa/s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
U	2.20E-10	2.47E-10	8%	1.12
X	2.05E-10	1.93E-10	8%	0.94
V	1.65E-10	1.62E-10	9%	0.98
Z	1.93E-10	1.87E-10	8%	0.97
Y	1.60E-10	1.49E-10	10%	0.93
W	1.66E-10	1.66E-10	8%	1.00

Note:
Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period and are the average neutron exposure over the irradiation period for each capsule. See Section C.1.2 for details describing the Best-Estimate (BE) exposure rates.

For Capsules U, X, V, Z, and Y, the calculated, best-estimate values, uncertainties of best estimate, and BE/C ratios were taken from Reference C-14.

Table C-10 Comparison of Measured/Calculated (M/C) Sensor Reaction Rate Ratios Including all Fast Neutron Threshold Reactions from McGuire Unit 1

Reaction	M/C Ratio					
	Capsule V	Capsule X	Capsule U	Capsule Y	Capsule Z	Capsule W
$^{63}Cu(n, \alpha)^{60}Co$	1.17	1.08	1.13	1.12	1.01	1.11
$^{54}Fe(n,p)^{54}Mn$	1.05	0.92	0.93	0.91	0.93	1.01
$^{58}Ni(n,p)^{58}Co$	1.06	0.93	0.93	0.93	0.92	1.09
$^{238}U(n,f)^{137}Cs$ (Cd)	1.20	1.12	1.11	1.12	N/A	1.08
$^{237}Np(n,f)^{137}Cs$ (Cd)	1.22	0.82	N/A	0.92	N/A	0.91
Average	1.14	0.97	1.03	1.00	0.95	1.04
% Standard Deviation	7.0%	12.7%	10.7%	11.0%	5.2%	7.9%

Note:

The overall average M/C ratio for the set of 27 sensor measurements is 1.03 with an associated standard deviation of 10.5%.

For Capsules U, X, V, Z, and Y, the M/C ratios were taken from Reference C-14.

Table C-11 Comparison of Best-Estimate/Calculated (BE/C) Exposure Rate Ratios from McGuire Unit 1

Capsule ID	BE/C Ratio	
	ϕ(E > 1.0 MeV)	dpa/s
U	1.11	1.12
X	0.94	0.94
V	0.97	0.98
Z	0.96	0.97
Y	0.92	0.93
W	1.02	1.00
Average	0.99	0.99
% Standard Deviation	7.0%	6.9%

Note:

For Capsules U, X, V, Z, and Y, the BE/C ratios were taken from Reference C-14.

C.2 REFERENCES

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