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WCAP-14280
Rev. 1

Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel

Westinghouse Energy Systems

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ATTACHMENT 5

Letter from Mark L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

November 18, 1998

Proposed Amendment 157

WCAP-14280 Revision 1
Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel

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Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel

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PREFACE

This report has been technically reviewed and verified by:

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FORWARD

This report along with four other companion documents have been prepared by Westinghouse Electric Corporation and ATI Consulting to assess and document the integrity of the Kewaunee Nuclear Power Plant (KNPP)reactor vessel relative to the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, (which encompass pressurized thermal shock (PTS) and upper shelf energy (USE) evaluations), and any potential impact on low temperature overpressure (LTOP) limits or pressure-temperature limits. These reports: (1) summarize the KNPP weld metal (1P3571) surveillance capsule test results performed to date (WCAP-15074); (2) document supplemental surveillance capsule fracture toughness testing results for the KNPP weld metal both in the unirradiated and irradiated condition (WCAP-14279, Rev. 1); (3) introduce and apply a new methodology, based on the Master Curve Approach, for assessing the integrity of the KNPP reactor vessel (WCAP-15075); (4) include various PTS evaluations for KNPP conducted in accordance with the methodology given in 10 CFR 50.61 and the Master Curve Approach (WCAP-14280, Rev. 1); and (5) present heatup and cooldown curves corresponding to end of plant life fluence (WCAP-14278, Rev. 1). The heatup and cooldown limit curves presented in WCAP-14278, Rev. 1 are derived using ASME Code Case N-588. These five documents support a new proposed amendment to modify the KNPP Technical Specification limits for heatup, cooldown, and low temperature overpressure protection. The current Technical Specification heatup and cooldown limit curves will expire at 20 EFPY which is scheduled to occur in spring of 1999. The engineering evaluations incorporate all known data pertinent to the analysis of structural integrity of the KNPP reactor vessel and therefore meet and exceed the intent of NRC regulation and expectations.

Background for much of this work is linked to ongoing efforts by the NRC staff to generically resolve concerns raised during their review of reactor vessel integrity for the Yankee Rowe Nuclear Power Station. As part of this effort, the NRC staff issued Generic Letter 92-01, Revision 1 and Generic Letter 92-01, Revision 1, Supplement 1. These generic communiqué seek to obtain certain information that will permit the NRC staff to independently assess and ensure that licensees are in compliance with requirements regarding reactor pressure vessel integrity.

During review of the responses to Generic Letter 92-01, Revision 1 and Generic Letter 92-01, Revision 1, Supplement 1 the NRC discovered inconsistencies within the industry concerning the methodology used to assess reactor pressure vessel integrity including:

1. Large variability in the reported chemistries, i.e., copper and nickel contents, for welds fabricated from the same heat of weld wire.
2. Different initial properties (RT_{NDT}) for welds fabricated from the same heat and weld wire.
3. Different transition temperature shifts for welds fabricated from the same heat and weld wire.
4. Operation with irradiation temperature less than 525°F.
5. Different approaches for determining fluence of the limiting material.

In response to this discovery, to provide assurance that all plants will maintain adequate protection against PTS events, the practice of the NRC staff has been to require that evaluations be performed using conservative inputs. This increase in conservatism seems to apply equally to all areas of assessment of reactor vessel integrity. When best estimate values have been used by utilities for the chemical composition of the reactor vessel, it appears that the NRC staff may require the use of increased margin terms to account for potential variability in chemistries. Furthermore, through the process of issuing RAIs, the NRC staff has requested that evaluations be performed using generic values for initial properties and a corresponding higher margin value from either 28°F to 56°F (if the initial RT_{NDT} is measured) or 44°F to 66°F (if the generic RT_{NDT} is used). Other recent changes include the mandatory use of the ratio procedure, if applicable; a 1°F penalty for each degree Fahrenheit when the irradiation temperature is less than 525°F; and other penalties on the projected fluence of the limiting reactor vessel beltline material at end of license. Collectively, this practice of requiring multiple conservative inputs in a layered fashion for assessment of reactor vessel integrity has the effect that a reactor vessel would be predicted to reach the PTS screening criteria at an earlier date than that given by the PTS assessment methodology given in 10 CFR 50.61. A situation of applying too much conservatism can create the illusion that a reactor vessel is unsafe to operate when in fact it may possess sufficient fracture toughness. If too much conservatism is applied the overall affect can be a decrease in safety because of unnecessary changes made to plant operations and design for the sole reason of addressing a conservative but erroneous PTS evaluation.

At about the same time Generic Letter 92-01, Revision 1, Supplement 1 was being issued, the NRC staff became aware of ABB-CE proprietary data that could affect the PTS assessment of the KNPP reactor vessel. Subsequently, ABB-CE provided KNPP a summary of the data for its evaluation in a letter dated April 6, 1995. The NRC staff met with the KNPP staff on April 13, 1995 to discuss the effect that the ABB-CE data and its plant specific surveillance data would have on their PTS assessment. Prior to this meeting, the NRC staff verbally expressed concern to KNPP management that the KNPP reactor vessel may reach the PTS screening criteria before the end of their license. The KNPP staff presented its plant specific surveillance program results and some new information related to the reactor vessel chemistry variability. Based upon using best estimate input parameters, the KNPP staff showed that the KNPP reactor vessel will not reach the PTS screening criteria before the end of their license. Recognizing that the NRC staff was still concerned about the possibility of the KNPP reactor vessel reaching the PTS screening criteria prior to end of license, the KNPP staff remained steadfast in their use of best estimate input parameters for assessment of reactor vessel integrity. At the same time KNPP committed resources to develop industry programs that would facilitate implementation of the applicable requirements specified in the 1992 Edition of Appendix G to 10 CFR 50 should it become necessary: supplemental fracture toughness tests of the beltline material after exposure to neutron irradiation; perform analysis that demonstrates the existence of equivalent margins of safety for continued operation, and thermal annealing. At the conclusion of the April 13, 1995 meeting, the KNPP staff described their future plans to ensure compliance with the requirements for reactor vessel integrity. These plans included participation with industry groups to create programs and a data base detailing the chemical composition of reactor vessel beltline materials; demonstration of the feasibility for annealing of a PWR reactor vessel of US design; and direct measurement of fracture toughness from irradiated surveillance capsule specimens.

In a NRC internal memorandum (dated May 6, 1995 from Jack R. Strosnider, Chief - Materials and Chemical Engineering Branch, Division of Engineering to Ashok C. Thadani, Associate Director for Technical Assessment, Office of Nuclear Reactor Regulation) released following the April 13, 1995 meeting, the NRC staff wrote that they had not completed their review of the new information on the KNPP reactor vessel. The NRC staff noted that the new chemistry data could significantly change the KNPP PTS evaluation. However, based on conservative evaluations, the NRC staff concluded that the KNPP reactor vessel will not reach the PTS screening criteria in the near future. During this same time period, WPSC submitted a proposed amendment to the NRC to modify KNPP Technical Specification limits relating to heatup, cooldown, and low temperature overpressure protection (LTOP). The NRC issued two requests for additional information regarding this proposed amendment, dealing with surveillance capsule fluence and material properties, and then requested that WPSC withdraw it from the docket pending resolution of Generic Letter 92-01, Revision 1, Supplement 1 activities.

While the NRC was performing a detailed review of licensee responses to Generic Letter 92-01, Revision 1, each of the PWR NSSS Owners Groups developed and implemented programs dealing with measurement of fracture toughness for reactor vessel materials. WPSC has funded both the WOG and ABB-CE/RVWG to measure the fracture toughness of the 1P3571 weld metal using the Master Curve Approach. The WOG and ABB-CE/RVWG have obtained unirradiated T_0 values for weld metal 1P3571 in accordance with ASTM E1921-97. The WOG has also obtained the fracture toughness for 1P3571 weld metal from unirradiated 1/2T-CT specimens. Furthermore, the WOG has generated irradiated T_0 values for two coils of 1P3571 weld metal reconstituted from surveillance capsule specimens from the KNPP and Maine Yankee reactor vessels that were irradiated to 3.36×10^{19} n/cm^2 and 6.11×10^{19} n/cm^2, respectively. The ASME B&PVC is currently working under the direction of PVRC to develop recommendations and guidelines for the use of T_0 values in lieu of RT_{NDT} values for assessment of reactor vessel integrity. The results of the supplemental fracture toughness testing for both the unirradiated and irradiated 1P3571 weld metal along with application of the results has been presented to the PVRC and ASME.

WPSC concluded that it is prudent to report the results of the recently completed fracture toughness testing of the EOL and beyond EOL irradiated 1P3571 weld metal along with the values derived for the various PTS evaluations given by the methodology described in 10 CFR 50.61. The results of the irradiated fracture toughness testing will serve as a means of assuring adequate conservatism is incorporated into the integrity assessment of the KNPP reactor vessel. Furthermore, since the fracture toughness transition shift is larger and more accurate than the Charpy transition shift it is felt that continued use of the Charpy results could be inappropriate. The KNPP has volunteered to be a lead plant on behalf of the WOG for application of the Master Curve Approach. NRC feedback obtained on this application of the Master Curve Method will be considered, as appropriate, by the WOG. The fracture toughness results along with the methodology presented in WCAP-15075 indicate that the KNPP 1P3571 weld metal will continue to conservatively provide adequate fracture toughness beyond twice end of life fluence.

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EXECUTIVE SUMMARY

This report provides the methods and results of the Pressurized Thermal Shock (PTS) evaluation performed for the beltine region materials of the Kewaunee Nuclear Power Plant (KNPP) reactor vessel. This evaluation is based on the latest available information on chemistry measurements of welds made from weld wire heat 1P3571, surveillance capsule results from the Maine Yankee program, Charpy V-notch and Master Curve fracture toughness results from the surveillance capsule S from Kewaunee, re-evaluation of the fluences for the Kewaunee surveillance capsules and the reactor pressure vessel, and re-evaluation of the fluence for Maine Yankee surveillance capsule A-35, evaluation of the initial mechanical properties that are realistic for the Kewaunee reactor vessel.

Capsule S was the fourth surveillance capsule tested from the Kewaunee surveillance program. The combined results from capsule S (particularly including the Master Curve fracture toughness data on the weld) provide the bases for evaluating the RT_{PTS} values for the Kewaunee vessel. All of the analyses performed for the forging and girth weld metal of the Kewaunee reactor vessel indicate that no material will violate the PTS screening criteria up to 51 EFPY.

1 INTRODUCTION

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 a steam line break. Such transients may challenge the integrity of a 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 in the vessel wall.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the nil-ductility reference temperature, termed RT_{PTS}[1]. RT_{PTS} screening values were set for beltline base metal, longitudinal seams, and circumferential weld seams for the end-of-license plant operation. The screening criteria were determined using conservative fracture mechanics analysis techniques (see SECY-82-465)[2]. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with these criteria through end of license. The evaluation of reactor pressure vessel embrittlement requires the determination of RT_{PTS} values for the vessel materials. These values are then projected to the end-of-license and compared to the established screening criteria. The procedures for determining RT_{PTS} are described in the PTS Rule, which was published in the Federal Register, December 19, 1995 with an effective date of January 18, 1996. The PTS Rule was amended to make the procedure for calculating RT_{PTS} values consistent with the methods given for the determination of the adjusted reference temperature (ART) in Regulatory Guide 1.99, Revision 2[3].

The intent of the PTS Rule is to provide an upper limit on the fracture toughness reference temperature, RT_{NDT}, for an irradiated reactor pressure vessel. RT_{PTS} is defined in the PTS rule as the projected value of RT_{NDT} at the end-of-license. This projected end-of-license value is then compared to the PTS screening criteria. As described in the report describing the Master Curve approach to transition toughness (WCAP 15075)[4], RT_{NDT} is used as a reference temperature for the K_{IC} and K_{IR} curves, which describe the fracture toughness of a material as a function of temperature. The ASME Code, Paragraph NB-2331 provides a definition that can be used to determine RT_{NDT} for unirradiated materials from drop weight and Charpy V-notch tests. At the time that the PTS rule was published, the only practical means of estimating the irradiated value of RT_{NDT} was to shift the unirradiated by an amount equal to the shift in Charpy V-notch transition temperature plus a reasonable margin term. This Charpy V-notch based procedure is described in the PTS Rule. The development of the Master Curve technology and the recent adoption of ASTM E1921-97[5], have made it possible to evaluate the fracture toughness transition temperature directly. As described in WCAP 15075, the Master Curve technology can be used to define a reference temperature (RT_{To}) that is functionally equivalent to RT_{NDT}.

Because it can be applied to Charpy size specimens, the new ASTM Test Method makes it feasible to estimate RT_{NDT} values directly from irradiated reactor surveillance program materials. This new procedure can provide a more accurate determination of the end-of-license RT_{NDT} value (RT_{PTS}). Section c(3) of 10 CFR 50.61 requires that

> "Any information that and is believed to improve the accuracy of the RT_{PTS} value significantly must be reported to the Director, Office of Nuclear Reactor Regulation."

The purpose of this report is to determine and document the RT_{PTS} values for the Kewaunee reactor vessel to address the revised PTS Rule. Section 2.0 discusses the Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Kewaunee reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusions and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK

The PTS Rule outlines regulations to address the potential for PTS events on pressurized water reactor vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (USNRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a critical size flaw exists along with a reduced fracture resistance do to neutron irradiation. Such an event may result in the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

Appendix G[6] to Part 50 of the Code of Federal Regulations requires testing to determine the fracture toughness of pressure retaining components of the reactor coolant boundary. Materials in the reactor beltline must be monitored with a surveillance program, as described in Appendix H[7]. The purpose of the testing is to demonstrate equivalence with the fracture toughness requirements of Part 50. Although neither Appendix G nor Appendix H require testing beyond the Charpy V-notch and drop weight tests used to determine the unirradiated RT_{NDT}, the Charpy transition temperature shift and Charpy upper shelf energy, both appendices clearly anticipate that supplemental fracture toughness data will be used to improve the analysis. The PTS Rule requires a PTS submittal that must be updated whenever there are changes in core loadings, surveillance measurements or other information that indicates a significant change in projected RT_{PTS} values.

The PTS Rule establishes the following requirements for all domestic, operating PWRs:

- All plants must submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for time of submittal, the expiration date of the operating license, and the projected expiration date if a change in the operating license or renewal has been requested. This assessment must be submitted within six months after the effective date of this Rule if the value of RT_{PTS} for any material is projected to exceed the screening criteria. Otherwise, it must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance capsule report, or within 5 years from the effective date of this Rule change, whichever comes first. These values must be calculated based on the methodology specified in this rule. The submittal must include the following:

  1. the bases for the projection (including any assumptions regarding core loading patterns), and

  2. copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to the NRC, justification must be provided.)

- The RT_{PTS} screening criteria for the reactor vessel beltline region is:

  270°F for base metal (forgings and plates) and longitudianl weld materials, and
  300°F for circumferential weld materials.

- The following equations must be used to calculate the RT_{PTS} values for each weld or base metal in the reactor vessel beltline:

  Equation 1: $RT_{PTS} = I + M + \Delta RT_{PTS}$
  Equation 2: $\Delta RT_{PTS} = CF * f^{(0.28 - 0.10 \log f)}$

- All values of RT_{PTS} must be verified to be bounding values for the specific reactor vessel. In doing this each plant should consider plant-specific information that could affect the level of embrittlement.

- Plant-specific PTS safety analyses are required before a plant is within three years of reaching the screening criteria, including analyses of alternatives to minimize the PTS concern.

- NRC approval for operation beyond the screening criteria is required.

A PTS evaluation for the Kewaunee reactor vessel was documented as WCAP 14280, Rev. 0[8] in March 1995. That evaluation was based on:

- Up-to-date weight percent of copper and nickel for the Kewaunee reactor vessel beltline materials (including the new Lasalle Unit 1 data),

- Projected vessel fluences determined using the neutron dosimetry results from the first four surveillance capsules removed and the ENDF/B-V[9] data set with updated integrated analytical predictions,

- A measured initial RT_{NDT} for the limiting girth weld material developed from the drop weight tests performed in April 1994 (WCAP-14042[10]) and the unirradiated Charpy test data provided in the Kewaunee unirradiated surveillance capsule program (WCAP-8107[11]), and

- Credible surveillance capsule data for the circumferential weld from four surveillance capsules tested to date without applying the ratio procedure.

The Kewaunee PTS submittal was discussed in a meeting with the NRC on April 13, 1995. At that meeting, the NRC requested that the utility perform additional work to resolve uncertainties in the material condition of the Kewaunee reactor vessel. Many of these uncertainties were generated through the evaluation of the Maine Yankee surveillance weld, which was fabricated from the same heat of weld wire (1P3571). The supplemental evaluations performed by the utility are described in WCAP 15074[12]. At the April 13th meeting, the utility also stated their intention of measuring fracture toughness values on irradiated specimens. The results of those supplemental tests are included in this updated PTS submittal.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS}, as defined in the PTS Rule (Code of Federal Regulations, 10 CFR 50.61), is " the reference temperature, RT_{NDT}, evaluated for the EOL fluence for each of the beltline materials." Section C of 10 CFR 50.61 provides specific procedures for the calculation of RT_{PTS}. This calculation requires determination of data on the initial reference temperature (RT_{NDT}), the composition of the steels and the Charpy transition behavior of irradiated specimens. The RT_{NDT} value describes the fracture toughness transition temperature for the material and defines the ASME reference toughness curve. As described in WCAP 15075, the development of Master Curve technology has provided an alternative fracture toughness reference temperature, RT_{To}:.

$$RT_{To} \equiv T_o + 35°F \tag{1}$$

A proposed ASME Code Case currently being considered by ASME Sections III and XI would make the reference temperature determined from the Master Curve , RT_{To}, an alternative means of establishing the reference toughness curve. This proposed Code Case could affect the current definition of RT_{PTS} in two ways. First, the proposed Code Case would allow a new means of determining the initial reference temperature (this is the stated intention of the ASME Code Case). Second, the Code Case opens the possibility of measuring the reference temperature in irradiated materials eliminating the two step estimation based on unirradiated measurements and Charpy shifts. Although this second possibility is not specifically mentioned in the current draft Code Case, the supporting documentation clearly indicates that this is a logical extension of the argument. As such, it must be considered as "information that is believed to improve the accuracy of the RT_{PTS} value," under Section c(3) of 10 CFR 50.61. The following discussion has been broken into four sections:

1. Determination of RT_{PTS} using traditionally measured RT_{NDT} values and Charpy transition temperature shifts per Sections c(1) and c(2) of 10 CFR 50.61.

2. Determination of RT_{PTS} using the Master Curve to determine the unirradiated reference temperature values and Charpy transition shifts per methods sinilar to those described in Sections c(1) and c(2) of 10 CFR 50.61.

3. Determination of RT_{PTS} using the Master Curve to measure irradiated reference temperature values at EOL fluences.

4. Determination of RT_{PTS} using the Master Curve to measure irradiated reference temperature values at arbitrary fluences and extrapolating to EOL fluences.

3.1 DETERMINATION OF RT_{PTS} USING TRADITIONALLY MEASURED RT_{NDT} VALUES AND CHARPY SHIFTS

In the PTS Rule,[2] the NRC Staff has outlined a conservative and uniform method for determining plant-specific values of RT_{PTS} based on Charpy and drop weight methods. For the purpose of comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld or forging in the beltline region as follows.

$$RT_{PTS} = I + M + \Delta RT_{PTS}, \text{ where } \Delta RT_{PTS} = CF * FF \tag{2}$$

where

I = Initial reference temperature (RT_{NDT}) in °F of the unirradiated material as determined according to procedures outlined in Paragraph NB-2331 of the ASME Code, or estimated from generic data

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures, per 10 CFR 50.61 in °F.

$$M = \text{margin} = 2\sqrt{\sigma_\Delta^2 + \sigma_i^2}, {}^\circ F$$

σ_i = 0°F when I is a measured value
σ_i = 17°F when I is a generic value

For base metal plates and forgings:

σ_Δ = 17°F when surveillance capsule data is 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 used
σ_Δ = 14°F when credible surveillance capsule data is used

σ_Δ not to exceed $0.5^*\Delta RT_{PTS}$

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

f = neutron fluence (E>1.0 MeV) at the clad/base metal interface divided by 10^{19} n/cm^2

CF = Chemistry Factor in °F from the tables[3] for welds and base metals (plates and forgings). If plant-specific surveillance data has been deemed credible per Regulatory Guide 1.99, Revision 2, it may be considered in the calculation of the chemistry factor.

3.2 DETERMINATION OF RT_{PTS} USING THE MASTER CURVE TO DETERMINE THE UNIRRADIATED REFERENCE TEMPERATURE AND CHARPY SHIFTS

The simplest application of the Master Curve is to use the unirradiated estimate of RT_{NDT} (RT_{To}) as the initial toughness value (I) in Equation 2. This application of the Master Curve corresponds to Cases 5a and 5b as described in WCAP 15075. In addition to the improved accuracy of T_o as an indexing parameter for the fracture toughness curve, the Master Curve will make it possible to use measured values of toughness in many situations where generic values are currently being used due to the lack of drop-weight data. The uncertainty in the determination of the unirradiated value is defined in ASTM Standard Test Method E1921-97. Equation 2 was originally designed to combine a transition temperature determined using Charpy and drop-weight data with Charpy shifts to determine a conservative reference temperature for the irradiated fracture toughness curve. The use of the Master Curve can significantly improve the accuracy of the unirradiated reference temperature. Equation 2 uses the shift in Charpy transition temperature to estimate the irradiated reference temperature. While data collected by Oak Ridge National Laboratory indicates that there is a general correlation between the Charpy transition temperature shift and fracture toughness transition temperature shift, the correlation contains a large standard deviation term (σ =27°F). This data is discussed in more detail in WCAP 15075. To include this additional source of variability, the definition of σ_Δ has been modified:

$$\sigma_\Delta = \sqrt{\sigma^2_{\Delta CVN} + (27)^2} \quad \text{°F}, \tag{3}$$

where $\sigma_{\Delta CVN}$ is the shift uncertainty for Charpy V-notch measurements defined as above. The margin term, M, is then

$$M = 2\sigma_\Delta . \tag{4}$$

3.3 DETERMINATION OF RT_{PTS} USING THE MASTER CURVE TO MEASURE IRRADIATED RT_{NDT} VALUES AT EOL FLUENCES

Fracture toughness transition curves can be developed by testing Charpy size specimens in accordance with ASTM E1921-97 test method. It is therefore feasible to determine the fracture toughness reference temperature by testing irradiated reactor pressure vessel surveillance materials and applying Equation 1. If surveillance specimens with EOL fluences are available, measurement of RT_{To} would be a direct determination of RT_{PTS}. This procedure corresponds to Case 6 in WCAP 15075. In this case:

$$RT_{PTS} = RT_{To} + M = T_o + 35\text{°F} + M, \tag{5}$$

where M reflects the uncertainty in the determination of T_o.

As indicated in WCAP 15075, the margin term for direct determinations is equivalent to the uncertainty in the T_o value, which is defined in ASTM E1921-97.

3.4 DETERMINATION OF RT_{PTS} USING THE MASTER CURVE TO MEASURE IRRADIATED REFERENCE TEMPERATURE$_T$ VALUES AT ARBITRARY FLUENCES AND EXTRAPOLATING TO EOL FLUENCES

If the available surveillance specimens were not irradiated to end-of-license fluences, a procedure for projecting EOL values is required. This projection can be accomplished by fitting the ΔRT_{NDT} vs. fluence curve shape provided in Paragraph c(1)(ii)(B) of 10 CFR 50.61:

$$\Delta RT_{NDT} = (CF) f^{(0.28-0.1\log f)} \tag{6}$$

to measured values of ΔRT_{NDT}. This projection procedure is described as Case 6 in WCAP 15075. The determination of ΔRT_{NDT} (RT_{NDTi} - RT_{NDTu}) requires measurement of RT_{NDT} in both the unirradiated (RT_{NDTu}) and irradiated (RT_{NDTi})states. Generally, ΔRT_{NDT} values cannot be determined directly using traditional methods because it is not feasible to obtain RT_{NDT} in the irradiated state. However, the use of the Master Curve technology makes determinations of the irradiation induced shift in fracture toughness reference temperatures relatively straight forward. With this data, the chemistry factor, CF, may then be calculated according to the formula:

$$CF = \frac{\sum_{i=1}^{n}\left[A_i x f_i^{(0.28-0.1\log f_i)}\right]}{\sum_{i=1}^{n} f_i^{(0.56-0.2\log f_i)}}, \tag{7}$$

where "A_i " is the measured shift in fracture toughness reference temperature and "f_i " is the fluence for each determination. The PTS Rule requires a minimum of two ΔRT_{NDT} measurements to determine a chemistry factor from Charpy data using this formula. While a similar number of T_o measurements would be preferable, for the purposes of this analysis, chemistry factors have been calculated on the basis of a single determination. The margin applied to these projections should be consistent with the margin applied to projected RT_{NDT} values under the existing procedures. For the purposes of these calculations, a σ_Δ value of 14°F, consistent with the use of measured credible Charpy shifts has been assumed.

4 VERIFICATION OF PLANT-SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Kewaunee vessel was conducted. The beltline region is defined by the Appendix G to 10 CFR 50 to be "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 irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Figure 4.1 identifies and indicates the location of all beltline region materials for the Kewaunee reactor vessel. A review of this figure reveals that the Kewaunee reactor vessel is made of two forgings and one circumferential weld (i.e., no plates or axial welds).

Material property values were obtained from the Kewaunee Generic Letter 92-01 response (NRC-92-081)[13], and WCAP-14279 Rev. 1[14]. The average copper and nickel values used in the calculations are from WCAP 15074 and Appendix A of WCAP 14278, Rev. 1. A summary of the pertinent chemical and mechanical properties of the beltline region materials of the Kewaunee reactor vessel is given in Table 4.1.

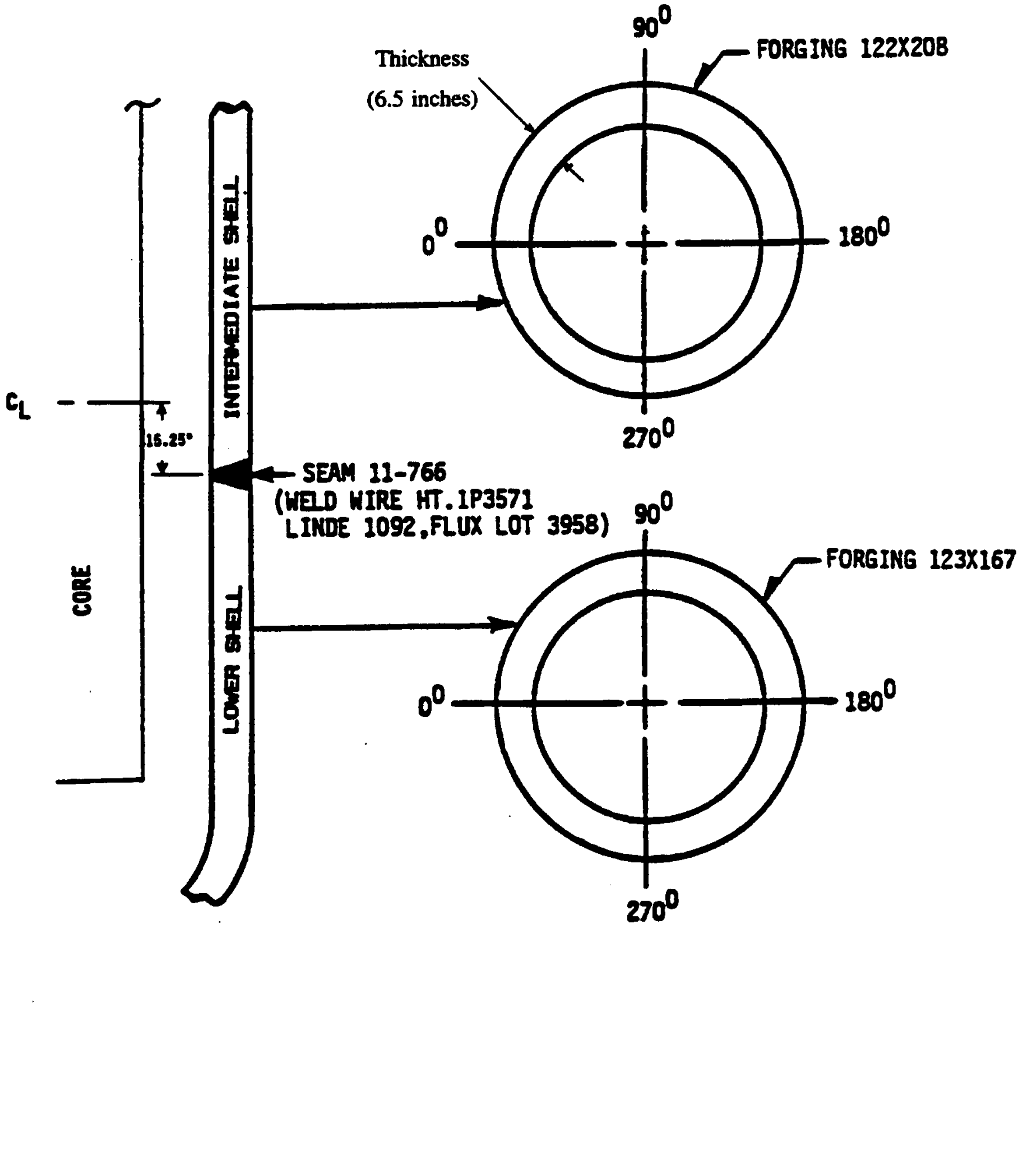

Figure 4-1 Identification and Location of Beltline Region Materials for the Kewaunee Reactor Vessel[8]

Table 4.1 Kewaunee Reactor Vessel Beltline Region Material Properties			
Material	Cu %[1]	N_i %[1]	Initial RT_{NDT}[2]
Intermediate Shell Forging 122X208VA1	0.06	0.71	60
Lower Shell Forging 123X167VA1	0.06	0.75	20
Circumferential Weld 1P3571	0.287	0.756	-50

NOTE:

1. Average weight percents of copper and nickel are from WCAP 15074.
2. Initial RT_{NDT} values are measured values and were obtained from WCAP 14279 Rev. 1.

5 NEUTRON FLUENCE VALUES

The calculated peak fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Kewaunee reactor vessel are shown in Table 5.1. These values were projected using the results of the Capsule S radiation surveillance program documented in WCAP 14279, Rev. 1. The projections were calculated using the ENDF/B-VI dosimetry cross sections. The RT_{PTS} calculations were performed using peak fluence values, which occur at the 0° azimuthal angle of the Kewaunee reactor vessel.

Table 5.1 Neutron Exposure (n/cm², E > 1.0 MeV) Projections at Peak Location on the Kewaunee Pressure Vessel Clad/Base Metal Interface[14]

EFPY	0° Base Metal & Circ. Weld
16.2	1.73×10^{19}
33	3.34×10^{19}
51	5.06×10^{19}

6 DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Kewaunee reactor vessel for fluence values at end of license (33 EFPY) and end of extended license (51 EFPY). The PTS Rule requires that each plant assess the RT_{PTS} values based on plant specific surveillance capsule data whenever:

- Plant specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99, Revision 2, or
- RT_{PTS} values change significantly. (Changes to RT_{PTS} values are considered significant if the value determined with RT_{PTS} equations (1) and (2), or that using capsule data, or both, exceed the screening criteria prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.)

6.1 ANALYSIS OF BELTLINE FORGINGS

Based upon the latest surveillance data available for the Kewaunee beltline forgings contained in WCAP-14279, Rev. 1, the RT_{PTS} values were reassessed. There are two methods for determining the RT_{PTS} values: (1) generic use of the chemistry factor table in the PTS rule (which relies solely on the measured copper and nickel chemistries) with margin (M) = 34°F, and (2) plant-specific use of the forging Charpy shift surveillance data to determine the best-fit chemistry factor (providing it is credible) with M = 17°F. Table 6.1 provides the analysis of the forging Charpy shift data to determine the best-fit chemistry factors. These chemistry factors are less than the value obtained from the chemistry factor table (CF = 37°F for both forging materials). The forging data do not meet the credibility requirements specified in the PTS Rule since both the lowest and highest fluence shift results differ by more than one standard deviation (17°F) from the predicted value using the surveillance-specific CF value. Therefore, the appropriate CF value to use for the forgings is the value from the chemistry factor table with M = 34°F.

For comparison purposes, both methods for determining the chemistry factor have been evaluated in Tables 6.2 and 6.3. Note that the initial RT_{NDT} (I) values for the forging materials have been measured following ASME Code rules. The end-of-license fluence of 3.34×10^{19} n/cm^2 (E > 1MeV) results in the RT_{PTS} values in Table 6.2. Similarly, RT_{PTS} values for an extended end-of-license fluence of 5.06×10^{19} n/cm^2 (E > 1MeV) are shown in Table 6.3. The higher values, corresponding to the method labeled "Without S/C Data" (indicating without surveillance capsule data), are the most appropriate values to be used.

6.2 ANALYSIS OF CIRCUMFERENTIAL WELD

6.2.1 Traditional RT_{NDT} and Charpy Shifts

In addition to the data from the fourth surveillance capsule, several additional sources of chemistry data were identified for the Kewaunee circumferential weld. This additional data is discussed in detail in WCAP 15074. Recent requests by the NRC, outlined in Generic Letter 92-01, have required that all possible sources of data must be evaluated. This requirement to evaluate such a diverse population of data has increased the complexity of the problem and leaves many subtle questions of interpretation of the regulations that can significantly affect the outcome. Although both the PTS Rule and Regulatory Guide 1.99, Revision 2 provide procedures for the determination of reference temperature, there are a number of remaining issues when data from multiple sources are applied to a particular heat of steel. While these issues are not important in the evaluation of the Kewaunee forging materials, they are critical to the evaluation of the circumferential weld. This situation is illustrated in WCAP 15074, where four different methods of evaluating the critical circumferential weld using traditional RT_{NDT} determinations and Charpy transition temperature shifts in the Kewaunee reactor vessel are presented. The EOL evaluations for the circumferential weld ($3.34x10^{19}$ n/cm^2) are summarized in WCAP 15075 as Table 7.1, which is reproduced here in Table 6.4. Similar evaluations for extended operations (circumferential weld fluence equals $5.06x10^{19}$ n/cm^2) are presented in WCAP 15075 as Table 7.3 and reproduced here as Table 6.5. The four Charpy based cases illustrate the difficulty of combining multiple measurements of initial RT_{NDT}, with chemical analysis from multiple sources and with surveillance data from two surveillance programs. These issues are discussed at length in WCAP 15074. On the basis of that discussion, it was concluded that Case 3, which combined the I (IRT) value measured for the Kewaunee vessel (-50°F) with a Charpy shift determined from surveillance measurements (i.e., credible surveillance data), was the most appropriate method for determining ART (RT_{PTS}.) Applying this calculational procedure for the Kewaunee circumferential weld provides the following best estimates for traditional application of Charpy data:

EOL ($3.34x10^{19}$ n/cm^2) $RT_{PTS} = 267°F$

Ext. EOL ($5.06x10^{19}$ n/cm^2) $RT_{PTS} = 287°F$

6.2.2 Unirradiated Master Curve Measurements and Charpy Shifts

Due to the potential variability and conservatism in IRT direct measurement of fracture toughness testing of archival weld material from the Kewaunee surveillance program was conducted to provide supplemental information on the unirradiated material condition. These results are described in WCAP 14279 Rev. 1.. Procedures for using Master Curve test results to estimate unirradiated RT_{NDT} values are described in WCAP 15075 and summarized in Section 3.2 of the current volume.. If the Master Curve is used only to set the unirradiated reference temperature (IRT), then Charpy data must be used to estimate the radiation induced shift in the reference temperature. The radiation induced shift has been estimated using the standard, Charpy based procedures (with and without credible surveillance results) in

WCAP 15075 (Cases 5a and 5b). In these two cases, data from the Maine Yankee surveillance program have been considered to the calculated ΔRT_{NDT} value either through the use of industry best estimate chemistry values or through the use of the ratio method. Consistent with the preceding discussion, the surveillance data for the Kewaunee surveillance weld was judged to be credible. The prediction of the EOL ($f = 3.34\times10^{19}$ n/cm^2) RT_{PTS} value based on unirradiated Master Curve testing and Charpy shifts is summarized in Table 6.4. Similar calculations for the extended EOL ($f = 5.06\times10^{19}$ n/cm^2) are summarized in Table 6.5. These calculations are included only for informational purposes, as irradiated T_0 measurements are available and more accurate predictions are possible.

6.2.3 Measurement of EOL Value using the Master Curve

Fracture toughness testing of small three point bend specimens reconstituted from broken Charpy bars were conducted to provide supplemental information on the irradiated material condition. Irradiated specimens from both the Maine Yankee and Kewaunee surveillance programs were tested. The Kewaunee surveillance program includes one capsule (capsule S) irradiated to a fluence that corresponds to the end-of-license fluence for the circumferential weld. Therefore, it is possible to use procedures described in WCAP 15075 and summarized in Section 3.3 of the current volume to make a direct determination of RT_{PTS} (no projections required). The direct determination corresponds to case 6 as outlined in WCAP 15075. In that case, an additional 35°F was added to the reference temperature to account for the generally higher ductile-to-brittle transition temperature observed in the Maine Yankee surveillance material. This 35°F value was determined by analyzing the irradiated data from the Maine Yankee program and applying a slightly modified ratio method. Details of this calculation are outlined in WCAP 15075. The EOL ($f = 3.34\times10^{19}$ n/cm^2) RT_{PTS} value for the Kewaunee circumferential weld, based on direct measurement of the irradiated reference temperature, is 234°F. This calculation is summarized in Table 6.4. Slightly modified procedures are required for the fluence evaluations because fracture toughness measurements on Kewaunee surveillance specimens at 5.06×10^{19} n/cm^2 are not available.

6.2.4 Extrapolation to Extended EOL Value Using Master Curve

The extension of the reactor operation to 51 EFPY requires extrapolation beyond the currently available Kewaunee surveillance data. One procedure for using Master Curve data as the basis for such a projection is outlined in WCAP 15075 (Case 6) and summarized in Section 3.4 of the current report. This extrapolation procedure matches the embrittlement curve provided in 10 CFR 50.61 to the Master Curve data by appropriately adjusting the chemistry factor. Estimates based on extrapolations of the embrittlement curve to the extended operation fluence (5.06×10^{19} n/cm^2) are summarized in Table 6.5.

Although fracture toughness data on the Kewaunee surveillance weld is currently limited to EOL fluence level, the existence of higher fluence data on the 1P3571 weld from the Maine Yankee surveillance program allows more accurate predictions of the fracture toughness at extended EOL fluences. The calculations of the improved estimates are summarized as Case 6 in Table 6.5. RT_{T_0} for the Maine Yankee surveillance weld at 6.11×10^{19} n/cm^2 is 267°F. Based on fluence factor included in the 10 CFR 50.61 trend curve, the Maine Yankee surveillance weld reference temperature at 5.06×10^{19} n/cm^2 is 257°F. Adding the appropriate margin terms gives an extended EOL ART value for the Maine Yankee surveillance weld of 281°F. The Maine Yankee weld has a substantially higher copper content than the industry average value applied to the Kewaunee circumferential weld. Therefore, the RT_{PTS} value for the Kewaunee vessel is less than 281°F. Applying a ratio method to the Maine Yankee prediction gives a best estimate of 249°F for RT_{PTS} in the Kewaunee circumferential weld.

Table 6.1 Calculation of Chemistry Factors for Forging Material Using Kewaunee Surveillance Capsule Data[11,14]

Material	Capsule	f	FF	ΔRT_{NDT}	$FF^*\Delta RT_{NDT}$	FF^2
Intermediate Shell Forging 122X208VA1 (Tangential)	V	0.597	0.86	0	0	0.732
	R	1.81	1.16	15	17.4	1.352
	P	2.74	1.27	25	31.7	1.610
	S	3.36	1.32	60	79.0	1.735
				SUM	128.2	5.429
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = 23.6$					
Lower Shell Forging 123X167VA1 (Tangential)	V	0.597	0.86	0	0	0.732
	R	1.81	1.16	20	23.3	1.352
	P	2.74	1.27	20	25.4	1.610
	S	3.36	1.32	50	65.9	1.735
				SUM	114.5	5.429
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = 21.1$					

<u>NOTES</u>:

f = fluence (E > 1.0 MeV) ÷ 10^{19} n/cm^2; All values taken from Capsule S analysis, WCAP-14279, Rev. 1.

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

∴ Chemistry Factor for Intermediate Shell Forging 122X208VA1 based on surveillance capsule data = 23.6°F

∴ Chemistry Factor for Lower Shell Forging 123X167VA1 based on surveillance capsule data = 21.1°F

Table 6.2 PTS Calculations for Kewaunee Forging Material at 33 EFPY								
Material	Method	$CF^{(1)}$	$f^{(2)}$	$FF^{(3)}$	$I^{(4)}$	$M^{(5)}$	$\Delta RT_{PTS}{}^{(6)}$	$RT_{PTS}{}^{(7)}$
33 EFPY								
Intermediate Shell Forging	Without S/C Data	37	3.34	1.32	60	34	48.7	143
122X208VA1	With S/C Data	23.6	3.34	1.32	60	17	31.1	108
Lower Shell Forging	Without S/C Data	37	3.34	1.32	20	34	48.7	103
123X167VA1	With S/C Data	21.1	3.34	1.32	20	17	27.8	65

<u>NOTE:</u>

1. CF = chemistry factor, °F
2. f = Peak surface fluence, 10^{19} n/cm^2 (E > 1.0 MeV) + 10^{19} n/cm^2 (E > 1.0 MeV)
3. FF = fluence factor = $f^{(0.28\ 0.1 * \log(f))}$
4. I = initial RT_{NDT} of material, °F (all values are measured)
5. M = margin, °F
6. ΔRT_{PTS} = CF * FF, °F
7. RT_{PTS} = I + M + ΔRT_{PTS}, °F

Table 6.3 PTS Calculations for Kewaunee Forging Material at 51 EFPY								
Material	Method	CF(1)	f(2)	FF(3)	I(4)	M(5)	ΔRT_{PTS}(6)	RT_{PTS}(7)
51 EFPY								
Intermediate Shell Forging	Without S/C Data	37	5.06	1.40	60	34	68.4	162
122X208VA1	With S/C Data	23.6	5.06	1.40	60	17	43.6	121
Lower Shell Forging	Without S/C Data	37	5.06	1.40	20	34	68.4	122
123X167VA1	With S/C Data	21.1	5.06	1.40	20	17	39.0	76

NOTE:

1. CF = chemistry factor, °F
2. f = Peak surface fluence, 10^{19} n/cm^2 (E > 1.0 MeV) ÷ 10^{19} n/cm^2 (E > 1.0 MeV)
3. FF = fluence factor = $f^{(0.28\ 0.1 * \log(f))}$
4. I = initial RT_{NDT} of material, °F (all values are measured)
5. M = margin, °F
6. ΔRT_{PTS} = CF * FF, °F
7. RT_{PTS} = I + M + ΔRT_{PTS}, °F

Table 6-4. ART Determination for the Kewaunee Weld and Vessel (Table 7.1 from WCAP 15075)									
Method	Best Estimate of Initial RT_{NDT} Value (IRT), °F	Standard Deviation for IRT (σ_I), °F	Kewuanee Surveillance Estimate of Shift (ΔRT), °F	Standard Deviation for ΔRT (σ_D), °F	Best Estimate of Irradiated Value, °F	Total Margin (M) °F	Adjusted Reference Temperature (ART) °F	Additional Adjustment for Heat Uncertainty (ΔRT_{HT}) °F	Heat Adjusted Reference Temperature (ART_{HT}) °F
1.) Current Technology Measured IRT; No Credible CVN Data	Measured Value, RT_{NDT} = -50	"Assumed" 0	RG1.99R2, CF Table 246	RG1.99R2 28	IRT+ΔRT 196	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 56	IRT+ΔRT+M 252	Ind. Mean Chemistry 36	288
2.) Current Technology; Generic IRT; No Credible CVN Data	PTS Rule RT_{NDT} = -56	PTS Rule 17	RG1.99R2, CF Table 246	RG1.99R2 28	IRT+ΔRT 190	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 66	IRT+ΔRT+M 256	Ind. Mean Chemistry 36	292
3.) Current Technology; Measured IRT; Credible CVN Data	Measured Value, RT_{NDT} = -50	"Assumed" 0	RG1.99R2, Data Fit 253	RG1.99R2 14	IRT+ΔRT 203	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 28	IRT+ΔRT+M 231	Ratio Adj. 36	267
4.) Current Technology ; Generic IRT; Credible CVN Data	PTS Rule RT_{NDT} = -56	PTS Rule 17	RG1.99R2, Data Fit 253	RG1.99R2 14	IRT+ΔRT 197	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 44	IRT+ΔRT+M 241	Ratio Adj. 36	277
5a.) Master Curve ; Unirradiated To ; Credible CVN Data	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	RG1.99R2, Data Fit 253	RG1.99R2 & To to CVN 30	$RT_{NDT(U)}$ $+\Delta RT_{NDT}$ 144	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 62	IRT_{To} $+\Delta RT_{NDT}$+M 206	Ratio Adj. 36	242
5b.) Master Curve; Unirradiated To ; No Credible CVN Data	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	RG1.99R2, CF Table 246	RG1.99R2 & To to CVN 39	$RT_{NDT(U)}$ $+\Delta RT_{NDT}$ 137	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 79	IRT_{To} $+\Delta RT_{NDT}$+M 216	Ind. Mean Chemistry 36	252
6.) Master Curve; Irradiated To	NA	NA	NA	ASTM $\sigma_{To} = \beta/\sqrt{n}$ 8	Irradiated To +35°F 183	$2\sigma_{To}$ 16	$RT_{To(irr)}$ + M 199	MY meas. w/ Ratio Adj. 35	234
7.) Master Curve Shift; Measured $RT_{NDT(U)}$; Irr. To-Unirr. To	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	Data Fit, CF = 222 292	Similar to RG1.99R2 14	RT_{To} $+\Delta RT_{To}$ 183	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 31	IRT $+\Delta RT_{To}$+M 214	MY meas. w/ Ratio Adj. 35	249

Table 6-5. ART Determination for the Kewaunee Weld and Vessel ($5.06x10^{19}$ n/cm^2 version) (Table 7.3 from WCAP 15075)									
Method	Best Estimate of Initial RT_{NDT} Value (IRT), °F	Standard Deviation for IRT (σ_I), °F	Kewuanee Surveillance Estimate of Shift (ΔRT), °F	Standard Deviation for ΔRT (σ_D), °F	Best Estimate of Irradiated Value, °F	Total Margin (M) °F	Adjusted Reference Temperature (ART) °F	Additional Adjustment for Heat Uncertainty (ΔART_{HT}) °F	Heat Adjusted Reference Temperature (ART_{HT}) °F
1.) Current Technology Measured IRT; No Credible CVN Data	Measured Value, RT_{NDT} = -50	"Assumed" 0	RG1.99R2, CF Table 263	RG1.99R2 28	IRT+ΔRT 213	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 56	IRT+ΔRT+M 269	Ind. Mean Chemistry 39	308
2.) Current Technology; Generic IRT; No Credible CVN Data	PTS Rule RT_{NDT} = -56	PTS Rule 17	RG1.99R2, CF Table 263	RG1.99R2 28	IRT+ΔRT 207	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 66	IRT+ΔRT+M 273	Ind. Mean Chemistry 39	312
3.) Current Technology; Measured IRT; Credible CVN Data	Measured Value, RT_{NDT} = -50	"Assumed" 0	RG1.99R2, Data Fit 270	RG1.99R2 14	IRT+ΔRT 220	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 28	IRT+ΔRT+M 248	Ratio Adj. 39	287
4.) Current Technology ; Generic IRT; Credible CVN Data	PTS Rule RT_{NDT} = -56	PTS Rule 17	RG1.99R2, Data Fit 270	RG1.99R2 14	IRT+ΔRT 214	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 44	IRT+ΔRT+M 258	Ratio Adj. 39	297
5a.) Master Curve ; Unirradiated To ; Credible CVN Data	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	RG1.99R2, Data Fit 270	RG1.99R2 & To to CVN 30	RT_{To} +ΔRT_{NDT} 161	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 62	IRT_{To} +ΔRT_{NDT}+M 223	Ratio Adj. 39	262
5b.) Master Curve; Unirradiated To ; No Credible CVN Data	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	RG1.99R2, CF Table 263	RG1.99R2 & To to CVN 39	RT_{To} +ΔRT_{NDT} 154	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 79	IRT_{To} +ΔRT_{NDT}+M 233	Ind. Mean Chemistry 39	272
6.) Master Curve; Irradiated To *See Note* *M.Y. Fluence* *Kewaunee Ext. EOL*	NA	NA	NA	ASTM $\sigma_{To} = \beta/\sqrt{n}$ 12	To +35°F Meas. 267 Est. 257	$2\sigma_{To}$ 24	$RT_{To(irr)}$ + M 291 281	Kew. meas. w/ Ratio Adj. -32	$6.1x10^{19}$ 259 $5.1x10^{19}$ 249
7.) Master Curve Shift; Measured $RT_{NDT(U)}$; Irr. To-Unirr. To	Unirradiated To +35°F RT_{To} = -109	ASTM $\beta/\sqrt{n}$ 7	Data Fit, CF = 222 311	Similar to RG1.99R2 14	RT_{To} +ΔRT_{To} 202	$2(\sigma_I^2+\sigma_\Delta^2)^{1/2}$ 31	IRT_{To} +ΔRT_{To}+M 233	MY meas. w/ Ratio Adj. 37	270

Note: Case 6 based on the Maine Yankee Measurement at $6.1x10^{19}$ n/cm^2. The result was then ratioed back to the Kewaunee vessel chemistry.

7 CONCLUSIONS

As shown in Tables 6.2 and 6.3, RT_{PTS} values for the limiting Kewaunee forging remain well below the NRC screening values of 270°F for PTS. The changes in the ductile-to-brittle transition temperature are relatively small and the forging raises no significant PTS issues.

RT_{PTS} values were calculated for the Kewaunee circumferential weld using two different procedures:

1. Traditional RT_{NDT} and Charpy based procedures,
2. Direct determination using irradiated and unirradiated Master Curve data.

Various permutations on these basic procedures were illustrated in Tables 6.4 (EOL fluences) and 6.5 (extended EOL fluences). Best estimates of the RT_{PTS} value were based on the most appropriate application of these procedures. The best estimate RT_{PTS} values are summarized in Table 7.1. The circumferential weld seam is the limiting beltline material in the Kewaunee reactor vessel. However, all of the proposed procedures predict RT_{PTS} values below the PTS screening criterion (300°F) for a circumferential weld. The Master Curve based methodology is recommended as the most reliable estimate of fracture toughness transition temperature. It should be noted that the Master Curve prediction for extended EOL operation is based on fracture toughness data from the Maine Yankee surveillance program and application of the ratio procedure. Additional fracture toughness testing of higher fluence surveillance weld metal materials from the Kewaunee surveillance program will provide further verification of the reference temperature for extended EOL operation.

Table 7.1 Summary of RT_{PTS} Calculations for Kewaunee Circumferential Weld

Methodology	**EOL (3.34×10^{19} n/cm^2)**	**Extended EOL (5.06×10^{19} n/cm^2)**
Traditional RT_{NDT} and Charpy	267°F	287°F
Irradiated Master Curve	234°F	249°F

8 REFERENCES

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5. ASTM E1921-97, "Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range", in ASTM Standards, Section 3, American Society for Testing and Materials.

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7. Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, No. 243 dated December 19, 1995.

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