Westinghouse Non-Proprietary Class 3

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Revision 2

# BYRON UNIT 1 HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION AND SURVEILLANCE WELD METAL INTEGRATION FOR BYRON & BRAIDWOOD

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WCAP-14824, Revision 2

# Byron Unit 1
# Heatup and Cooldown Limit Curves
# For Normal Operation
# and Surveillance Weld Metal Integration
# for Byron and Braidwood

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Prepared by the Westinghouse Electric Corporation
for the Commonwealth Edison Company

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## PREFACE

This report has been technically reviewed and verified by:

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# 1 INTRODUCTION

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

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

## 2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[2]. The pre-irradiation fracture-toughness properties of the Byron Unit 1 reactor vessel are presented in Table 3. Credible surveillance data is available for two capsules (Capsules U and X) for Byron Unit 1. The post-irradiation fracture toughness properties of the reactor vessel beltline material were obtained directly from the Byron Unit 1 Reactor Vessel Radiation Surveillance Program[3]. This capsule data is used to calculate chemistry factors (See Table 4) in addition to those calculated per Regulatory Guide 1.99, Revision 2.

Additionally, per the request of the Commonwealth Edison Company, the surveillance weld data from the Byron Unit 1 and Byron Unit 2 surveillance programs[4] has been integrated pursuant to 10 CFR 50.61 in accordance with Regulatory Guide 1.99, Revision 2, Position 2. In addition to the credible surveillance weld data from Byron Unit 1, credible surveillance weld data is available for two capsules (Capsules U and W) for Byron Unit 2. The chemistry factor values resulting from the weld metal integration for Byron Units 1 and 2 are presented in Section 4 of this report. See Tables 1 through 4.

A complete technical justification for the Byron Units 1 and 2 weld metal integration is presented in Appendix A of this report.

# 3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[5] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components"[6], *Vessels*, contain the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, $K_I$, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, $K_{Ia}$, for the metal temperature at that time. $K_{Ia}$ is obtained from the reference fracture toughness curve, defined in Appendix G of the ASME Code, Section XI[7]. The $K_{Ia}$ curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.233 * e^{[0.0145(T - RT_{NDT} + 160)]} \quad (1)$$

where,

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

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

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

where,

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

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

$K_{Ia}$ = function of temperature relative to the $RT_{NDT}$ of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

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

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

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

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

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

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

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

10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated $RT_{NDT}$ by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is 621 psig for Byron Unit 1.

The limiting unirradiated $RT_{NDT}$ of 60°F occurs in the closure head flange of the Byron Unit 1 reactor vessel, so the minimum allowable temperature of this region is 180°F at pressures greater than 621 psig. This limit is shown in Figures 1 through 4 wherever applicable.

# 4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

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

$$ART = Initial\ RT_{NDT} + \Delta\ RT_{NDT} + Margin \qquad (3)$$

Initial $RT_{NDT}$ is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8]. If measured values of initial $RT_{NDT}$ for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

$\Delta RT_{NDT}$ is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta\ RT_{NDT} = CF * f^{(0.28-0.10 \log f)} \qquad (4)$$

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

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (5)$$

where x inches (vessel beltline thickness is 8.5 inches[14]) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 4 to calculate the $\Delta RT_{NDT}$ at the specific depth. The calculated surface fluence for Byron Unit 1 upper and lower shell forgings and circumferential weld at 12 EFPY is $8.10 \times 10^{18}$ n/cm$^2$. This fluence value was calculated from the surveillance Capsule X analysis presented in WCAP-13880[9].

Explanation for the Application of the Credibility Criteria and the Ratio Procedure

In calculating $RT_{PTS}$ values in accordance with 10 CFR 50.61 (which incorporates Regulatory Guide 1.99 Revision 2 in total) and ART values for input to 10 CFR 50 Appendix G pressure-temperature limit curves, Commonwealth Edison (ComEd) uses the methodology described in Regulatory Guide 1.99 Revision 2. When there are two or more sets of surveillance data available, which there is in this case, Regulatory Guide 1.99 Revision 2 provides criteria for evaluating the credibility of the surveillance data and subsequently a procedure for determining a best-fit chemistry factor which represents the actual behavior of the material, normalized to the vessel of interest.

In particular, the third credibility criteria from Reg. Guide 1.99 Rev. 2 states that the "scatter of $\Delta RT_{NDT}$ 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." This evaluation of credibility becomes a comparison between the actual measured surveillance data shifts, and a line drawn using Equation 2 from Reg. Guide 1.99 Rev. 2 with a Regulatory Position 2.1 least-squares-fit chemistry factor based on the actual surveillance data set. This provides an indication of whether the surveillance material, with its specific measured chemistry, is behaving as the Reg. Guide 1.99 Rev. 2 Equation 2 irradiation damage correlation would predict.

Thus, for welds if it is determined that the surveillance data is credible relative to Reg. Guide 1.99 Rev. 2, and "if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, i.e., differs from the average [currently taken to be the 'best estimate' chemistry] for the weld wire heat number associated with the vessel weld and the surveillance weld,"[1] then for subsequent $RT_{PTS}$ and ART calculations, the ratio procedure of Regulatory Position 2.1 is used to normalize the observed behavior of the surveillance material to the expected behavior of the vessel weld. The measured values of $\Delta RT_{NDT}$ obtained from surveillance data are adjusted by multiplying them by the ratio of the Regulatory Position 1.1 (Table 1) chemistry factor for the vessel weld to that for the surveillance weld. The ratio-adjusted surveillance data is used to calculate a least-squares-fit chemistry factor appropriate to the vessel weld.

The chemistry factors (CF, °F) obtained from the tables in Reg. Guide 1.99 Revision 2 using the average values of copper and nickel content as calculated in Tables 1 and 2 are reported in Table 3. The chemistry factors were also calculated using the surveillance capsule data in Table 4.

The Ratio Procedure, as documented in Regulatory Guide 1.99 Revision 2 Position 2.1, was used to adjust the measured values of $\Delta RT_{NDT}$ for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material (best-estimate chemistry) to that for the surveillance weld.

All materials in the beltline region of Byron Unit 1 reactor vessel were considered in determining the limiting material and the calculations to determine the ART values at 12 EFPY are shown in Table 5. The resulting ART values for all beltline region materials at the 1/4T and 3/4T locations are summarized in Table 6, where it can be seen that the limiting material is the Intermediate Shell Forging 5P-5933 (based on credible surveillance capsule data). The 1/4T and 3/4T ART values for Intermediate Shell Forging 5P-5933 (based on credible surveillance capsule data) will be used in the generation of heatup and cooldown curves applicable to 12 EFPY.

TABLE 1
Calculation of Average Cu and Ni Weight Percent Values
for the Byron Unit 1 Base Materials

| Reference | Intermediate Shell Forging 5P-5933 | | Lower Shell Forging 5P-5951 | |
|---|---|---|---|---|
| | Cu% | Ni% | Cu% | Ni% |
| **Byron Unit 1 HU/CD Limit Curves** | 0.034 | 0.73 | 0.04 | 0.64 |
| | 0.032 | 0.791 | | |
| | 0.03 | 0.75 | | |
| **Letter Report FDRT/ SRPLO-09(94)** | 0.05 | 0.73 | | |
| **January 1994** | 0.036 | 0.735 | | |
| Average | 0.0364 | 0.747 | 0.04 | 0.64 |
| Standard Deviation | 0.007 | 0.023 | 0 | 0 |

TABLE 2

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

| | Reference | Best-Estimate Cu | Best-Estimate Ni | | | | |
|---|---|---|---|---|---|---|---|
| B&W Weld Qualification | BAW-2261 | 0.024 | 0.7 | | | | |
| B&W Weld Qualification | " | 0.031 | 0.46 | | | | |
| B&W Weld Qualification | " | 0.03 | 0.72 | | | | |
| B&W Weld Qualification | " | 0.068 | 0.48 | | | | |
| B&W Weld Qualification | " | 0.114 | 0.54 | | | | |
| B&W Weld Qualification | " | 0.148 | 0.6 | | | | |
| B&W Weld Qualification | " | 0.053 | 0.62 | | | | |
| B&W Weld Qualification | " | 0.059 | 0.62 | | | | |
| B&W Weld Qualification | Ref. 23 | 0.029 | 0.65 | | | | |
| Byron 1 Surveillance Data | See Below | 0.022 | 0.690 | ---> | 0.02 | 0.69 | Surv. CF = 27 |
| Byron 2 Surveillance Data | See Below | 0.023 | 0.712 | ---> | 0.02 | 0.71 | Surv. CF = 27 |
| Best-Estimate Chemistry(d): | | 0.055 | 0.617 | ---> | **0.05** | **0.62** | **Best Est. CF = 68** |
| Standard Deviation: | | 0.042 | 0.091 | | | **Byron 1 & 2 Ratio = 2.5(c)** | |

*Surveillance Data Chemistry Results:*

***Byron Unit 1***

| Reference | Cu | Ni |
|---|---|---|
| WCAP-9517[3] | 0.026 | 0.71 |
| WCAP-11651[21] | 0.023 | 0.67 |
| | 0.022 | 0.665 |
| | 0.021 | 0.714 |
| | 0.021 | 0.741 |
| | 0.022 | 0.713 |
| | 0.021 | 0.714 |
| | 0.020 | 0.704 |
| | 0.020 | 0.694 |
| | 0.020 | 0.706 |
| | 0.021 | 0.677 |
| | 0.023 | 0.677 |
| | 0.021 | 0.680 |
| | 0.021 | 0.680 |
| | 0.021 | 0.667 |
| | 0.024 | 0.677 |
| | 0.022 | 0.697 |
| | 0.021 | 0.634 |
| WCAP-13880[9] | 0.024 | 0.682 |
| | 0.022 | 0.678 |
| | 0.025 | 0.705 |
| Average | **0.022** | **0.690** |

***Byron Unit 2***

| Reference | Cu | Ni |
|---|---|---|
| WCAP-10398[4] | 0.03 | 0.65 |
| WCAP-12431[22] | 0.024 | 0.740 |
| | 0.024 | 0.786 |
| | 0.022 | 0.704 |
| | 0.020 | 0.681 |
| | 0.021 | 0.706 |
| | 0.020 | 0.697 |
| | 0.019 | 0.668 |
| | 0.022 | 0.759 |
| | 0.021 | 0.714 |
| | 0.020 | 0.678 |
| | 0.020 | 0.695 |
| | 0.019 | 0.689 |
| | 0.021 | 0.744 |
| | 0.022 | 0.738 |
| | 0.022 | 0.771 |
| WCAP-14064[11] | 0.024 | 0.705 |
| | 0.023 | 0.706 |
| | 0.023 | 0.698 |
| | 0.024 | 0.696 |
| | 0.023 | 0.711 |
| | 0.024 | 0.708 |
| | 0.024 | 0.716 |
| | 0.024 | 0.715 |
| | 0.024 | 0.707 |
| | 0.024 | 0.720 |
| | 0.024 | 0.717 |
| | 0.024 | 0.711 |
| | 0.024 | 0.706 |
| | 0.024 | 0.707 |
| | 0.025 | 0.717 |
| Average | **0.023** | **0.712** |

TABLE 2 NOTES:

(a) The weld material in the Byron Unit 1 surveillance program was made of the same wire and flux as the reactor vessel intermediate to lower shell girth seam weld. (Weld seam WF-336, Wire Heat No. 442002, Flux Type Linde 80, Flux Lot No. 8873)

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

(c) Actual ratio is 2.5 (68.0 ÷ 27.0 = 2.5), however, for conservatism a ratio of 3.0 will be used herein.

(d) The best estimate chemistry values was obtained using the "average of averages" approach.

TABLE 3
Byron Unit 1 Reactor Vessel Material Properties

| Material Description | Cu (%) | Ni (%) | Chemistry Factor[a] | Initial $RT_{NDT}$ (°F)[b] |
|---|---|---|---|---|
| Closure Head Flange | -- | 0.74 | -- | 60[c] |
| Vessel Flange | -- | 0.73 | -- | 10[c] |
| Intermediate Shell Forging 5P-5933 | 0.0364 | 0.747 | 23.8 | 40 |
| Lower Shell Forging 5P-5951 | 0.04 | 0.64 | 26.0 | 10 |
| Circumferential Weld WF-336 | 0.05 | 0.62 | 68.0 | -30 |

NOTES:

(a) Chemistry Factors are calculated from Cu and Ni values per Regulatory Guide 1.99, Revision 2, Position 1.1

(b) Initial $RT_{NDT}$ values are measured values.

(c) Closure head and vessel flange Initial $RT_{NDT}$ values are used for considering flange requirements[5] for the heatup/cooldown curves.

TABLE 4
Calculation of Chemistry Factors Using Credible Byron Units 1 and 2 Surveillance Capsule Data

| Material | Capsule | Capsule Fluence f | FF[(a)] | Meas. $\Delta RT_{NDT}$ | | FF* $\Delta RT_{NDT}$ | FF[2] |
|---|---|---|---|---|---|---|---|
| Inter. Shell Forging 5P-5933 (Tangential) | U | $3.72\times10^{18}$ | 0.727 | 0 | | 0 | 0.529 |
| | X | $1.39\times10^{19}$ | 1.091 | 30 | | 32.73 | 1.19 |
| Inter. Shell Forging 5P-5933 (Axial) | U | $3.72\times10^{18}$ | 0.727 | 0 | | 0 | 0.529 |
| | X | $1.39\times10^{19}$ | 1.091 | 30 | | 32.73 | 1.19 |
| | | | | | Sum: | 65.46 | 3.44 |
| | Chemistry Factor[(d)] = 65.46 ÷ 3.44 = **19.0°F** | | | | | | |
| Byron 1 Weld Metal WF-336[(b)] | U | $3.72\times10^{18}$ | 0.727 | 0 | 0 | 0.00 | 0.529 |
| | X | $1.39\times10^{19}$ | 1.091 | 35 | 105[(e)] | 114.56 | 1.19 |
| Byron 2 Weld Metal WF-447[(c)] | U | $3.996\times10^{18}$ | 0.746 | 0 | 0 | 0.00 | 0.557 |
| | W | $1.211\times10^{19}$ | 1.053 | 30 | 90[(e)] | 94.77 | 1.110 |
| | | | | | Sum: | 209.33 | 3.386 |
| | Chemistry Factor[(d)] = 209.33 ÷ 3.386 = **61.8°F** | | | | | | |

NOTES:

(a) FF = Fluence Factor = $f^{(0.28 - 0.1 * \log f)}$

(b) Byron Unit 1 $\Delta RT_{NDT}$ values were obtained from the surveillance Capsule X analysis (WCAP-13880). The Byron Unit 1 capsule fluence values were recalculated using the ENDF/B-V scattering cross sections in 1994 and are documented in WCAP-14044[10].

(c) Byron Unit 2 capsule fluence, FF, and $\Delta RT_{NDT}$ values were obtained from the surveillance Capsule W analysis (WCAP-14064[11]) using the ENDF/B-V scattering cross sections.

(d) Chemistry Factor = $\Sigma(FF^*\Delta RT_{NDT}) \div \Sigma(FF^2)$

(e) Adjusted $\Delta RT_{NDT}$ per Ratio Procedure of RG1.99R2 Position 2.1. Ratio = 3.0 (See Table 2). Actual ratio is 2.5 (68.0 ÷ 27.0 = 2.5), however, for conservatism a ratio of 3.0 was used in this case. As for adjustments due to temperature difference between Units 1 and 2, see Appendix C page C-5 for explanation.

Explanation of Margin Terms used for Byron Unit 1

When there are "two or more credible surveillance data sets"[1] available for Byron Unit 1, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for $\sigma_\Delta$ may be cut in half". Equation 4 from RG1.99R2 is as follows: $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The values of $\sigma_\Delta$ are referred to as "28°F for welds and 17°F for base metals."

Standard Deviation for Initial $RT_{NDT}$ Margin Term, $\sigma_I$

Since the initial $RT_{NDT}$ values are measured values in the case of Byron Unit 1, then $\sigma_I$ is taken to be 0°F

Standard Deviation for $\Delta RT_{NDT}$ Margin Term, $\sigma_\Delta$

Per RG1.99R2 Position 1.1, the values of $\sigma_\Delta$ are referred to as "28°F for welds and 17°F for base metal, except that $\sigma_\Delta$ need not exceed 0.50 times the mean value of $\Delta RT_{NDT}$." The "mean value of $\Delta RT_{NDT}$" is defined in RG1.99R2 by Equation 2 and defined herein by Equation 4. The chemistry factor in RG1.99R2 Equation 2 is calculated from Tables 1 and 2 of RG1.99R2.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, $\sigma_\Delta$ is taken to be the lesser of ½ $\Delta RT_{NDT}$ or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. $\Delta RT_{NDT}$ again is defined herein by Equation 4, while utilizing a "Best-Fit Chemistry Factor" calculated in accordance with Position 2.1 of RG1.99R2 and shown herein on Table 4.

Summary of the Margin Term

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

- Position 1.1: Lesser of $\Delta RT_{NDT}$ or 56°F for Welds
  Lesser of $\Delta RT_{NDT}$ or 34°F for Base Metal
- Position 2.1: Lesser of $\Delta RT_{NDT}$ or 28°F for Welds
  Lesser of $\Delta RT_{NDT}$ or 17°F for Base Metal

The following is a sample calculation of the margin term for the weld metal at the ¼ T location. The results for this calculation as well as the results for the remaining reactor vessel beltline materials are documented in Table 5.

Margin Term for Weld Metal (1/4T Location):

- From Equation 8 → $\Delta RT_{NDT}$ = CF x FF

where, CF = 68.0 (R.G. Position 1.1)
= 61.8 (R.G. Position 2.1; i.e. using Surv. Caps. Data)

FF = 0.799 (@ 12 EFPY and Fluence = $8.10 \times 10^{18}$ n/cm$^2$)

Therefore, $\Delta RT_{NDT}$ = 54.30 (R.G. Position 1.1)
= 49.40 (R.G. Position 2.1; i.e. using Surv. Caps. Data)

- From Equation 4 (of R.G. 1.99 R2) → $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$

where, ½ $\Delta RT_{NDT}$ = 27.15 (R.G. Position 1.1)
= 24.70 (R.G. Position 2.1; i.e. using Surv. Caps. Data)

$\sigma_I$ = 0°F (Initial $RT_{NDT}$ is Measured)

$\sigma_\Delta$ = Lesser of (½ $\Delta RT_{NDT}$) or (28°F)
= 27.15 (R.G. Position 1.1)

$\sigma_\Delta$ = Lesser of (½ $\Delta RT_{NDT}$) or (14°F)
= 14.00 (R.G. Position 2.1; i.e. using Surv. Caps. Data)

Therefore, $M = 2\sqrt{0^2 + 27.15^2}$ = 54.30 (R.G. Position 1.1)
$M = 2\sqrt{0^2 + 14.0^2}$ = 28.00 (R.G. Position 2.1; i.e. using Surv. Caps. Data)

## TABLE 5
### Calculation of Adjusted Reference Temperatures (ART) at 12 EFPY for all Byron Unit 1 Reactor Vessel Material
### (based on credible surveillance capsule data)

| Reactor Vessel Beltline Region Location | Material Identification | Cu% | Ni% | CF[e] | f @ 12[b] EFPY ($\times 10^{19}$) | ¼-t f[a] ¾-t f | ¼-t FF ¾-t FF | I | $\Delta RT_{NDT}$[d] | $\sigma_I$ | $\sigma_\Delta$ | M | ART[c] |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| ¼ T Calculation | | | | | | | | | | | | | |
| Intermediate Shell Forging | 5P-5933 | 0.0364 | 0.747 | 23.8 | 0.810 | 0.486 | 0.799 | 40 | 19.0 | 0 | 9.5 | 19.0 | 78 |
| Intermediate Shell Forging → using S/C Data | | | | 19.0 | 0.810 | 0.486 | 0.799 | 40 | 15.2 | 0 | 7.6 | 15.2 | **70** |
| Lower shell Forging | 5P-5951 | 0.04 | 0.64 | 26.0 | 0.810 | 0.486 | 0.799 | 10 | 20.8 | 0 | 10.4 | 20.8 | 52 |
| Girth Weld Metal | WF-336 | 0.06 | 0.61 | 68.0 | 0.810 | 0.486 | 0.799 | -30 | 54.3 | 0 | 27.15 | 54.3 | 79 |
| Girth Weld Metal → using S/C Data | | | | 61.8 | 0.810 | 0.486 | 0.799 | -30 | 49.4 | 0 | 14.0 | 28.0 | 47 |
| ¾ T Calculation | | | | | | | | | | | | | |
| Intermediate Shell Forging | 5P-5933 | 0.0364 | 0.70 | 23.8 | 0.810 | 0.175 | 0.538 | 40 | 12.8 | 0 | 6.4 | 12.8 | 66 |
| Intermediate Shell Forging → using S/C Data | | | | 19.0 | 0.810 | 0.175 | 0.538 | 40 | 10.2 | 0 | 5.1 | 10.2 | **60** |
| Lower shell Forging | 5P-5951 | 0.04 | 0.64 | 26.0 | 0.810 | 0.175 | 0.538 | 10 | 14.0 | 0 | 7.0 | 14.0 | 38 |
| Girth Weld Metal | WF-336 | 0.06 | 0.61 | 68.0 | 0.810 | 0.175 | 0.538 | -30 | 36.6 | 0 | 18.3 | 36.6 | 43 |
| Girth Weld Metal → using S/C Data | | | | 61.8 | 0.810 | 0.175 | 0.538 | -30 | 33.2 | 0 | 14.0 | 28.0 | 31 |

NOTES:

(a) The Byron Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.

(b) Fluence, f, is based upon $f_{surf}$ ($10^{19}$ n/cm$^2$, E>1.0 MeV) = 0.810 at 12 EFPY.

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

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

(e) The CF is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).

TABLE 6
Summary of Adjusted Reference Temperatures (ART) at 1/4T and 3/4T Locations for 12 EFPY

| Material | 12 EFPY | |
|---|---|---|
| | 1/4T ART | 3/4T ART |
| Intermediate Shell Forging 5P-5933 (RG Position 1[a]) | 78 | 66 |
| using credible surveillance capsule data (RG Position 2[a]) | 70[b] | 60[b] |
| Lower Shell Forging 5P-5951 (RG Position 1[a]) | 52 | 38 |
| Circumferential Weld WF-336 (RG Position 1[a]) | 79 | 43 |
| using credible surveillance capsule data (RG Position 2[a]) | 47 | 31 |

NOTES:

(a) Calculated using a chemistry factor based on Regulatory Guide (RG) 1.99, Revision 2, Positions 1 and 2.

(b) These ART values were used to generate the Byron Unit 1 heatup and cooldown curves.

# 5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods[12] discussed in Section 3 and 4 of this report. The 1989 edition methodology is also presented in WCAP-14040-NP-A[13], dated January 1996.

Since indication of reactor vessel beltline pressure is not available on the plant, the pressure difference between the wide-range pressure transmitter and the limiting beltline region must be accounted for when using pressure-temperature limit curves presented in Figures 1 and 2. Generic calculations (based upon four active loops and one operating RHR pump) have determined that the pressure indicated by the reactor coolant system wide-range instrumentation should be assumed to be 74 psig less than that at the reactor vessel beltline for Byron Unit 1[15]. Figures 3 and 4 do include this pressure difference of 74 psig.

Figures 1 and 3 present the heatup curves without margins for instrumentation errors using a heatup rate of 100°F/hr applicable for the first 12 EFPY. Figures 2 and 4 present the cooldown curves without margins for instrumentation errors using cooldown rates up to 100°F/hr applicable for the first 12 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 through 4. This is in addition to other criteria which must be met before the reactor is made critical.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 through 4. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code as follows:

$$1.5K_{Im} < K_{Ia} \qquad (6)$$

where,

$K_{Im}$ is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ia} = 26.78 + 1.233\ e^{[0.0145\ (T\ -\ RTNDT\ +\ 160)]}$,
T is the minimum permissible metal temperature, and
$RT_{NDT}$ is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 5. The pressure-temperature limits or core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperature for the inservice hydrostatic leak tests for the Byron Unit 1 reactor vessel at 12 EFPY is 203°F at 2485 psig. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 4 define all of the above limits for ensuring prevention of nonductile failure for the Byron Unit 1 reactor vessel. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 4 are presented in Tables 6 and 7.

Additionally, Westinghouse Engineering has reviewed the minimum boltup temperature requirements for the Byron Unit 1 reactor pressure vessel. According to Paragraph G-2222 of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, the reactor vessel may be bolted up and pressurized to 20 percent of the initial hydrostatic test pressure at the initial $RT_{NDT}$ of the material stressed by the boltup. Therefore, since the most limiting initial $RT_{NDT}$ value is 60°F (closure head flange), the reactor vessel can be bolted up at this temperature.

MATERIAL PROPERTY BASIS

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

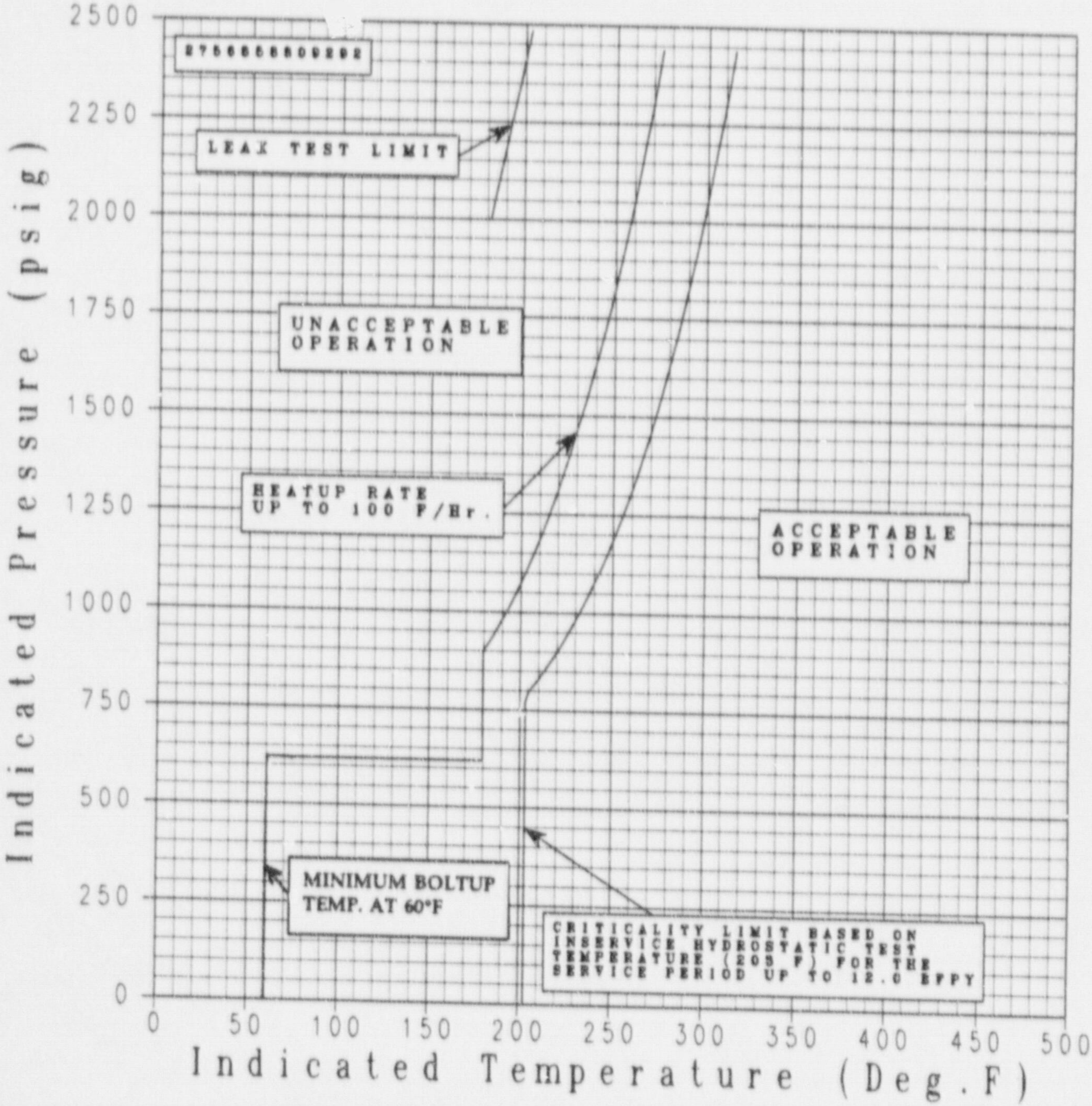


FIGURE 1 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

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

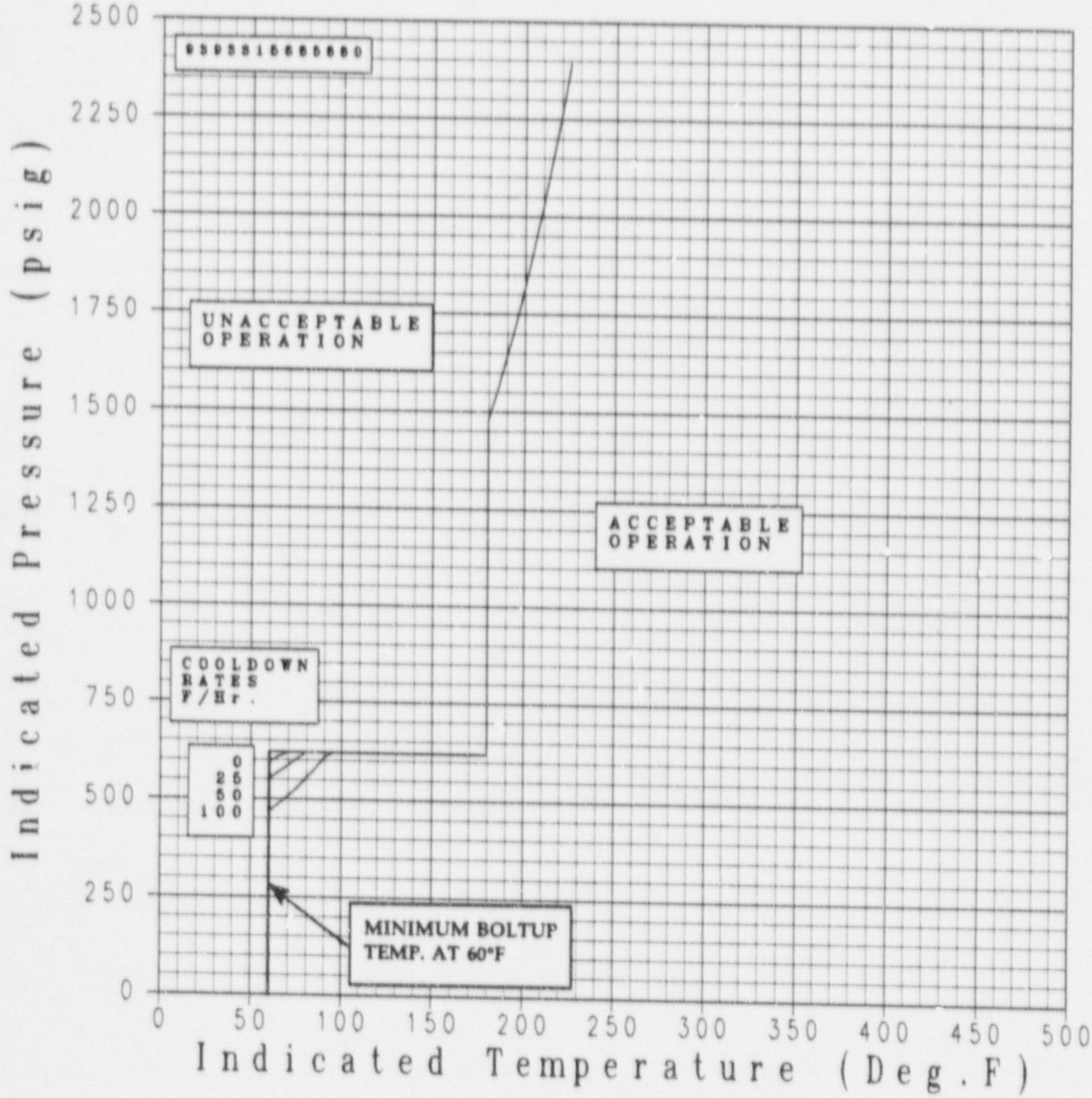


FIGURE 2 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

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

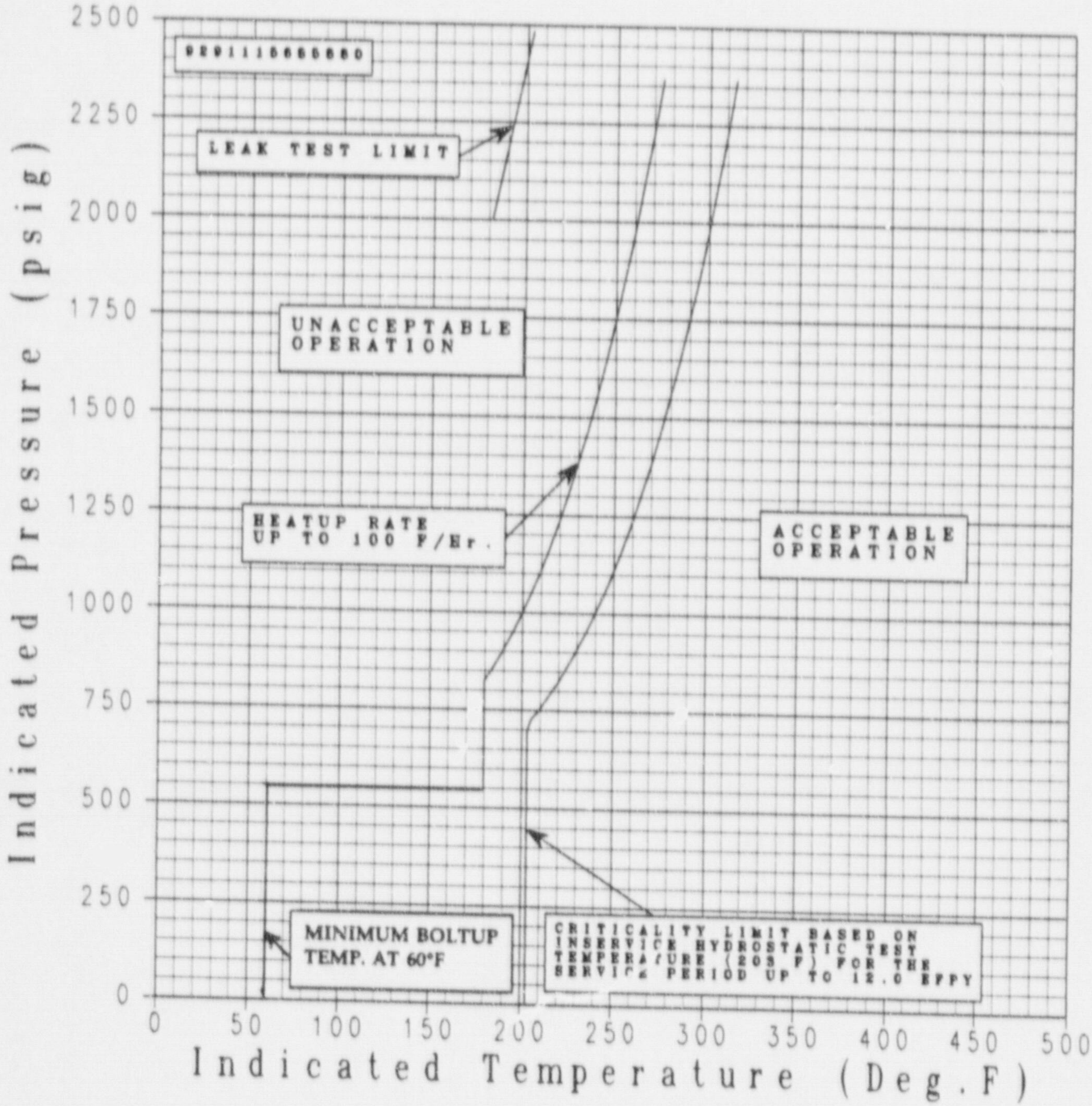


FIGURE 3 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors; Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and the Reactor Vessel Beltline Region)

MATERIAL PROPERTY BASIS

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

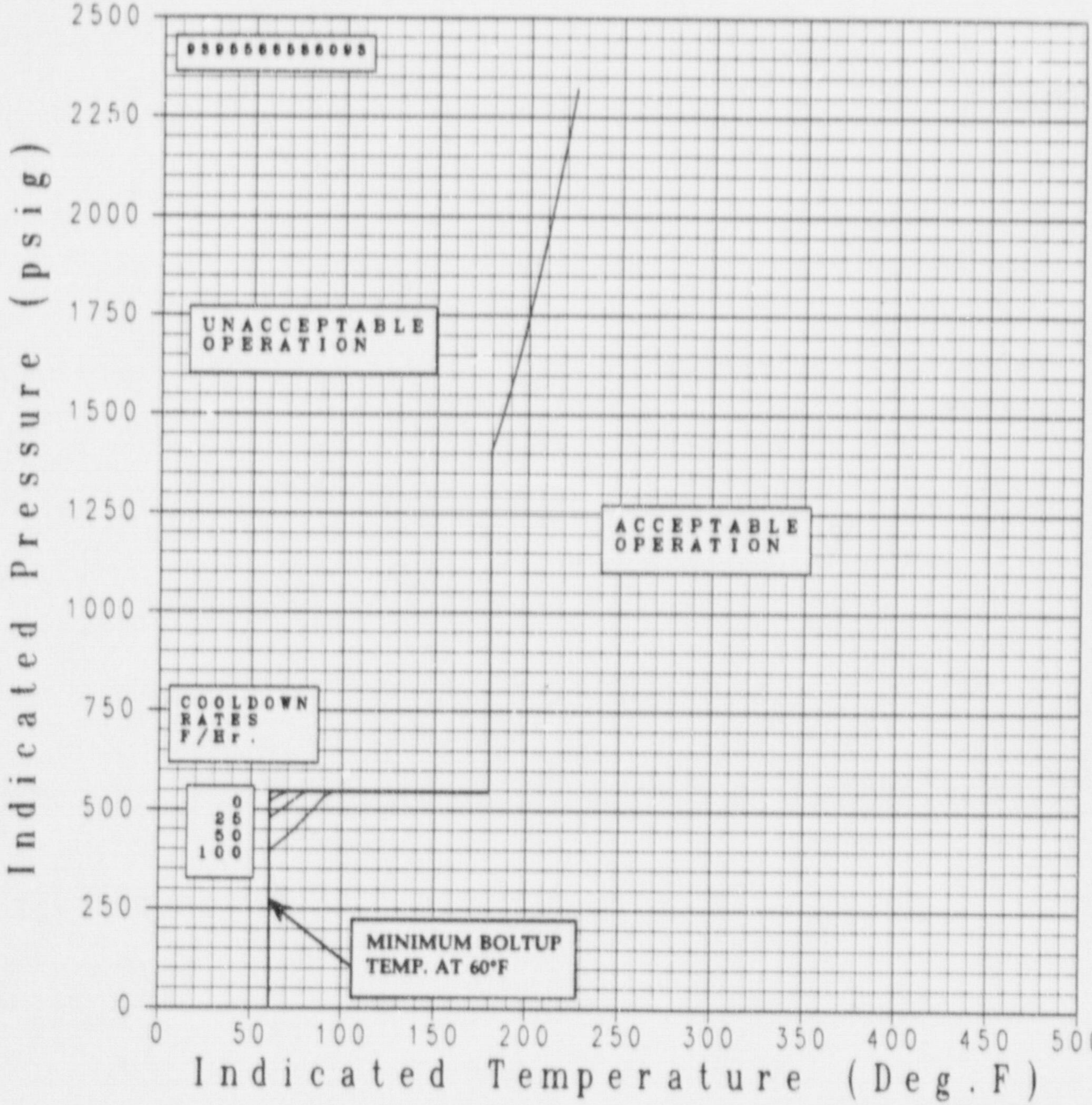


FIGURE 4 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors; Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and the Reactor Vessel Beltline Region)

# TABLE 7
## Byron Unit 1 Heatup and Cooldown Data at 12 EFPY Without Margins for Instrumentation Errors

Includes 1) Vessel flange requirements of 180°F and 621 psig per 10CFR50.

| Cooldown Curves Steady State | | 25F | | 50F | | 100F | | Heatup Curve 100F | | Criticality. Limit | | Leak Test Limit | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| T | P | T | P | T | P | T | P | T | P | T | P | T | P |
| 60 | 621 | 60 | 595 | 60 | 554 | 60 | 470 | 60 | 621 | 203 | 0 | 182 | 2000 |
| 65 | 621 | 65 | 610 | 65 | 570 | 65 | 489 | 65 | 621 | 203 | 0 | 203 | 2485 |
| 70 | 621 | 70 | 621 | 70 | 587 | 70 | 509 | 70 | 621 | 203 | 0 | | |
| 75 | 621 | 75 | 621 | 75 | 605 | 75 | 531 | 75 | 621 | 203 | 0 | | |
| 80 | 621 | 80 | 621 | 80 | 621 | 80 | 554 | 80 | 621 | 203 | 671 | | |
| 85 | 621 | 85 | 621 | 85 | 621 | 85 | 579 | 85 | 621 | 203 | 657 | | |
| 90 | 621 | 90 | 621 | 90 | 621 | 90 | 607 | 90 | 621 | 203 | 646 | | |
| 95 | 621 | 95 | 621 | 95 | 621 | 95 | 621 | 95 | 621 | 203 | 639 | | |
| 100 | 621 | 100 | 621 | 100 | 621 | 100 | 621 | 100 | 621 | 203 | 634 | | |
| 105 | 621 | 105 | 621 | 105 | 621 | 105 | 621 | 105 | 621 | 203 | 632 | | |
| 110 | 621 | 110 | 621 | 110 | 621 | 110 | 621 | 110 | 621 | 203 | 633 | | |
| 115 | 621 | 115 | 621 | 115 | 621 | 115 | 621 | 115 | 621 | 203 | 637 | | |
| 120 | 621 | 120 | 621 | 120 | 621 | 120 | 621 | 120 | 621 | 203 | 642 | | |
| 125 | 621 | 125 | 621 | 125 | 621 | 125 | 621 | 125 | 621 | 203 | 651 | | |
| 130 | 621 | 130 | 621 | 130 | 621 | 130 | 621 | 130 | 621 | 203 | 661 | | |
| 135 | 621 | 135 | 621 | 135 | 621 | 135 | 621 | 135 | 621 | 203 | 674 | | |
| 140 | 621 | 140 | 621 | 140 | 621 | 140 | 621 | 140 | 621 | 203 | 689 | | |
| 145 | 621 | 145 | 621 | 145 | 621 | | | 145 | 621 | 203 | 707 | | |
| 150 | 621 | 150 | 621 | | | | | 150 | 621 | 203 | 727 | | |
| 155 | 621 | | | | | | | 155 | 621 | 203 | 749 | | |
| 160 | 621 | | | | | | | 160 | 621 | 203 | 774 | | |
| 165 | 621 | | | | | | | 165 | 621 | 205 | 801 | | |
| 170 | 621 | | | | | | | 170 | 621 | 210 | 831 | | |
| 175 | 621 | | | | | | | 175 | 621 | 215 | 864 | | |
| 180 | 621 | | | | | | | 180 | 621 | 220 | 900 | | |
| 180 | 1483 | | | | | | | 180 | 900 | 225 | 938 | | |
| 185 | 1559 | | | | | | | 185 | 938 | 230 | 980 | | |
| 190 | 1640 | | | | | | | 190 | 980 | 235 | 1026 | | |
| 195 | 1728 | | | | | | | 195 | 1026 | 240 | 1075 | | |
| 200 | 1821 | | | | | | | 200 | 1075 | 245 | 1128 | | |
| 205 | 1921 | | | | | | | 205 | 1128 | 250 | 1186 | | |
| 210 | 2029 | | | | | | | 210 | 1186 | 255 | 1247 | | |
| 215 | 2143 | | | | | | | 215 | 1247 | 260 | 1313 | | |
| 220 | 2266 | | | | | | | 220 | 1313 | 265 | 1385 | | |
| 225 | 2397 | | | | | | | 225 | 1385 | 270 | 1461 | | |
| | | | | | | | | 230 | 1461 | 275 | 1543 | | |
| | | | | | | | | 235 | 1543 | 280 | 1630 | | |
| | | | | | | | | 240 | 1630 | 285 | 1724 | | |
| | | | | | | | | 245 | 1724 | 290 | 1825 | | |
| | | | | | | | | 250 | 1825 | 295 | 1933 | | |
| | | | | | | | | 255 | 1933 | 300 | 2048 | | |
| | | | | | | | | 260 | 2048 | 305 | 2171 | | |
| | | | | | | | | 265 | 2171 | 310 | 2302 | | |
| | | | | | | | | 270 | 2302 | 315 | 2441 | | |
| | | | | | | | | 275 | 2441 | | | | |

(Configuration #9393315685880 for Cooldown, #2756858809292 for Heatup)

# TABLE 8
## Byron Unit 1 Heatup and Cooldown Data at 12 EFPY Without Margins for Instrumentation Errors

Includes 1) Vessel flange requirements of 180°F and 621 psig per 10CFR50, and 2) Pressure adjustment of 74 psig to account for pressure difference between the wide-range pressure transmitter and the limiting beltline region of the reactor vessel.

| Cooldown Curves | | | | | | | | Heatup Curve | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 25F | | 50F | | 100F | | 100F | | Criticality Limit | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P | T | P | T | P |
| 60 | 547 | 60 | 521 | 60 | 480 | 60 | 396 | 60 | 547 | 203 | 0 | 182 | 2000 |
| 65 | 547 | 65 | 536 | 65 | 496 | 65 | 415 | 65 | 547 | 203 | 0 | 203 | 2485 |
| 70 | 547 | 70 | 547 | 70 | 513 | 70 | 435 | 70 | 547 | 203 | 0 | | |
| 75 | 547 | 75 | 547 | 75 | 531 | 75 | 457 | 75 | 547 | 203 | 0 | | |
| 80 | 547 | 80 | 547 | 80 | 547 | 80 | 480 | 80 | 547 | 203 | 597 | | |
| 85 | 547 | 85 | 547 | 85 | 547 | 85 | 505 | 85 | 547 | 203 | 583 | | |
| 90 | 547 | 90 | 547 | 90 | 547 | 90 | 523 | 90 | 547 | 203 | 572 | | |
| 95 | 547 | 95 | 547 | 95 | 547 | 95 | 547 | 95 | 547 | 203 | 565 | | |
| 100 | 547 | 100 | 547 | 100 | 547 | 100 | 547 | 100 | 547 | 203 | 560 | | |
| 105 | 547 | 105 | 547 | 105 | 547 | 105 | 547 | 105 | 547 | 203 | 558 | | |
| 110 | 547 | 110 | 547 | 110 | 547 | 110 | 547 | 110 | 547 | 203 | 559 | | |
| 115 | 547 | 115 | 547 | 115 | 547 | 115 | 547 | 115 | 547 | 203 | 563 | | |
| 120 | 547 | 120 | 547 | 120 | 547 | 120 | 547 | 120 | 547 | 203 | 568 | | |
| 125 | 547 | 125 | 547 | 125 | 547 | 125 | 547 | 125 | 547 | 203 | 577 | | |
| 130 | 547 | 130 | 547 | 130 | 547 | 130 | 547 | 130 | 547 | 203 | 587 | | |
| 135 | 547 | 135 | 547 | 135 | 547 | 135 | 547 | 135 | 547 | 203 | 600 | | |
| 140 | 547 | 140 | 547 | 140 | 547 | 140 | 547 | 140 | 547 | 203 | 615 | | |
| 145 | 547 | 145 | 547 | 145 | 547 | | | 145 | 547 | 203 | 633 | | |
| 150 | 547 | 150 | 547 | | | | | 150 | 547 | 203 | 653 | | |
| 155 | 547 | | | | | | | 155 | 547 | 203 | 675 | | |
| 160 | 547 | | | | | | | 160 | 547 | 203 | 700 | | |
| 165 | 547 | | | | | | | 165 | 547 | 205 | 727 | | |
| 170 | 547 | | | | | | | 170 | 547 | 210 | 757 | | |
| 175 | 547 | | | | | | | 175 | 547 | 215 | 790 | | |
| 180 | 547 | | | | | | | 180 | 547 | 220 | 826 | | |
| 180 | 1409 | | | | | | | 180 | 826 | 225 | 864 | | |
| 185 | 1485 | | | | | | | 185 | 864 | 230 | 906 | | |
| 190 | 1566 | | | | | | | 190 | 906 | 235 | 952 | | |
| 195 | 1654 | | | | | | | 195 | 952 | 240 | 1001 | | |
| 200 | 1747 | | | | | | | 200 | 1001 | 245 | 1054 | | |
| 205 | 1847 | | | | | | | 205 | 1054 | 250 | 1112 | | |
| 210 | 1955 | | | | | | | 210 | 1112 | 255 | 1173 | | |
| 215 | 2069 | | | | | | | 215 | 1173 | 260 | 1239 | | |
| 220 | 2192 | | | | | | | 220 | 1239 | 265 | 1311 | | |
| 225 | 2323 | | | | | | | 225 | 1311 | 270 | 1387 | | |
| | | | | | | | | 230 | 1387 | 275 | 1469 | | |
| | | | | | | | | 235 | 1469 | 280 | 1556 | | |
| | | | | | | | | 240 | 1556 | 285 | 1650 | | |
| | | | | | | | | 245 | 1650 | 290 | 1751 | | |
| | | | | | | | | 250 | 1751 | 295 | 1859 | | |
| | | | | | | | | 255 | 1859 | 300 | 1974 | | |
| | | | | | | | | 260 | 1974 | 305 | 2097 | | |
| | | | | | | | | 265 | 2097 | 310 | 2228 | | |
| | | | | | | | | 270 | 2228 | 315 | 2367 | | |

(Configuration #9395568588093 for Cooldown, #9291115685880 for Heatup)

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# APPENDIX A

# WELD METAL INTEGRATION FOR BYRON UNITS 1 AND 2

**INTRODUCTION:**

Westinghouse performed an evaluation to determine if the weld wire data of the Byron Units 1 and 2 surveillance programs can be integrated. The evaluation was based on the following criteria:

1. What weld wire heat number, flux, and flux lot were used to fabricate the surveillance program weld metal of each unit,
2. What vendor fabricated the welds and in what time frame,
3. What heat treatment did each surveillance program weld receive,
4. Is the initial $RT_{NDT}$ of the welds the same or relatively close,
5. Is the initial upper shelf energy of the welds the same or relatively close,
6. Is the geometry of the plants the same,
7. Is the type of fuel in all plants the same,
8. Are the fuel loading patterns in the plants similar (i.e., low leakage, etc.),
9. What is the projected 32 effective full power year surface fluence of each plant,
10. What vessel inlet temperatures do the plants operate at,
11. What are the differences in the capsule lead factors of the plants,
12. Can the criteria for credibility in 10 CFR Part 50.61 be met for each plant?

**EVALUATION:**

1. *What weld wire heat number, flux and flux lot numbers were used to fabricate the welds?*
   The surveillance program weld metal for each unit was fabricated with the following weld wire and flux:
   Byron 1: The weld metal is type Linde MnMoNi, heat number 442002, with a Linde 80 type flux, lot number 8873. This is the same heat number used in the limiting beltline weld (seam WF-336).
   Byron 2: The weld metal is type Linde MnMoNi, heat number 442002, with a Linde 80 type flux, lot number 8064. This is the same heat number used in the limiting beltline weld (seam WF-447).

   The Byron Units 1 and 2 surveillance program weld metals were fabricated with the same heat of weld wire and the same type of flux. Therefore, this information supports the integration of the surveillance program test results for these welds.

2. *What vendor fabricated the welds and in what time frame ?*
   Byron 1: B&W fabricated the welds in the mid. 1970's
   Byron 2: B&W fabricated the welds in the mid. 1970's

   The Byron Units 1 and 2 surveillance program weld metals were fabricated in the same time frame and by the same vendor. Therefore, this information supports the integration of the surveillance program test results for these welds.

3. *What heat treatment did each weld receive?*
   The surveillance program weld metals received the following post-weld stress relief heat treatments:
   Byron 1: 1125 ± 25°F for 12 hours and 16 minutes; furnace-cooled
   Byron 2: 1150 ± 50°F for 13.5 hours; furnace-cooled

   The post-weld stress relief heat treatment given to the Byron 1 and 2 surveillance program welds was slightly different. However, based on engineering judgement, the slight differences in temperature and time should not cause a significant difference in the material toughness properties.

4. *Is the initial $RT_{NDT}$ of the welds the same or relatively close?*
   Byron 1: -30 °F
   Byron 2: 10 °F

   Based on the data specific to the Byron 1 and Byron 2 vessel beltline welds (WF-336 and WF-447, respectively, with the same weld wire heat and different flux lots), the initial $RT_{NDT}$ of the welds differ. However, the surveillance materials have performed similarly under irradiation, and it is irradiation shift data that is used in the integration of data. As can be seen in Table 4 (page 11 of this report), the measured shifts in $RT_{NDT}$ are relatively the same. For example, the shift for the first capsules from Byron 1 and Byron 2 is 0°F. For the second capsules removed from Byron Units 1 and 2, the measured shifts are equal to 30°F and 35°F, respectively. These results are very close. Therefore, this information supports the integration of the surveillance program test results for these welds.

5. *Is the initial upper shelf energy of the surveillance welds the same or relatively close?*

| | |
|---|---|
| Byron 1: | 74 ft-lb |
| Byron 2: | 67 ft-lb |

The initial upper shelf energy values for the surveillance weld materials in the Byron surveillance programs are very similar. Therefore, this information supports the integration of the surveillance program test results for these welds.

6. *Is the geometry of the plants the same?*
Byron Units 1 and 2 have a reactor vessel inner diameter of 173 inches, a reactor vessel beltline thickness of 8.5 inches (excluding the cladding). Both have a power rating of 3411 MWt and are Westinghouse 4-loop NSSS plants. Both vessels have neutron pads and the surveillance capsules are located at the same azimuthal angles.

7. *Is the fuel design in all plants the same?*
Byron 1 & 2 use 17X17 rod array fuel assemblies with the same fuel design, thus producing similar radiation effects at the surveillance capsules.

8. *Are the fuel loading patterns in the plants similar (i.e. low leakage, etc.)?*
Byron 1 & 2 use a low leakage loading pattern.

9. *What is the projected 32 effective full power year surface fluence of each plant?*
Based on the information provided below, the projected vessel surface fluence values (E>1.0 MeV) at 32 EFPY for Byron Unit 1 are essentially the same as Byron Unit 2.

Byron Unit 1

| 0° | 15° | 25° | 35° | 45° |
|---|---|---|---|---|
| $1.290 \times 10^{19}$ | $1.947 \times 10^{19}$ | $2.159 \times 10^{19}$ | $1.705 \times 10^{19}$ | $1.939 \times 10^{19}$ |

Byron Unit 2

| 0° | 15° | 25° | 35° | 45° |
|---|---|---|---|---|
| $1.353 \times 10^{19}$ | $1.979 \times 10^{19}$ | $2.192 \times 10^{19}$ | $1.772 \times 10^{19}$ | $2.026 \times 10^{19}$ |

10. *What are the vessel inlet temperatures?* (Per Reference 24)

| CYCLE # | Byron Unit 1 Tcold (°F) | Byron Unit 2 Tcold (°F) |
|---|---|---|
| 1 | 557 (Cap. U)* | 557 (Cap. U)* |
| 2 | 557 | 551 |
| 3 | 551 | 551 |
| 4 | 551 | 551 (Cap. W)* |
| 5 | 551 (Cap. X)* | 551 |
| 6 | 551 | 551 |
| 7 | 551 | 551 |
| 8 | 551 | - - |

* See explanation in Appendix C, page C-5.

11. *What are the differences in the capsule lead factors of the plants?*
Based on the information provide in Table 1, the lead factors of the surveillance capsules in Byron Unit 1 are essentially the same as Byron Unit 2.

TABLE A-1
Surveillance Capsule Lead Factors for Byron Units 1 & 2

| Byron Unit 1 | | | Byron Unit 2 | | |
|---|---|---|---|---|---|
| Capsule | Location | Lead Factor | Capsule | Location | Lead Factor |
| U | 58.5° | 3.85 | U | 58.5° | 3.96 |
| X | 238.5° | 3.79 | W | 121.5° | 3.89 |
| V | 61.0° | 3.59 | V | 61.0° | 3.64 |
| Y | 241.0° | 3.59 | Y | 241.0° | 3.64 |
| W | 121.5° | 3.79 | X | 238.5° | 3.89 |
| Z | 301.5° | 3.79 | Z | 301.5° | 3.89 |

Based on the projected vessel surface fluence and lead factor values for Byron 1 and 2, the Byron 1 and 2 surveillance capsules will have approximately the same flux rates and irradiation temperatures. This supports the use of the weld results from both programs to evaluate the reactor vessel integrity of both units.

12. *Can the criteria for credibility in 10 CFR Part 50.61 be met for each plant?*
Credibility will be evaluated for all the surveillance capsule data (base metal & weld metal) for Byron Units 1 and 2.

***Criterion 1:*** ***The materials in the surveillance capsules must be those which are controlling materials with regard to radiation embrittlement.***

The following is a list of the beltline materials contained in the Byron Units 1 and 2 surveillance programs:

Byron Unit 1: Intermediate shell forging 5P-5933
Circumferential weld seam WF-336, heat number 442002, with a Linde 80 type flux, lot number 8873. (This is the same heat number used in the limiting beltline weld.)

Byron Unit 2: Lower shell forging 49D330/49C298-1-1
Circumferential weld seam WF-447, heat number 442002, with a Linde 80 type flux, lot number 8064. (This is the same heat number used in the limiting beltline weld.)

Based on the information provided in the material selection documents, WCAP-9517 (Byron 1, See Ref. 3) and WCAP-10398 (Byron 2, See Ref. 4), these materials are judged to be the most controlling with regard to radiation embrittlement for each unit. Therefore, Criteria #1 is met for both units.

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

Plots of Charpy energy versus temperature for the unirradiated condition are presented in WCAP-9517, "Commonwealth Edison Co. Byron Station Unit 1 Reactor Vessel Radiation Surveillance Program," dated July 1979 and WCAP-10398, "Commonwealth Edison Co. Byron Station Unit 2 Reactor Vessel Radiation Surveillance Program," dated December 1983. Plots of Charpy energy versus temperature for the irradiated conditions are presented in the WCAP reports for Capsules U & X (Unit 1) and U & W (Unit 2).

Based on engineering judgement, the scatter in the data presented in these reports is small enough to determine the 30 ft-lb temperature and the upper shelf energy of the Byron Units 1 & 2 surveillance weld metals unambiguously. Therefore, the Byron Units 1 & 2 surveillance materials meet this criteria.

*Criterion 3:* ***Where there are two or more sets of surveillance data from one reactor, the scatter of $\Delta RT_{NDT}$ values must be less than 28°F for welds and 17°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.***

The least squares method, as described in Regulatory Position 2.1, will be utilized in determining a best-fit line for this data to determine if this criteria is met. It should be noted here that the ratio procedure is not applied in this instance since only surveillance capsule data is being evaluated.

TABLE A-2[c]

Byron Units 1 & 2 Surveillance Capsule Data Chemistry Factor for Best Fit Line

| Material | Capsule | f[a] | FF[b] | Measured $\Delta RT_{NDT}$ | FF x $\Delta RT_{NDT}$ | $FF^2$ |
|---|---|---|---|---|---|---|
| Byron Unit 1 Inter. Shell Forging 5P-5933 (Axial) | U | $3.72 \times 10^{18}$ | 0.727 | 0 | 0 | 0.529 |
| | X | $1.39 \times 10^{19}$ | 1.091 | 30 | 32.73 | 1.190 |
| Byron Unit 1 Inter. Shell Forging 5P-5933 (Tangential) | U | $3.72 \times 10^{18}$ | 0.727 | 0 | 0 | 0.529 |
| | X | $1.39 \times 10^{19}$ | 1.091 | 30 | 32.73 | 1.190 |
| | | | | Sum: | 65.46 | 3.44 |
| | Chemistry Factor = 65.46 ÷ 3.44 = **19.0°F** | | | | | |
| Byron Unit 2 Lower Shell Forging 49D330-1/49C298-1 (Axial) | U | $3.996 \times 10^{18}$ | 0.746 | 0 | 0 | 0.557 |
| | W | $1.211 \times 10^{19}$ | 1.053 | 5 | 5.27 | 1.109 |
| Byron Unit 2 Lower Shell Forging 49D330-1/49C298-1 (Tangential) | U | $3.996 \times 10^{18}$ | 0.746 | 25 | 18.65 | 0.557 |
| | W | $1.211 \times 10^{19}$ | 1.053 | 40 | 42.12 | 1.109 |
| | | | | Sum: | 66.04 | 3.332 |
| | Chemistry Factor = 66.04 ÷ 3.332 = **19.8°F** | | | | | |
| Byron Unit 1 Weld Metal WF-336 | U | $3.72 \times 10^{18}$ | 0.727 | 0 | 0.00 | 0.529 |
| | X | $1.39 \times 10^{19}$ | 1.091 | 35[d] | 38.185 | 1.19 |
| Byron Unit 2 Weld Metal WF-447 | U | $3.996 \times 10^{18}$ | 0.746 | 0 | 0.00 | 0.557 |
| | W | $1.211 \times 10^{19}$ | 1.053 | 30[d] | 31.600 | 1.110 |
| | | | | Sum: | 69.785 | 3.386 |
| | Chemistry Factor = 69.785 ÷ 3.386 = **20.6°F** | | | | | |

NOTES:

(a) f = Fluence ($10^{19}$ n/cm$^2$, E > 1.0 MeV)

(b) FF = Fluence Factor = $f^{(0.28 - 0.1 * \log f)}$

(c) Values of *f* and $\Delta RT_{NDT}$ for Byron 1 were taken from WCAP-14044 and WCAP-13880, respectively. The Byron Unit 2 values were taken from Table 3 of WCAP-14063.

(d) See Appendix C, page C-5, for explanation of temperature adjustment.

TABLE A-3

Best Fit Evaluation for Byron 1 & 2 Surveillance Materials

| Base Material | CF | FF | $\Delta RT_{NDT}$ (30 ft-lb) (°F) | Best Fit[a] $\Delta RT_{NDT}$ (°F) | Scatter of $\Delta RT_{NDT}$ (°F) | < 17°F (Base Metals) < 28°F (Weld Metal) |
|---|---|---|---|---|---|---|
| Byron 1 Weld Metal | 20.6 | 0.727 | 0 | 15.0 | -15.0 | Yes |
| | 20.6 | 1.091 | 35 | 22.5 | 12.5 | Yes |
| Byron 2 Weld Metal | 20.6 | 0.746 | 0 | 15.4 | -15.4 | Yes |
| | 20.6 | 1.053 | 30 | 21.7 | 8.3 | Yes |
| Byron Unit 1 Inter. Shell Forging 5P-5933 (Axial) | 19.0 | 0.727 | 0 | 13.8 | -13.8 | Yes |
| | 19.0 | 1.091 | 30 | 20.7 | 9.3 | Yes |
| Byron Unit 1 Inter. Shell Forging 5P-5933 (Tangential) | 19.0 | 0.727 | 0 | 13.8 | -13.8 | Yes |
| | 19.0 | 1.091 | 30 | 20.7 | 9.3 | Yes |
| Byron Unit 2 Lower Shell Forging 49D330-1/49C298-1 (Axial) | 19.8 | 0.746 | 0 | 14.8 | -14.8 | Yes |
| | 19.8 | 1.053 | 5 | 20.8 | -15.8 | Yes |
| Byron Unit 2 Lower Shell Forging 49D330-1/49C298-1 (Tangential) | 19.8 | 0.746 | 25 | 14.8 | 10.2 | Yes |
| | 19.8 | 1.053 | 40 | 20.8 | 19.2 | No |

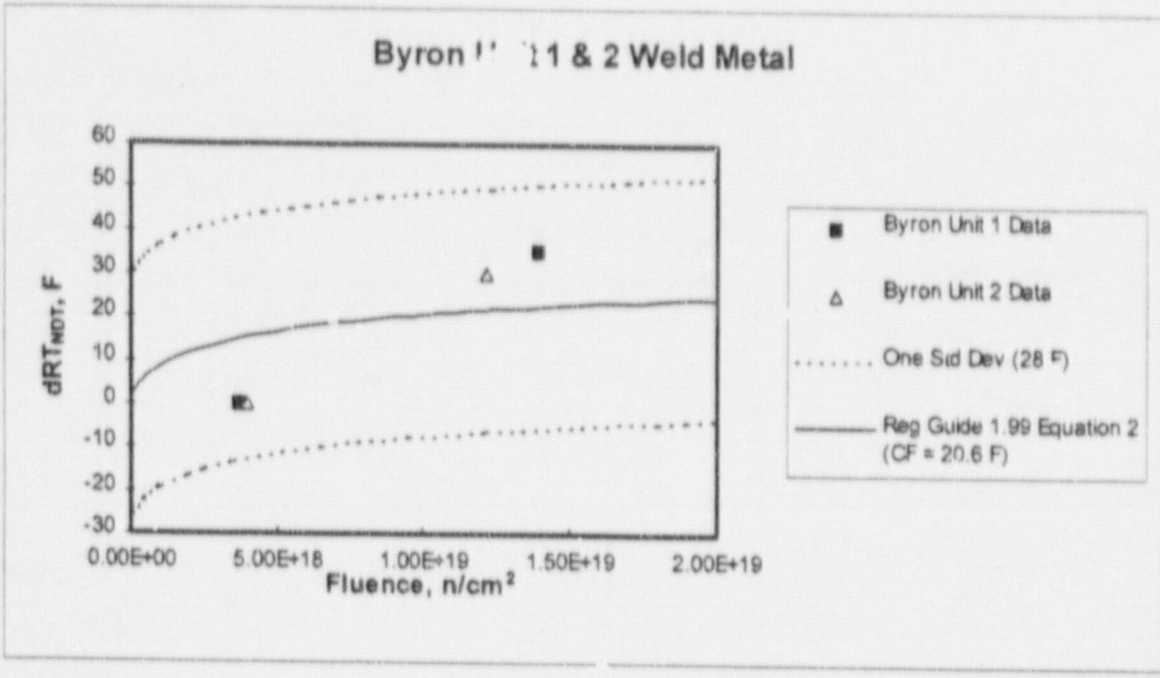
NOTES:

(a) Best Fit Line Per Equation 2 of Reg. Guide 1.99 Rev. 2 Position 1.1.

Weld Metal:
The scatter of $\Delta RT_{NDT}$ values (See Figure A-1) about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 28°F for weld metal. As shown above, the error is within 28°F of the best-fit line. Therefore, this criteria is met for the Byron Units 1 & 2 surveillance weld material.

Base Material:
The scatter of $\Delta RT_{NDT}$ values (See Figure A-1) about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 17°F for base metal. As shown in Table A-3, the error for Byron Unit 1 is within 17°F of the best-fit line and the error for one point for Byron Unit 2 is not within 17°F of the best-fit line. Therefore, this criteria is met for Byron Unit 1 base metal but not for Byron Unit 2 base metal. As a result of the Byron Unit 2 base metal exceeding this criteria, the margin term that is calculated for the Byron Unit 2 base metal using Position 2.1 of the Reg. Guide should now be doubled. Thus allowing this data to be used in the evaluations of pressure-temperature limit curves and PTS.

FIGURE A-1


Byron U 1 & 2 Weld Metal
dRT$_{NDT}$, F
Fluence, n/cm$^2$
60
50
40
30
20
10
0
-10
-20
-30
0.00E+00
5.00E+18
1.00E+19
1.50E+19
2.00E+19
Byron Unit 1 Data
Byron Unit 2 Data
One Std Dev (28 F)
Reg Guide 1.99 Equation 2
(CF = 20.6 F)

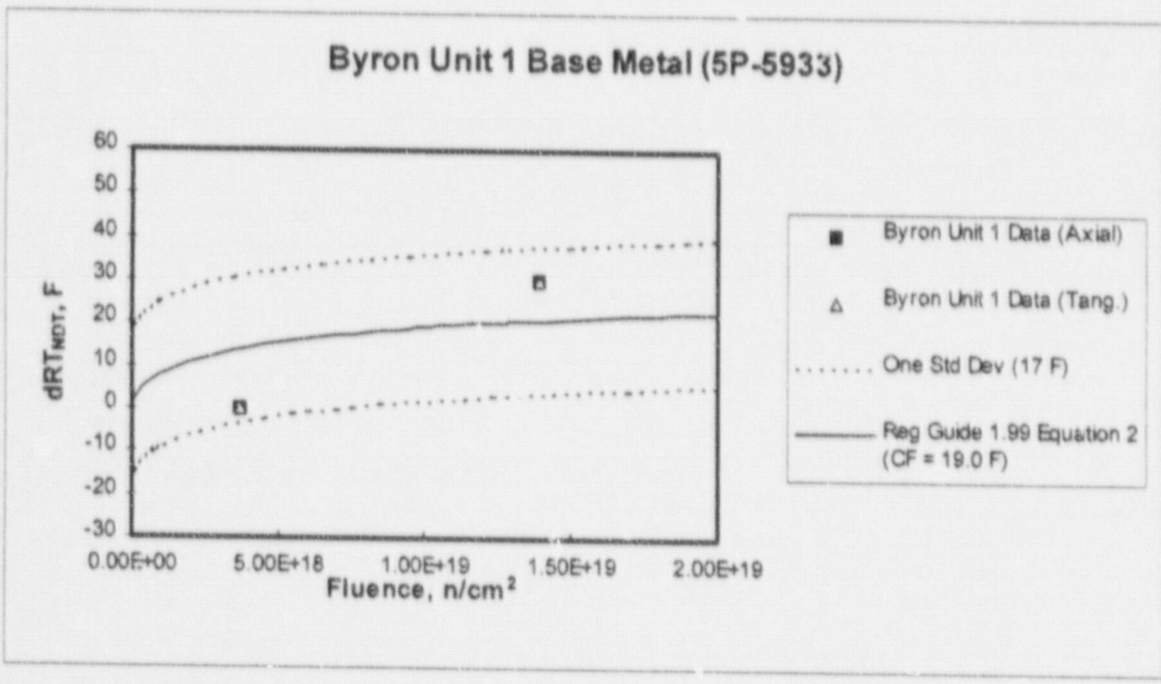
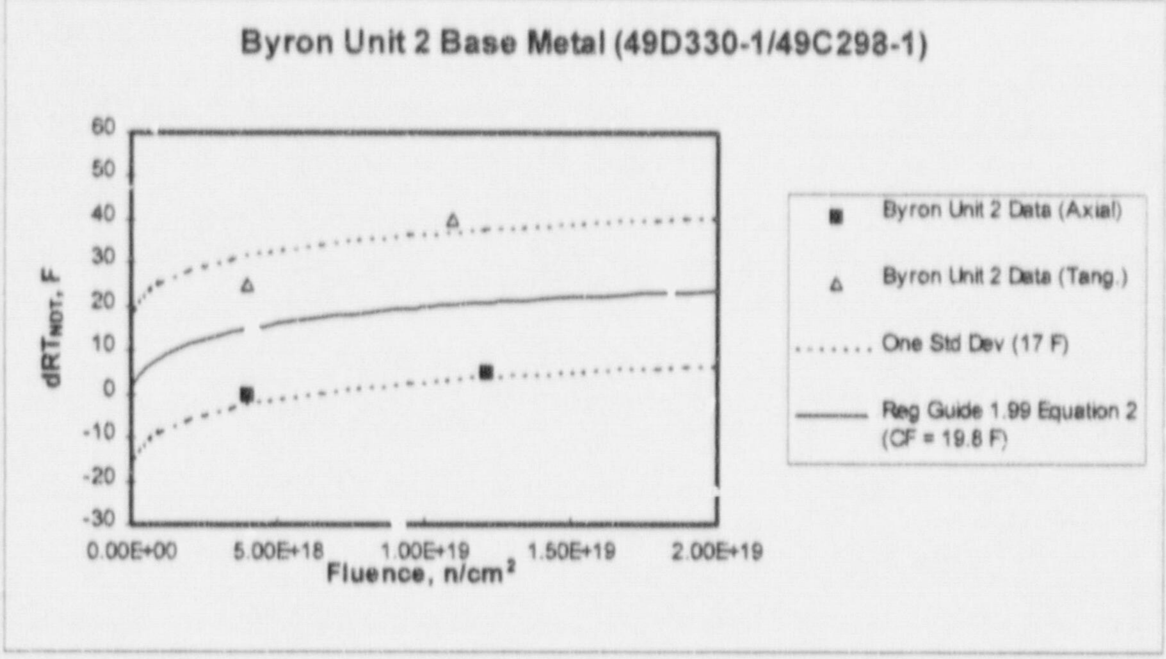

Byron Unit 1 Base Metal (5P-5933)
dRT$_{NDT}$, F
Fluence, n/cm$^2$
60
50
40
30
20
10
0
-10
-20
-30
0.00E+00
5.00E+18
1.00E+19
1.50E+19
2.00E+19
Byron Unit 1 Data (Axial)
Byron Unit 1 Data (Tang.)
One Std Dev (17 F)
Reg Guide 1.99 Equation 2
(CF = 19.0 F)

FIGURE A-1




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

The Byron Unit 1 & 2 surveillance capsules are located in the reactor between the neutron pads and the vessel wall and are positioned opposite the center of the core (See Figures A-2 and A-3). The test capsules are in baskets attached to the neutron pad. 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. Additionally, since the vessel inlet temperatures are the same, the irradiation temperatures will be the same.

Figure A-2: Arrangement of Surveillance Capsules in the Byron Unit 1 Reactor Vessel

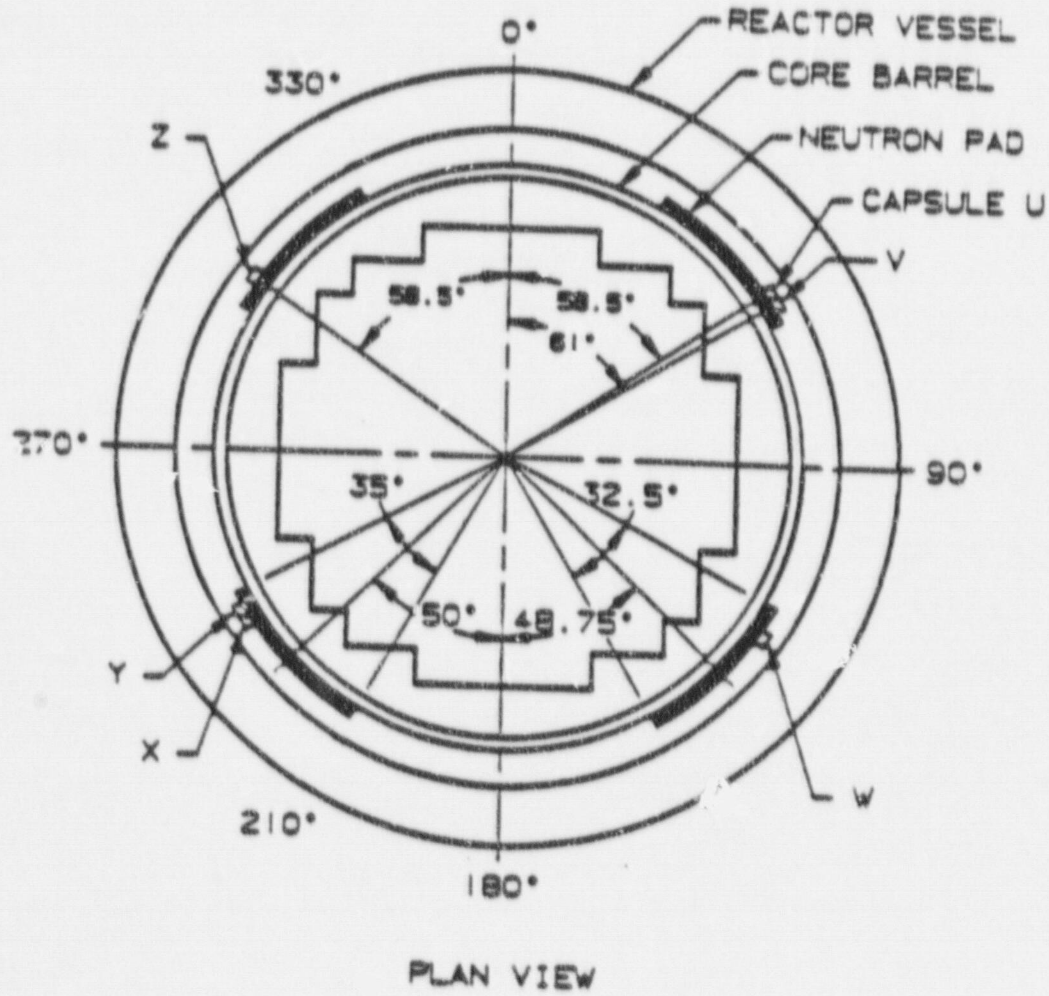
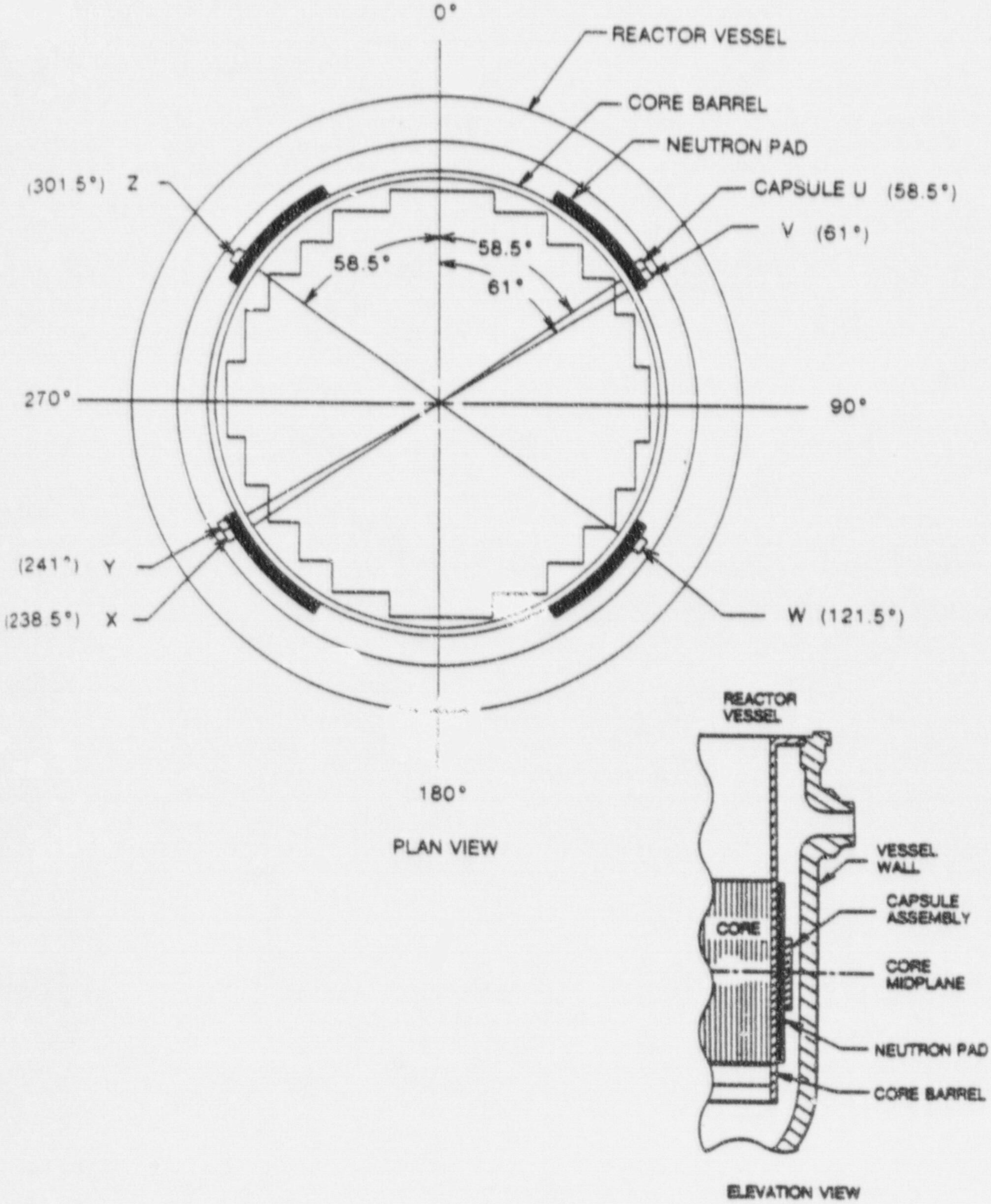


Figure A-3: Arrangement of Surveillance Capsules in the Byron Unit 2 Reactor Vessel



*Criterion 5:* ***The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.***

Byron Units 1 & 2 did not incorporate correlation monitor material in their surveillance program, since this was not a requirement of E185-82. Therefore, Criterion 5 is not applicable.

**RESULTS & CONCLUSIONS**:

Based on the evaluation performed above, it has been determined that there is sufficient data to support integrating the Byron Unit 1 weld metal surveillance data with Byron Unit 2 weld metal surveillance data.

## EFFECT OF WELD METAL INTEGRATION ON BYRON P-T LIMIT CURVES:

| Plant | Previous 1/4T ART | Previous 3/4T ART | New 1/4T ART | New 3/4T ART | Results |
|---|---|---|---|---|---|
| Byron 1 Curves at 8 EFPY<br><br>FDRT/SRPLO-009(94) | 66.37[b] | 57.15[b] | 70[c] | 60[c] | --[a] |
| Byron 2 Curves at 16 EFPY<br><br>WCAP-14063 | 43.5 | 33.2 | 92.6 | 75.8 | Using weld metal integration will be more restrictive to Byron 2 Pressure-temperature curves.<br><br>Therefore, Byron 2 curves will be regenerated and documented in WCAP-14940 |

NOTES:
(a) Even after weld metal integration, still forging-limited. Weld metal integration has no effect.
(b) Calculated at 8 EFPY.
(c) Calculated at 12 EFPY.

The new ART values for Byron Unit 2 are significantly larger. A reasonable applicability date cannot be determined. New curves are to be generated for Byron Unit 2. The results will be documented in WCAP-14940, "Byron Unit 2 Heatup and Cooldown Curves for Normal Operation".

## EFFECT OF WELD METAL INTEGRATION ON BYRON PTS CALCULATIONS:

The weld metal integration CF values were calculated in Section 4 of this report. Specifically, the following weld metal CF values were used to determine the $RT_{PTS}$ values:

| | RG Position 1 CF | RG Position 2 CF |
|---|---|---|
| Byron Units 1 and 2 | 68.0°F | 61.8°F |

The values listed in 'bold' below are those that were affected by the weld integration between Byron Unit 1 and Byron Unit 2. All other vessel material data was obtained from the latest PTS evaluation reports[17,18]. Note that for the Byron Units 1 and 2 $RT_{PTS}$ calculations at 48 EFPY, new fluence values were interpolated to 48 EFPY. The vessel surface fluence results reported in Section 6.0 of the latest Byron Unit 1[9] and Byron Unit 2[11] surveillance capsule analysis reports were used.

TABLE A-4: $RT_{PTS}$ Values for Byron Unit 1

| Material | CF (°F) | f[a] | FF[b] | $RT_{NDT(U)}$ (°F) | M (°F) | $\Delta RT_{PTS}$ (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|
| 32 EFPY | | | | | | | |
| Inter. Shell Forging 5P-5933 | 23.8 | 2.159 | 1.209 | 40 | 28.8 | 28.8 | 97.6 |
| Using surv. capsule data[c] | 19.1 | 2.159 | 1.209 | 40 | 17 | 23.1 | 80.1 |
| Lower Shell Forging 5P-5951 | 26 | 2.159 | 1.209 | 10 | 31.4 | 31.4 | 72.8 |
| Weld Metal WF-336 | **68.0** | 2.159 | 1.209 | -30 | **56** | **82.2** | **108.2** |
| Using surv. capsule data[c] | **61.8** | 2.159 | 1.209 | -30 | **28** | **74.7** | **72.7** |
| 48 EFPY | | | | | | | |
| Inter. Shell Forging 5P-5933 | 23.8 | 3.238 | 1.309 | 40 | 31.2 | 31.2 | 102.4 |
| Using surv. capsule data[c] | 19.1 | 3.238 | 1.309 | 40 | 17 | 25.0 | 82.0 |
| Lower Shell Forging 5P-5951 | 26 | 3.238 | 1.309 | 10 | 34.0 | 34.0 | 78.0 |
| Weld Metal WF-336 | **68.0** | 3.238 | 1.309 | -30 | **56** | **89.0** | **115.0** |
| Using surv. capsule data[c] | **61.8** | 3.238 | 1.309 | -30 | **28** | **80.9** | **78.9** |

NOTES:

(a) $2.159 \times 10^{19}$ n/cm$^2$ (E>1.0 MeV) for 32 EFPY from Byron 1 PTS report (WCAP-13881). The following calculation to obtain the 48 EFPY fluence value:

$$2.159\times10^{19} + \frac{(2.159\times10^{19} - 3.807\times10^{18})}{32 - 5.64\ \text{EFPY}}*(48 - 32\ \text{EFPY}) = 3.238\times10^{19}\ \text{n/cm}^2$$

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

(c) Calculated using a CF based on surveillance capsule data per Regulatory Guide 1.99, Revision 2, Position 2.

## TABLE A-5
## $RT_{PTS}$ VALUES FOR BYRON UNIT 2

| Material | CF (°F) | f[a] | FF[b] | $RT_{NDT(U)}$ (°F) | M (°F) | $\Delta RT_{PTS}$ (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|
| 32EFPY | | | | | | | |
| Lower Shell Forging 49D330-1/49C298-1 | 32.2 | 2.192 | 1.213 | -20 | 34.0 | 39.1 | 53.1 |
| Using surv. Capsule data[c] | 19.8 | 2.192 | 1.213 | -20 | 34[d] | 24.0 | 38.0 |
| Inter. Shell Forging 49D330-1/49C298-1 | 20.0 | 2.192 | 1.213 | -20 | 24.3 | 24.3 | 28.6 |
| Circ. Weld Metal WF447 | **68.0** | 2.192 | 1.213 | 10 | **56** | **82.5** | **148.5** |
| Using surv. capsule data[c] | **61.8** | 2.192 | 1.213 | 10 | **28** | **75.0** | **113.0** |
| 48 EFPY | | | | | | | |
| Lower Shell Forging 49D330-1/49C298-1 | 32.2 | 3.288 | 1.312 | -20 | 34.0 | 42.2 | 56.2 |
| Using surv. Capsule data[c] | 19.8 | 3.288 | 1.312 | -20 | 34[d] | 26.0 | 40.0 |
| Inter. Shell Forging 49D330-1/49C298-1 | 20.0 | 3.288 | 1.312 | -20 | 26.2 | 26.2 | 32.4 |
| Circ. Weld Metal WF447 | **68.0** | 3.288 | 1.312 | 10 | **56** | **89.2** | **155.2** |
| Using surv. capsule data[c] | **61.8** | 3.288 | 1.312 | 10 | **28** | **81.1** | **119.1** |

NOTES:

(a) $2.192 \times 10^{19}$ n/cm$^2$ (E>1.0 MeV) for 32 EFPY from Byron 2 PTS report (WCAP-14054). The following calculation to obtain the 48 EFPY fluence value:

$$2.192\text{x}10^{19} + \frac{(2.192\text{x}10^{19} - 3.[illegible]74\text{x}10^{18})}{32 - 4.634 \text{ EFPY}} * (48 - 32 \text{ EFPY}) = 3.288\text{x}10^{19} \text{ n/cm}^2$$

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

(c) Calculated using a CF based on surveillance capsule data per Regulatory Guide 1.99, Revision 2, Position 2.

(d) Double margin is used here due to the base metal surveillance capsule data exceeding the one sigma criteria from the credibility evaluation. See page A-9.

# APPENDIX B

## WELD METAL INTEGRATION FOR BRAIDWOOD UNITS 1 AND 2

**INTRODUCTION:**

Westinghouse performed an evaluation to determine if the weld wire data of the Braidwood Units 1 and 2 surveillance programs can be integrated. The evaluation was based on the following criteria:

1. What weld wire heat number, flux, and flux lot were used to fabricate the surveillance program weld metal of each unit,
2. What vendor fabricated the welds and in what time frame,
3. What heat treatment did each surveillance program weld receive,
4. Is the initial $RT_{NDT}$ of the welds the same or relatively close,
5. Is the initial upper shelf energy of the welds the same or relatively close,
6. Is the geometry of the plants the same,
7. Is the type of fuel in all plants the same,
8. Are the fuel loading patterns in the plants similar (i.e., low leakage, etc.),
9. What is the projected 32 effective full power year surface fluence of each plant,
10. What vessel inlet temperatures do the plants operate at,
11. What are the differences in the capsule lead factors of the plants,
12. Can the criteria for credibility in 10 CFR Part 50.61 be met for each plant?

**EVALUATION:**

1. *What weld wire heat number, flux and flux lot numbers were used to fabricate the welds?*
   Braidwood 1: The weld metal is classification EF2N Low Cu, MnMoNi Heat number 442011, with a Linde grade 80 type flux, lot number 8061. This is the same heat number used in the limiting beltline weld (seam WF-562).
   Braidwood 2: The weld metal is classification EF2N Low Cu, MnMoNi Heat number 442011, with a Linde grade 80 type flux, lot number 8061. This is the same heat number used in the limiting beltline weld (seam WF-562).

   The Braidwood Units 1 and 2 surveillance program weld metals were fabricated with the same heat of weld wire and the same type of flux. Therefore, this information supports the integration of the surveillance program test results for these welds.

2. *What vendor fabricated the welds and in what time frame ?*
   Braidwood 1: B&W fabricated the welds in the late 1970's
   Braidwood 2: B&W fabricated the welds in the late 1970's

   The welds for Braidwood 1 and 2 were fabricated in the same time frame and by the same vendor. Therefore, this information supports the integration of the surveillance program test results for these welds.

3. *What heat treatment did each surveillance program weld receive?*
   Braidwood 1: 1100 - 1150°F for 12¼ hours; furnace cooled.
   Braidwood 2: 1150 ± 50°F for 12½ hours; furnace cooled.

   The post-weld stress relief heat treatment given to the Braidwood 1 and 2 surveillance program welds was slightly different. However, based on engineering judgement, the slight differences in temperature and time should not cause a significant difference in the material toughness properties.

4. *Is the initial $RT_{NDT}$ of the welds the same or relatively close?*
   Braidwood 1: 40 °F
   Braidwood 2: 40 °F

   The Braidwood Units 1 and 2 initial $RT_{NDT}$ values are identical. Therefore, this information supports the integration of the surveillance program test results for these welds.

5. *Is the initial upper shelf energy of the surveillance welds the same or relatively close?*
   Braidwood 1: 70 ft-lb
   Braidwood 2: 71 ft-lb

   The initial upper shelf energy values for the surveillance weld materials in the Braidwood surveillance programs are very similar. Therefore, this information supports the integration of the surveillance program test results for these welds.

6. *Is the geometry of the plants the same?*
   All four plants have a reactor vessel inner diameter of 173 inches, a reactor vessel beltline thickness of 8.5 inches (excluding the cladding), and a NSSS 4-loop power rating of 3411 MWT. In addition, all four plants have neutron pads and the surveillance capsules are located at the same azimuthal angles.

7. *Is the fuel design in all plants the same?*
   Braidwood 1 & 2 use 17X17 rod array fuel assemblies with the same fuel design, thus producing similar radiation effects at the surveillance capsules.

8. *Are the fuel loading patterns in the plants similar (i.e. low leakage, etc.)?*
   Braidwood 1 & 2 use a low leakage loading pattern.

9. *What is the projected 32 effective full power year surface fluence of each plant?*
   Based on the information provided below, the projected vessel surface fluence (E>1.0 MeV) values at 32 EFPY for Braidwood Unit 1 are essentially the same as Braidwood Unit 2.

Braidwood Unit 1

| 0° | 15° | 25° | 35° | 45° |
|---|---|---|---|---|
| $1.321 \times 10^{19}$ | $1.984 \times 10^{19}$ | $2.239 \times 10^{19}$ | 1.86E$\times 10^{19}$ | $2.162 \times 10^{19}$ |

Braidwood Unit 2

| 0° | 15° | 25° | 35° | 45° |
|---|---|---|---|---|
| $1.299 \times 10^{19}$ | $1.924 \times 10^{19}$ | $2.199 \times 10^{19}$ | $1.861 \times 10^{19}$ | $2.174 \times 10^{19}$ |

*10. What vessel inlet temperatures do the plants operate?* (Per Reference 25)

| CYCLE # | Braidwood Unit 1 Tcold (°F) | Braidwood Unit 2 Tcold (°F) |
|---|---|---|
| 1 | 557 (Cap. U)*** | 557 (Cap. U)*** |
| 2 | 551 | 551 |
| 3 | 551 | 551 |
| 4 | 551 (Cap. X)*** | 551 (Cap. X)*** |
| 5* | 551 | 551 |
| 5** | 554 | - - |
| 6 | 554 | 551 |

* Between A1R04 & A1M05 (Approximately ½ cycle duration) - Unit 1 Only.
** Between A1M05 & A1R05 (Approximately ½ cycle duration) - Unit 1 Only.
*** See Appendix C, page C-5.

11. *What are the differences in the capsule lead factors of the plants?*
Based on the information provide in Table B-1, the lead factors of the surveillance capsules in Braidwood Unit 1 are essentially the same as Braidwood Unit 2.

TABLE B-1
Surveillance Capsule Lead Factors for Braidwood Units 1 & 2

| Braidwood Unit 1 | | | Braidwood Unit 2 | | |
|---|---|---|---|---|---|
| Capsule | Location | Lead Factor | Capsule | Location | Lead Factor |
| U | 58.5° | 4.03 | U | 58.5° | 4.00 |
| X | 238.5° | 4.03 | X | 238.5° | 4.02 |
| W | 121.5° | 4.03 | W | 121.5° | 4.02 |
| Z | 301.5° | 4.03 | Z | 301.5° | 4.02 |
| V | 61.0° | 3.73 | V | 61.0° | 3.70 |
| Y | 241.0° | 3.73 | Y | 241.0° | 3.70 |

Based on the projected vessel surface fluence and lead factor values for Braidwood 1 & 2, the Braidwood 1 & 2 surveillance capsules will have approximately the same flux rates and irradiation temperatures. This supports the use of the surveillance weld data in both programs to evaluate the reactor vessel integrity of the Braidwood units.

12. *Can the criteria for credibility in 10 CFR Part 50.61 be met for each plant?*
    Credibility will be evaluated for all the surveillance capsule data (base metal & weld metal) for Braidwood Units 1 and 2.

***Criterion 1:*** ***The materials in the surveillance capsules must be those which are controlling materials with regard to radiation embrittlement.***

The following is a list of the beltline materials contained in the Braidwood Units 1 and 2 surveillance programs:

Braidwood Unit 1: Lower shell forging 49D867/49C813-1-1
Circumferential weld seam WF-562, heat number 442011, with a Linde grade 80 type flux, lot number 8061. (This is the same heat number used in the limiting beltline weld.)

Braidwood Unit 2: Lower shell forging 50D102/50C97-1-1
Circumferential weld seam WF-562, heat number 442011, with a Linde grade 80 type flux, lot number 8061. (This is the same heat number used in the limiting beltline weld.)

Based on the information provided in the material selection documents, WCAP-9807 (Braidwood 1, See Ref. 26) and WCAP-11188 (Braidwood 2, See Ref. 27), these materials are judged to be the most controlling with regard to radiation embrittlement for each unit. Therefore, Criteria #1 is met for both Braidwood units.

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

Plots of Charpy energy versus temperature for the unirradiated condition are presented in WCAP-9807, "Commonwealth Edison Company Braidwood Station Unit No. 1 Reactor Vessel Radiation Surveillance Program," dated February 1981 and WCAP-11188, "Commonwealth Edison Company Braidwood Station Unit No. 2 Reactor Vessel Radiation Surveillance Program," dated December 1986. Plots of Charpy energy versus temperature for the irradiated conditions are presented in the WCAP reports for Capsules U & X for both units.

Based on engineering judgement, the scatter in the data presented in these reports is small enough to determine the 30 ft-lb temperature and the upper shelf energy of the Braidwood Units 1 & 2 surveillance weld metals unambiguously. Therefore, the Braidwood Units 1 & 2 surveillance materials meet this criteria.

*Criterion 3:* ***Where there are two or more sets of surveillance data from one reactor, the scatter of $\Delta RT_{NDT}$ values must be less than 28°F for welds and 17°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.***

The least squares method, as described in Regulatory Position 2.1, will be utilized in determining a best-fit line for this data to determine if this criteria is met. It should be noted here that the ratio procedure is not applied in this instance since only surveillance capsule data is being evaluated.

# TABLE B-2[c]

## Braidwood Units 1 & 2 Surveillance Capsule Data Chemistry Factor for Best Fit Line

| Material | Capsule | f[a] | FF[b] | Measured $\Delta RT_{NDT}$ | FF x $\Delta RT_{NDT}$ | $FF^2$ |
|---|---|---|---|---|---|---|
| Braidwood Unit 1 Lower Shell Forging 49D867-1/49C813-1 (Axial) | U | $3.814 \times 10^{18}$ | 0.733 | 5 | 3.666 | 0.538 |
| | X | $1.144 \times 10^{19}$ | 1.038 | 30 | 31.127 | 1.077 |
| Braidwood Unit 1 Lower Shell Forging 49D867-1/49C813-1 (Tangential) | U | $3.814 \times 10^{18}$ | 0.733 | 0 | 0.000 | 0.538 |
| | X | $1.144 \times 10^{19}$ | 1.038 | 25 | 25.939 | 1.077 |
| | | | | Sum: | 60.733 | 3.228 |
| | Chemistry Factor = 60.733 ÷ 3.228 = **18.8°F** | | | | | |
| Braidwood Unit 2 Lower Shell Forging 50D102-1/50C97-1 (Axial) | U | $3.933 \times 10^{18}$ | 0.741 | 0 | 0.00 | 0.550 |
| | X | $1.126 \times 10^{19}$ | 1.033 | 3 | 3.099 | 1.067 |
| Braidwood Unit 2 Lower Shell Forging 50D102-1/50C97-1 (Tangential) | U | $3.933 \times 10^{18}$ | 0.741 | 5 | 3.707 | 0.550 |
| | X | $1.126 \times 10^{19}$ | 1.033 | 35 | 36.160 | 1.067 |
| | | | | Sum: | 42.966 | 3.234 |
| | Chemistry Factor = 42.966 ÷ 3.234 = **13.3°F** | | | | | |
| Braidwood Unit 1 Weld Metal[e] | U | $3.814 \times 10^{18}$ | 0.733 | 10 | 7.333 | 0.538 |
| | X | $1.144 \times 10^{19}$ | 1.038 | 25 | 25.95 | 1.077 |
| Braidwood Unit 2 Weld Metal[e] | U | $3.933 \times 10^{18}$ | 0.741 | 0 | 0.00 | 0.550 |
| | X | $1.126 \times 10^{19}$ | 1.033 | 20 | 20.66 | 1.067 |
| | | | | Sum: | 53.943 | 3.232 |
| | Chemistry Factor = 53.943 ÷ 3.232 = **16.7°F** | | | | | |

NOTES:

(a) f = Fluence ($10^{19}$ n/cm$^2$, E > 1.0 MeV)

(b) FF = Fluence Factor = $f^{(0.28 - 0.1 * \log f)}$

(c) Values of f, FF, and $\Delta RT_{NDT}$ values were taken from Table 2 of WCAP-14243 (Braidwood Unit 1 P-T Limits) and WCAP-14230 (Braidwood Unit 2 P-T Limits).

(d) $CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$

(e) For both welds: WF-562, heat # 442011.

TABLE B-3

Best Fit Evaluation for Braidwood 1 & 2 Surveillance Materials

| Base Material | CF | FF | $\Delta RT_{NDT}$ (30 ft-lb) (°F) | Best Fit[a] $\Delta RT_{NDT}$ (°F) | Scatter of $\Delta RT_{NDT}$ (°F) | < 17°F (Base Metals) < 28°F (Weld Metal) |
|---|---|---|---|---|---|---|
| Braidwood 1 Weld Metal | 16.7 | 0.733 | 10 | 12.2 | -2.2 | Yes |
| | 16.7 | 1.038 | 25 | 17.3 | 7.7 | Yes |
| Braidwood 2 Weld Metal | 16.7 | 0.741 | 0 | 12.4 | -12.4 | Yes |
| | 16.7 | 1.033 | 20 | 12.3 | 7.7 | Yes |
| Braidwood Unit 1 Lower Shell Forging 49D867-1/49C813-1 (Axial) | 18.8 | 0.733 | 5 | 13.8 | -8.8 | Yes |
| | 18.8 | 1.038 | 30 | 19.5 | 10.5 | Yes |
| Braidwood Unit 1 Lower Shell Forging 49D867-1/49C813-1 (Tangential) | 18.8 | 0.733 | 0 | 13.8 | -13.8 | Yes |
| | 18.8 | 1.038 | 25 | 19.5 | 5.5 | Yes |
| Braidwood Unit 2 Lower Shell Forging 50D102-1/50C97-1 (Axial) | 13.3 | 0.741 | 0 | 9.9 | -9.9 | Yes |
| | 13.3 | 1.033 | 3 | 13.7 | -10.7 | Yes |
| Braidwood Unit 2 Lower Shell Forging 50D102-1/50C97-1 (Tangential) | 13.3 | 0.741 | 5 | 9.9 | -4.9 | Yes |
| | 13.3 | 1.033 | 35 | 13.7 | 21.3 | No |

NOTES:

(a) Best Fit Line Per Equation 2 of Reg. Guide 1.99 Rev. 2 Position 1.1.

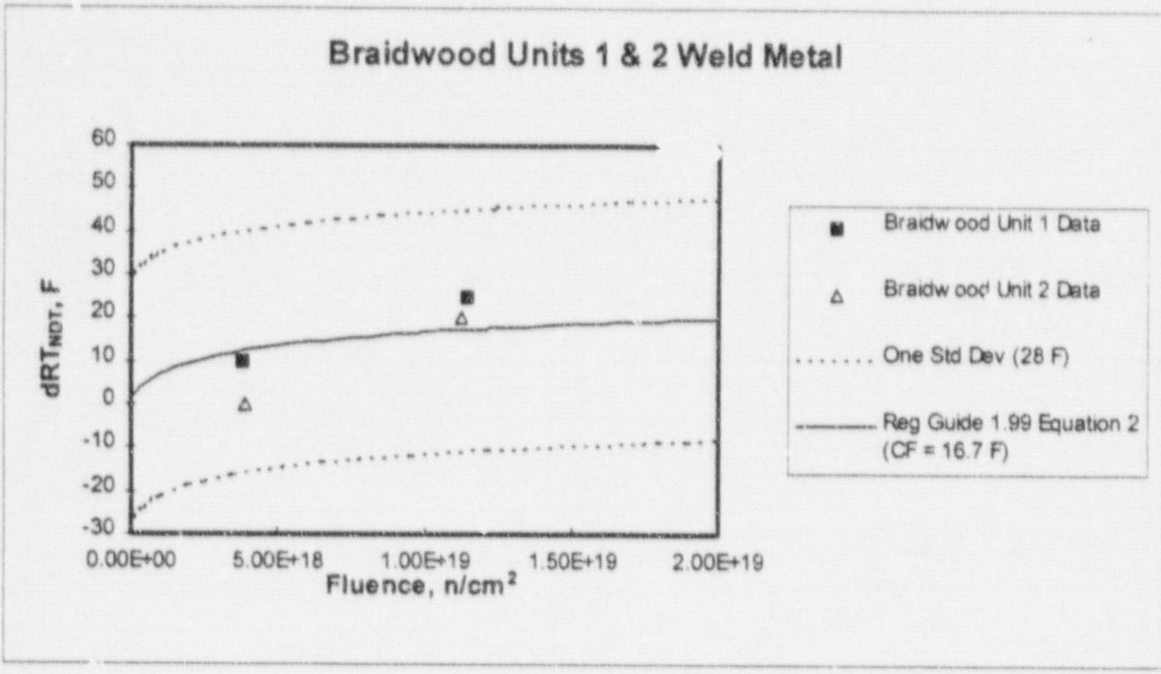
Weld Metal:

The scatter of $\Delta RT_{NDT}$ values (Figure B-1) about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 28°F for weld metal. As shown above, the error is within 28°F of the best-fit line. Therefore, this criteria is met for the Braidwood Units 1 & 2 surveillance weld material.

Base Material:

The scatter of $\Delta RT_{NDT}$ values (Figure B-1) about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 17°F for base metal. As shown in Table B-3, the error for Braidwood Unit 1 is within 17°F of the best-fit line and the error for one point for Braidwood Unit 2 is not within 17°F of the best-fit line. Therefore, this criteria is met for Braidwood Unit 1 base metal but not for Braidwood Unit 2 base metal. As a result of the Braidwood Unit 2 base metal exceeding this criteria, the margin term that is calculated for the Braidwood Unit 2 base metal using Position 2.1 of the Reg. Guide should now be doubled. Thus allowing this data to be used in the evaluations of pressure-temperature limit curves and PTS.

Figure B-1


Braidwood Units 1 & 2 Weld Metal
dRTNDT, F
Fluence, n/cm²
60
50
40
30
20
10
0
-10
-20
-30
0.00E+00
5.00E+18
1.00E+19
1.50E+19
2.00E+19
Braidwood Unit 1 Data
Braidwood Unit 2 Data
One Std Dev (28 F)
Reg Guide 1.99 Equation 2
(CF = 16.7 F)

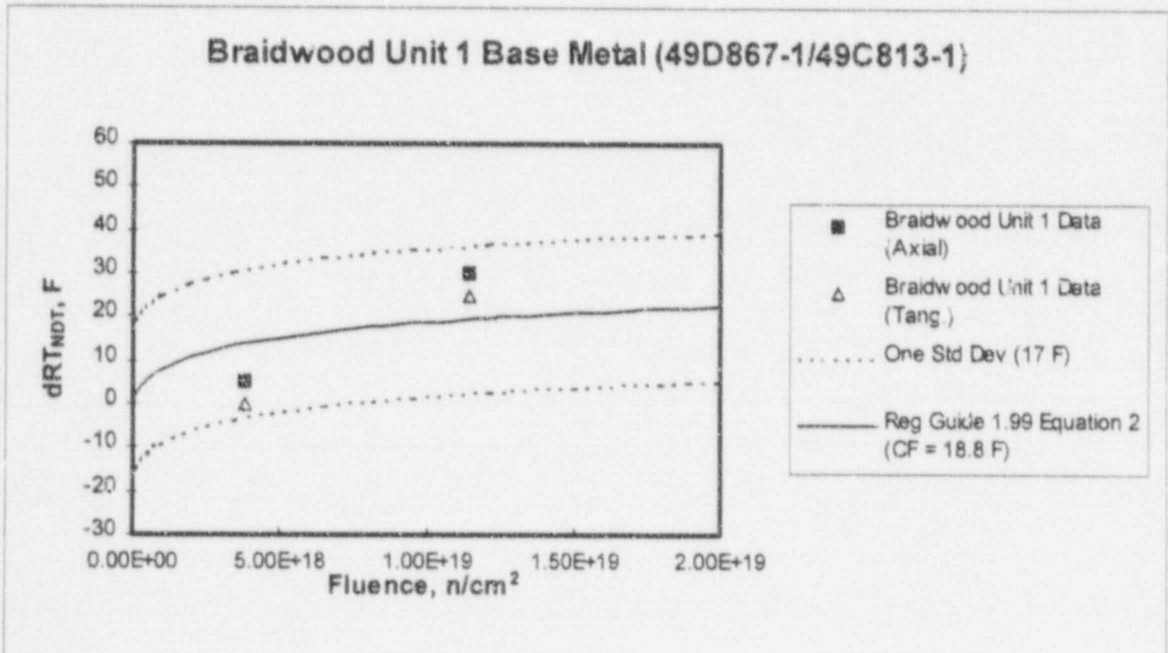
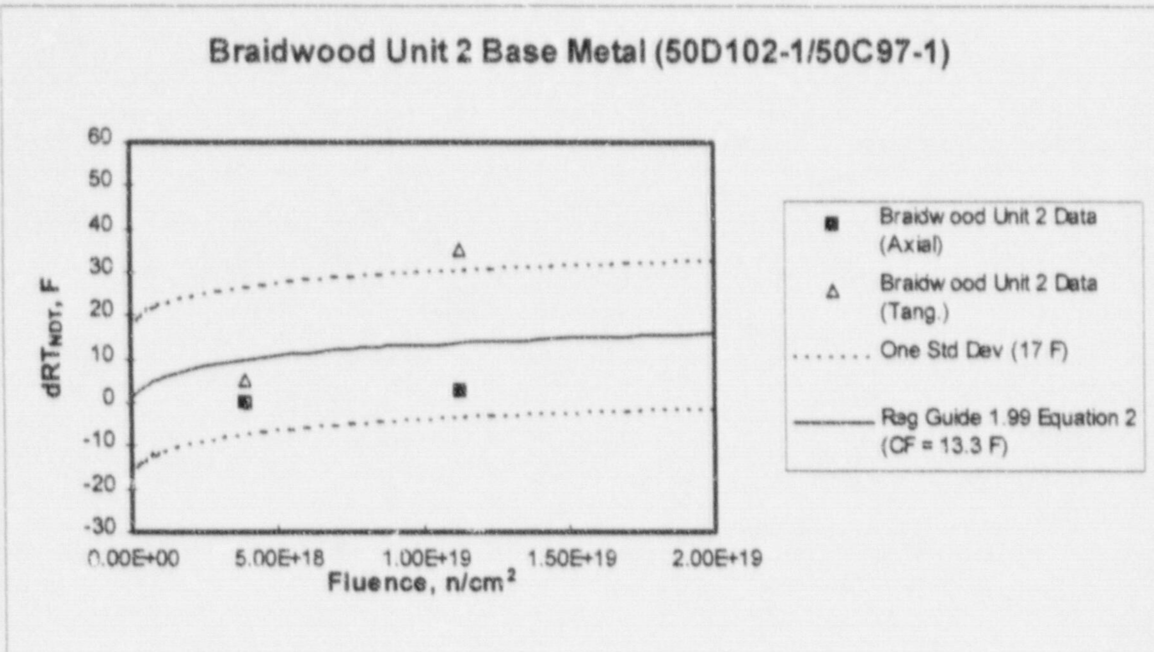

Braidwood Unit 1 Base Metal (49D867-1/49C813-1)
dRTNDT, F
Fluence, n/cm²
60
50
40
30
20
10
0
-10
-20
-30
0.00E+00
5.00E+18
1.00E+19
1.50E+19
2.00E+19
Braidwood Unit 1 Data
(Axial)
Braidwood Unit 1 Data
(Tang.)
One Std Dev (17 F)
Reg Guide 1.99 Equation 2
(CF = 18.8 F)

Figure B-1 (Continued)




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

The Braidwood Unit 1 & 2 surveillance capsules are located in the reactor between the neutron pads and the vessel wall and are positioned opposite the center of the core (See Figures B-2 and B-3). The test capsules are in baskets attached to the neutron pad. 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. Additionally, since the vessel inlet temperatures are the same, the irradiation temperatures will be the same.

Figure B-2: Arrangement of Surveillance Capsules in the Braidwood Unit 1 Reactor Vessel

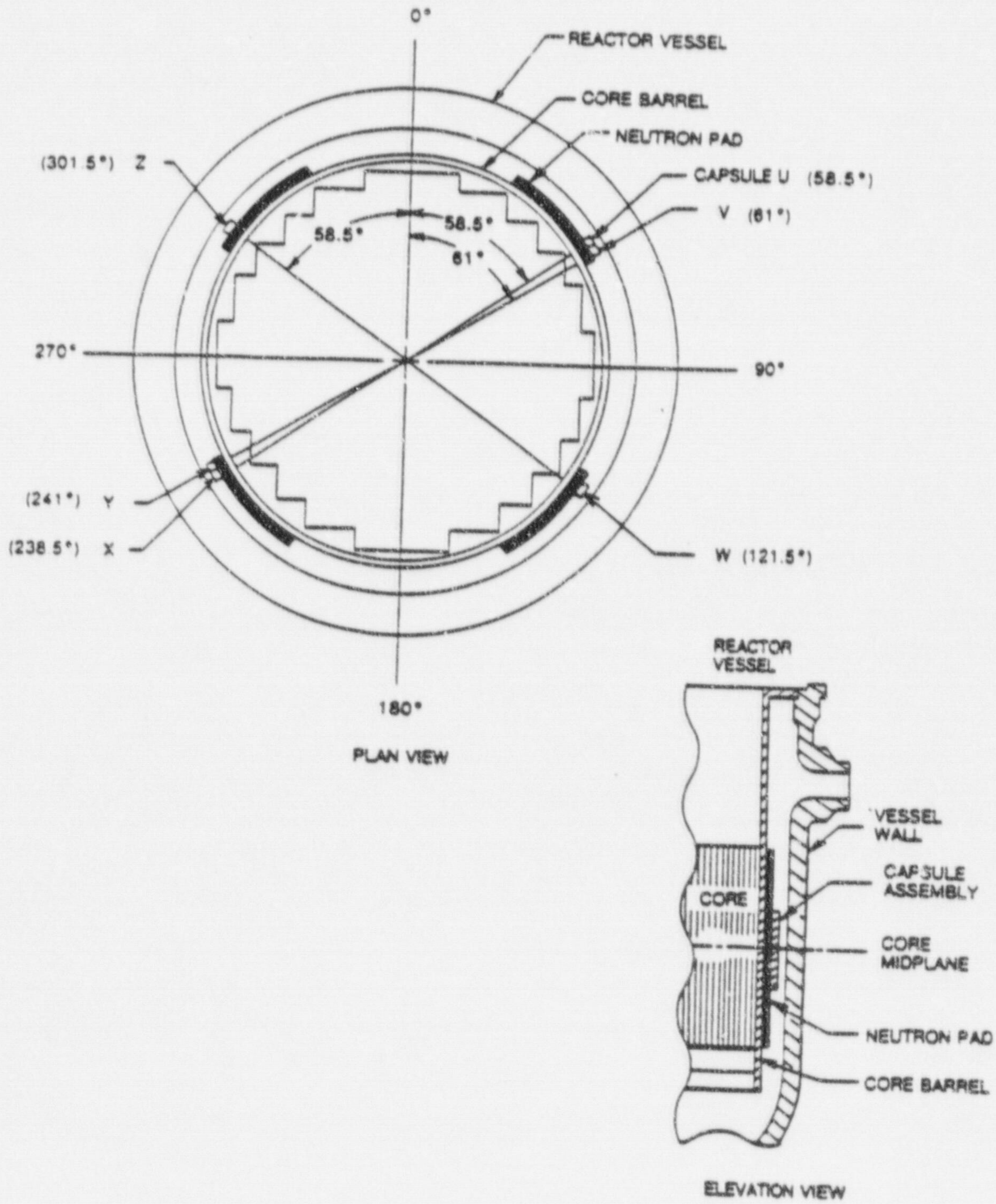
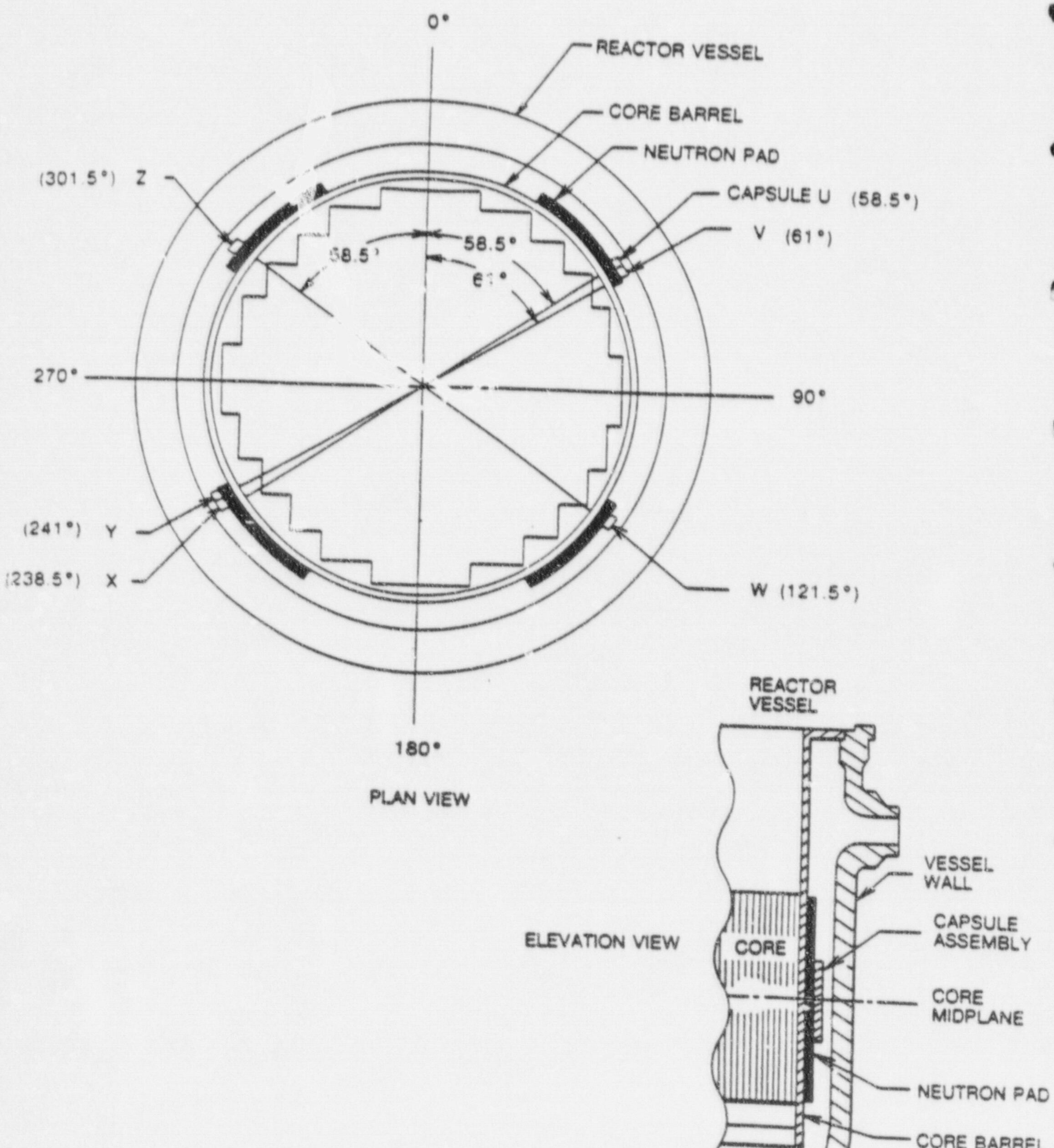


Figure B-3: Arrangement of Surveillance Capsules in the Braidwood Unit 2 Reactor Vessel



*Criterion 5:* ***The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.***

Braidwood Units 1 & 2 did not incorporate correlation monitor material in their surveillance program, since this was not a requirement of E185-82. Therefore, Criterion 5 is not applicable.

**RESULTS & CONCLUSIONS**:

Based on the evaluation performed above it has been determined that there is sufficient data to support integrating the Braidwood Unit 1 weld metal surveillance data with Braidwood Unit 2 weld metal surveillance data.

## TABLE B-4

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

| | Reference | Best-Estimate Cu | Ni | | | | |
|---|---|---|---|---|---|---|---|
| B&W Weld Qualification | BAW-2261 | 0.028 | 0.63 | | | | |
| B&W Weld Qualification | | 0.03 | 0.65 | | | | |
| B&W Weld Qualification | | 0.04 | 0.67 | | | | |
| Braidwood 1 Surv. Data | See Below | 0.032 | 0.671 ---> | 0.03 | 0.67 | Surv. CF = 41 |
| Braidwood 2 Surv. Data | See Below | 0.033 | 0.708 ---> | 0.03 | 0.71 | Surv. CF = 41 |
| Best-Estimate Chemistry**: | | 0.033 | 0.666 ---> | **0.03** | **0.67** | **Best Est. CF = 41** |
| Standard Deviation: | | 0.005 | 0.029 | **Braidwood 1 & 2** | | **Ratio = 1.0** |

Surveillance Chemistry Results:

***Braidwood Unit 1***

| Reference | Cu | Ni |
|---|---|---|
| WCAP-9807 | 0.04* | 0.67* |
| WCAP-12685 | 0.035 | 0.666 |
| | 0.033 | 0.666 |
| | 0.034 | 0.723 |
| | 0.035 | 0.709 |
| | 0.034 | 0.728 |
| | 0.035 | 0.699 |
| | 0.035 | 0.751 |
| | 0.031 | 0.683 |
| | 0.032 | 0.673 |
| | 0.029 | 0.66[illegible] |
| | 0.029 | 0.686 |
| | 0.034 | [illegible] |
| | 0.033 | 0.651 |
| | 0.033 | 0.698 |
| | 0.031 | 0.656 |
| WCAP-14241 | 0.031 | 0.655 |
| | 0.029 | 0.647 |
| | 0.028 | 0.638 |
| | 0.031 | 0.655 |
| | 0.031 | 0.650 |
| | 0.032 | 0.661 |
| | 0.033 | 0.667 |
| | 0.028 | 0.648 |
| | 0.02[illegible] | 0.644 |
| | 0.034 | 0.668 |
| | 0.033 | 0.656 |
| | 0.036 | 0.658 |
| | 0.036 | 0.671 |
| | 0.036 | 0.667 |
| Average | **0.032** | **0.671** |

***Braidwood Unit 2***

| Reference | Cu | Ni |
|---|---|---|
| WCAP-11188 | 0.040 | 0.64 |
| WCAP-14228 | 0.033 | 0.724 |
| | 0.034 | 0.711 |
| | 0.033 | 0.714 |
| | 0.038 | 0.780 |
| | 0.035 | 0.737 |
| | 0.033 | 0.728 |
| | 0.032 | 0.752 |
| | 0.032 | 0.743 |
| | 0.031 | 0.730 |
| | 0.032 | 0.711 |
| | 0.032 | 0.728 |
| | 0.031 | 0.703 |
| | 0.032 | 0.687 |
| | 0.033 | 0.703 |
| | 0.033 | 0.695 |
| WCAP-12845 | 0.032 | 0.704 |
| | 0.034 | 0.754 |
| | 0.032 | 0.698 |
| | 0.026 | 0.623 |
| | 0.028 | 0.635 |
| | 0.031 | 0.679 |
| | 0.029 | 0.644 |
| | 0.032 | 0.699 |
| | 0.034 | 0.765 |
| | 0.031 | 0.673 |
| | 0.034 | 0.724 |
| | 0.035 | 0.747 |
| | 0.033 | 0.711 |
| | 0.031 | 0.688 |
| | 0.035 | 0.750 |
| | 0.031 | 0.685 |
| Average | **0.033** | **0.708** |

* Not used in Average calculation; reported for completeness. The same value appears in the material test reports and the surveillance program report.

** The best estimate chemistry values was obtained using the "average of averages" approach.

## EFFECT OF WELD METAL INTEGRATION ON BRAIDWOOD P-T LIMIT CURVES:

| Plant | Previous 1/4T ART | Previous 3/4T ART | New 1/4T ART | New 3/4T ART | Result |
|---|---|---|---|---|---|
| Braidwood 1 Curves at 16 EFPY<br><br>WCAP-14243 | 76.6 | 65.4 | 69.7 | 60.6 | Current curves/PTS evaluation are conservative.<br><br>New Applicability Date: **27.9 EFPY** |
| Braidwood 2 Curves at 16 EFPY<br><br>WCAP-14230 | 62.6 | 55.7 | 69.5 | 60.4 | Current curves/PTS evaluation are NOT conservative. Using weld metal integration will be more restrictive.<br><br>New Applicability Date: **7.4 EFPY**<br><br>Braidwood 2 curves will be regenerated and documented in WCAP-14970 |

After the Braidwood Units 1 and 2 surveillance weld metal is integrated, the following calculations show the new applicability dates of the heatup/cooldown pressure-temperature limit curves for Braidwood Unit 1.

BRAIDWOOD UNIT 1:

<u>Weld Metal calculations based on a 1/4T ART = 76.6°F</u>:
(The following data is from Braidwood Unit 1 heatup/cooldown curve report, WCAP-14243)

Per Regulatory Guide (RG) 1.99, Revision 2: **ART = I + M + (CF * FF)**

Using the "Previous" ART values and initial $RT_{NDT}$, this equation was used to *back-calculate* the fluence factor (FF) and the vessel surface fluence value. This fluence value was then used to determine a new applicability date (in terms of EFPY) for the current pressure-temperature limit curves.

For Braidwood Units 1 and 2, the margin term from the above equation was calculated as (CF*FF) in the latest heatup/cooldown curve WCAP report. The following text explains this methodology from Regulatory Guide 1.99, Revision 2. A more detailed explanation can be found in Section 4 of this report.

The Margin term is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial $RT_{NDT}$ margin term ($\sigma_I$) is 0°F when the initial $RT_{NDT}$ is a measured value (as is the case for the Braidwood units). Additionally, the term $\sigma_\Delta$ need not exceed 0.5 times the mean value of $\Delta RT_{NDT}$.

Therefore, when the $\Delta RT_{NDT}$ value is multiplied by 0.5 and plugged into the above equation, the effect is 2 * ($\Delta RT_{NDT}$ /2), which is the $\Delta RT_{NDT}$ (or CF * FF).

**ART = I + (CF * FF) + (CF * FF)**

76.6°F = 40°F + (16.7 * 1/4T FF)°F + (16.7 * 1/4T FF)°F ==> 1/4T FF = 1.0958

1.0958 = 1/4T $f^{(0.28 - 0.1 \log 1/4T f)}$ ==> 1/4T f = $1.4124 \times 10^{19}$ n/cm$^2$

$1.4124 \times 10^{19} = f * e^{(-0.24 * 8.5 * 0.25)}$ ==> $f = 2.352 \times 10^{19}$ n/cm$^2$

This fluence value will occur after 32 EFPY, per Table 6-15 of WCAP-14241. The following calculation will determine the applicability date in terms of EFPY.

Fluence at X EFPY = Fluence at 32 EFPY + (X - 32 EFPY) * Fluence/EFPY

$$2.352 \times 10^{19} = 2.239 \times 10^{19} + (X - 32\ \text{EFPY}) * \frac{(2.239 \times 10^{19} - 1.120 \times 10^{19})}{32 - 16\ \text{EFPY}}$$

**X = 33.6 EFPY**

Weld Metal calculations based on a 3/4T ART = 65.4°F:
(The following data is from Braidwood Unit 1 heatup/cooldown curve report, WCAP-14243)

ART = I + M + (CF * FF)

65.4°F = 40°F + (16.7 * 3/4T FF)°F + (16.7 * 3/4T FF)°F ==> 3/4T FF = 0.76047

0.76047 = 3/4T $f^{(0.28 - 0.1 \log 3/4T f)}$ ==> 3/4T f = $0.4221 \times 10^{19}$ n/cm$^2$

$0.4221 \times 10^{19} = f * e^{(-0.24 * 8.5 * 0.75)}$ ==> $f = 1.9493 \times 10^{19}$ n/cm$^2$

This fluence value will occur between 16 and 32 EFPY, per Table 6-15 of WCAP-14241. The following calculation will determine the applicability date in terms of EFPY.

Fluence at X EFPY = Fluence at 16 EFPY + (X - 16 EFPY) * Fluence/EFPY

$$1.9493 \times 10^{19} = 1.120 \times 10^{19} + (X - 16\ EFPY) * \frac{(2.239 \times 10^{19} - 1.120 \times 10^{19})}{32 - 16\ EFPY}$$

**X = 27.9 EFPY**

Therefore, after the weld metal integration for Braidwood Units 1 and 2 is implemented, the Braidwood Unit 1 heatup/cooldown curves presented in WCAP-14243 will be applicable to 27.9 EFPY.

BRAIDWOOD UNIT 2

Weld Metal calculations based on a 1/4T ART = 62.6°F:

(The following data is from Braidwood Unit 2 heatup/cooldown curve report, WCAP-14230.)

ART = I + M + (CF * FF)

62.6°F = 40°F + (16.7 * 1/4T FF)°F + (16.7 * 1/4T FF)°F ==> 1/4T FF = 0.6766

$0.6766 = 1/4T\ f^{(0.28 - 0.1 \log 1/4T\ f)}$ ==> $1/4T\ f = 3.075 \times 10^{18}\ n/cm^2$

$3.075 \times 10^{18} = f * e^{(-0.24 * 8.5 * 0.25)}$ ==> $f = 5.120 \times 10^{18}\ n/cm^2$

This fluence value will occur between 4.215 and 16 EFPY, per Table 6-15 of WCAP-14228. The following calculation will determine the applicability date in terms of EFPY.

Fluence at X EFPY = Fluence at 4.215 EFPY + (X - 4.215 EFPY) * Fluence/EFPY

$$5.120 \times 10^{18} = 2.896 \times 10^{18} + (X - 4.215\ EFPY) * \frac{(1.100 \times 10^{19} - 2.896 \times 10^{18})}{16 - 4.215\ EFPY}$$

**X = 7.4 EFPY**

<u>Weld Metal calculations based on a 3/4T ART = 55.7°F:</u>

ART = I + M + (CF * FF)

55.7°F = 40°F + (16.7 * 3/4T FF)°F + (16.7 * 3/4T FF)°F ==> 3/4T FF = 0.47006

$0.47006 = 3/4T\ f^{(0.28 - 0.1 \log 3/4T\ f)}$ ==> $3/4T\ f = 0.1292 \times 10^{19}\ n/cm^2$

$0.1292 \times 10^{19} = f * e^{(-0.24 * 8.5 * 0.75)}$ ==> $f = 5.966 \times 10^{18}\ n/cm^2$

This fluence value will occur between 4.215 and 16 EFPY, per Table 6-15 of WCAP-14228. The following calculation will determine the applicability date in terms of EFPY.

Fluence at X EFPY = Fluence at 4.215 EFPY + (X - 4.215 EFPY) * Fluence/EFPY

$$5.966 \times 10^{18} = 2.896 \times 10^{18} + (X - 4.215\ EFPY) * \frac{(1.100 \times 10^{19} - 2.896 \times 10^{8})}{16 - 4.215\ EFPY}$$

**X = 8.7 EFPY**

After the weld metal integration for Braidwood Units 1 and 2 is implemented, the Braidwood Unit 2 heatup/cooldown curves presented in WCAP-14230 will be applicable to 7.4 EFPY.

## EFFECT OF WELD METAL INTEGRATION ON BRAIDWOOD PTS CALCULATIONS:

The weld metal integration CF values were calculated in Section 4 of this report. Specifically, the following weld metal CF values were used to determine the $RT_{PTS}$ values:

| | RG Position 1 CF | RG Position 2 CF |
|---|---|---|
| Braidwood Units 1 and 2 | 41.0°F | 16.7°F |

The values listed in 'bold' below are those that were affected by the weld integration between Braidwood Unit 1 and Braidwood Unit 2. All other vessel material data was obtained from the latest PTS evaluation reports[19,20].

TABLE B-4

$RT_{PTS}$ Values for Braidwood Unit 1

| Material | CF (°F) | f | FF[(a)] | $RT_{NDT(U)}$ (°F) | M (°F) | $\Delta RT_{PTS}$ (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|
| 32 EFPY | | | | | | | |
| Inter. Shell Forging | 31.0 | 2.239 | 1.218 | -30 | 34 | 37.77 | 41.8 |
| Lower Shell Forging | 26.0 | 2.239 | 1.218 | -20 | 31.63 | 31.68 | 43.4 |
| using S/C data[(b)] | 18.8 | 2.239 | 1.218 | -20 | 17 | 22.90 | 19.8 |
| Weld Metal WF-562 | 41.0 | 2.239 | 1.218 | 40 | 49.95 | 49.95 | 139.9 |
| using S/C data[(b)] | **16.7** | 2.239 | 1.218 | 40 | **20.34** | **20.34** | **80.7** |
| 48 EFPY | | | | | | | |
| Inter. Shell Forging | 31.0 | 3.358 | 1.317 | -30 | 34 | 40.83 | 44.8 |
| Lower Shell Forging | 26.0 | 3.358 | 1.317 | -20 | 34 | 34.25 | 48.3 |
| using S/C data[(b)] | 18.8 | 3.358 | 1.317 | -20 | 17 | 24.76 | 21.8 |
| Weld Metal WF-562 | 41.0 | 3.358 | 1.317 | 40 | 54.00 | 54.00 | 148.0 |
| using S/C data[(b)] | **16.7** | 3.358 | 1.317 | 40 | **21.99** | **21.99** | **84.0** |

NOTES:

(a) FF (Fluence factor) = $f^{(0.28 - 0.10 \cdot \log f)}$

(b) Calculated using a CF based on surveillance capsule data per Regulatory Guide 1.99, Revision 2, Position 2.

TABLE B-5
$RT_{PTS}$ Values for Braidwood Unit 2

| Material | CF (°F) | f | FF[a] | $RT_{NDT(U)}$ (°F) | M (°F) | $\Delta RT_{PTS}$ (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|
| 32 EFPY | | | | | | | |
| Upper Shell Forging | 20.0 | 2.199 | 1.214 | -30 | 24.28 | 24.28 | 18.6 |
| Lower Shell Forging | 37.0 | 2.199 | 1.214 | -30 | 34 | 44.92 | 48.9 |
| using S/C data[b] | 13.3 | 2.199 | 1.214 | -30 | 34[c] | 16.15 | 20.2 |
| Weld Metal WF-562 | 41.0 | 2.199 | 1.214 | 40 | 49.77 | 49.77 | 139.5 |
| using S/C data[b] | **16.7** | 2.199 | 1.214 | 40 | **20.27** | **20.27** | **80.5** |
| 48 EFPY | | | | | | | |
| Upper Shell Forging | 20.0 | 3.298 | 1.313 | -30 | 26.26 | 26.26 | 22.5 |
| Lower Shell Forging | 37.0 | 3.298 | 1.313 | -30 | 34 | 48.58 | 52.6 |
| using S/C data[b] | 13.3 | 3.298 | 1.313 | -30 | 34[c] | 17.46 | 21.5 |
| Weld Metal WF-562 | 41.0 | 3.298 | 1.313 | 40 | 53.83 | 53.83 | 147.7 |
| using S/C data[b] | **16.7** | 3.298 | 1.313 | 40 | **21.93** | **21.93** | **83.9** |

NOTES:
(a) FF (Fluence factor) = $f^{(0.28 - 0.10*\log f)}$
(b) Calculated using a CF based n surveillance capsule data per Regulatory Guide 1.99, Revision 2, Position 2.
(c) Double margin is used here due to the base metal surveillance capsule data exceeding the one sigma criteria from the credibility evaluation. See page B-9.

# APPENDIX C

## BYRON/BRAIDWOOD FLUENCE METHODOLOGY JUSTIFICATION AND TIME-DEPENDENT CAPSULE FLUENCE VALUES

1 - Fluence Methodology Justification

The fast neutron exposure methodology documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" is consistent with the requirements of Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" and makes use of neutron transport cross-sections derived from the ENDF/B-VI data base. The exposure evaluations documented in WCAPs 13880, 14064, 14241, and 14228 for the Byron Units 1 & 2 and Braidwood Units 1 & 2 pressure vessels were completed prior to the release of the ENDF/B-VI based Light Water Reactor neutron transport cross-section library. Consequently the neutron transport calculations performed as an integral part of these evaluations were based on the currently available ENDF/B-IV based cross-section library. In all respects other than the ENDF/B-VI vs ENDF/B-IV cross-section issue, the methodology applied to the Byron Units 1 & 2 and Braidwood Units 1 & 2 fluence evaluations was identical to the approved methodology described in WCAP-14040-NP-A.

Commonwealth Edison plans to re-evaluate capsule and vessel fluence estimates utilizing ENDF/B-VI neutron cross-section libraries in accordance with WCAP-14040-NP-A at the next scheduled capsule withdrawal for each set of units (capsule W for Byron Unit 1 at BIR08 in November 1997 and capsule W for Braidwood Unit 1 at AIR07 in September 1998), and for all subsequent capsule withdrawals, and proposes to integrate data pursuant to 10 CFR 50 Appendix H. This will replace estimates previously performed using a combination of ENDF/B-IV transport cross-sections and ENDF/B-V dosimetry cross-sections. Since this re-evaluation will impact the manner in which materials data are utilized, and, therefore constitutes a change in PTLR methodologies, all revised values of ΔRT resulting from the new fluence values along with an evaluation of their impact on pressure-temperature limits will be submitted to NRC for review and approval.

In addition to the methodology upgrade discussed in the preceding paragraph, the fluence updates for Byron Units 1 & 2 and Braidwood Units 1 & 2 will also include an evaluation of low leakage fuel management instituted at all four units. A qualitative examination of the loading patterns used at Byron Units 1 & 2 and Braidwood Units 1 & 2 indicates that accounting for the flux reduction brought about by the incorporation of low leakage fuel management will compensate for increases in projected fluence that may be introduced by the methods changes. The net effect of methods upgrades and low leakage fuel management on projected vessel fluence is, therefore, anticipated to be very small and may result in an overall reduction in fluence relative to that reported in WCAPs 13880, 14064, 14241, and 14228.

Based on the relatively small changes that are anticipated from updating the neutron fluence evaluations from those reported in WCAPs 13880, 14064, 14241, and 14228 to the approved methodology described in WCAP-14040-NP-A, including the impact of low leakage fuel management, coupled with the low sensitivity to irradiation damage exhibited by the materials comprising the Byron Units 1 & 2 and Braidwood Units 1 & 2 reactor pressure vessels, the use of the previously documented fluence values is justified until the update to the ENDF/B-VI based methodology is completed for each unit.

2 - Time Dependent Surveillance Capsule Fluences

Based on the documentation provided in WCAPs 13880, 14064, 14241, and 14228, it is noted that the last surveillance capsule withdrawal for Byron Units 1 & 2 and Braidwood Units 1 & 2 was at 5.64, 4.63, 4.23, and 4.21 effective full power years, respectively. Projection of fluence levels at the surveillance capsule locations for times beyond those withdrawal dates are needed in order to establish appropriate withdrawal schedules for the remaining capsules comprising the Reactor Vessel Surveillance Program for each of the units. These Best Estimate projections are provided in Tables C-1 through C-4 for Byron Units 1 & 2 and Braidwood Units 1 & 2, respectively. These projections are based on the assumption that the best estimate neutron flux averaged over the total irradiation time for each unit would remain applicable for the remainder of plant lifetime.

3 - Comparison of Irradiation Environments

Byron Units 1 and 2 as well as Braidwood Units 1 and 2 are Westinghouse reactors employing a reactor internals design using partially circumferential neutron pads. The surveillance capsules holding the materials test specimens are mounted on the outer radius of the neutron pads in the downcomer region between the pressure vessel wall and the thermal shield. Thus, the surveillance specimens are mounted behind the full thickness of the neutron pad. The location of the maximum fluence on each of the respective pressure vessels also occurs at an azimuthal angle behind the neutron pad. The geometry of the neutron pads, surveillance capsules and associated support structure, and the pressure vessel itself are all modeled explicitly in the neutron transport calculations performed for the Byron and Braidwood reactors.

The design of the Byron and Braidwood reactor internals includes former plates at several axial intervals spanning the radial distance between the external boundary of the baffle plates and the inner radius of the core barrel. Due to the shape of the perimeter of the reactor core, the radial extent of the former plates varies significantly with azimuthal angle. The presence of these former plates can have a localized effect on any dosimetry or materials test specimens that may be located directly in line with these steel plates. Studies have been performed to estimate the effect of the former plates on the irradiation conditions within the surveillance capsules. The results of these studies indicate that the formers have the largest effect on high threshold reactions such as Cu (n,$\alpha$), Fe (n,p), and Ni (n,p) and a minimal effect on the lower threshold reactions and exposure parameters such as U-238 (n,f), Np-237 (n,f), Fluence (E > 1.0 MeV), and dpa. The maximum effects noted were 11% for Cu, 10% for Fe, 9% for Ni, 7%

1.0 MeV), and dpa. The maximum effects noted were 11% for Cu, 10% for Fe, 9% for Ni, 7% for U-238, 1% for Np-237, 3% for Fluence (E > 1.0 MeV), and 1% for dpa. Each of these percentages represent a reduction in the calculated value at a location directly in line with the formerplates. In the case of the Byron and Braidwood units, neutron dosimeters are dispersed axially within the capsules such that the effects introduced by the presence of the former plates are minimized.

Since these four reactors were designed as identical units, the plant specific geometries in the vicinity of the surveillance capsules tend to result in radiation environments at the capsule positions that are almost identical both quantitatively and qualitatively. In all units, the capsules are designed to minimize the impact of gamma ray heating and, thus, maintain the irradiation temperature of the test specimens close to the temperature of the coolant in the downcomer region. Likewise, the temperature of the pressure vessel wall at the clad/base metal interface is also maintained very close to the downcomer coolant temperature, thus, providing compatibility between the test specimen irradiation temperatures and that of the pressure vessel wall.

To date eight surveillance capsules have been withdrawn from the same symmetric 31.5° azimuthal location at the four Byron and Braidwood reactors (two from each unit). Comparisons of the neutron dosimetry evaluations performed for these 8 surveillance capsules provide an indication of the similarity in the irradiation environments for each of these capsules.

a) Damage Rate

The following tabulation provides the neutron flux (E > 1.0 MeV) and iron atom displacement rate (dpa/sec) experienced by each of the surveillance capsules withdrawn from Buron Units 1 and 2 and Braidwood Units 1 and 2. The damage rates represent an average over the total irradiation period experienced by the respective capsules.

| Capsule | Flux (E > 1.0 MeV) [n/cm$^2$-sec] | Displacement Rate [dpa/sec] | Irradiation Time [efpy] |
|---|---|---|---|
| Byron 1 - U | 9.86e+10 | 1.89e-10 | 1.15 |
| Byron 2 - U | 1.10e+11 | 2.06e-10 | 1.15 |
| Braidwood 1 - U | 1.10e+11 | 2.10e-10 | 1.10 |
| Braidwood 2 - U | 1.08e+11 | 2.07e-10 | 1.15 |
| Byron 1 - X | 8.11e+10 | 1.55e-10 | 5.64 |
| Byron 2 - W | 8.27e+10 | 1.53e-10 | 4.64 |
| Braidwood 1 - X | 8.57e+10 | 1.57e-10 | 4.23 |
| Braidwood 2 - X | 8.46e+10 | 1.56e-10 | 4.22 |

An examination of this tabulation shows that in terms of neutron flux (E > 1.0 MeV) the range of damage rate extends from 8.11e+10 to 1.10e+11 and in terms of iron atom displacement rates the corresponding range extends from 1.53e-10 to 2.10e-10. The total range of damage rates for these capsules falls within approximately a factor of 1.4. Furthermore, there is no systematic difference among any of the units.

b) Spectral Balance

An indication of the differences in the energy distribution of neutrons at the surveillance capsule locations can be obtained via a comparison of the ratio of [dpa/sec]/[flux (E > 1.0 MeV)] at the respective capsule locations. A comparison of this type for the Byron and Braidwood surveillance capsules is provided as follows:

| Capsule | [dpa/Flux] | Irradiation Time [efpy] |
|---|---|---|
| Byron 1 - U | 1.92e-21 | 1.15 |
| Byron 2 - U | 1.87e-21 | 1.15 |
| Braidwood 1 - U | 1.91e-21 | 1.10 |
| Braidwood 2 - U | 1.92e-21 | 1.15 |
| Byron 1 - X | 1.91e-21 | 5.64 |
| Byron 2 - W | 1.85e-21 | 4.64 |
| Braidwood 1 - X | 1.83e-21 | 4.23 |
| Braidwood 2 - X | 1.84e-21 | 4.22 |

An examination of this data table shows that the spectral indices as expressed by the ratio of iron displacement rate to neutron flux (E > 1.0 MeV) varies by less than approximately 5% over the entire range of capsules included in the data set. Thus, from a spectrum balance viewpoint the Byron and Braidwood irradiation conditions are essentially identical.

c) gamma heating

Gamma heating effects the irradiation environment of the Byron and Braidwood reactor vessels in a similar fashion. At the surveillance capsule and reactor vessel locations the gamma ray heating is due primarily to secondary gamma rays induced by inelastic scattering and neutron capture in local materials. Since the secondary gamma ray production is a function of the neutron energy spectrum which, as described above, is essentially identical for these reactors, it follows that the gamma ray heating for these reactors is also very similar.

Furthermore, in the design of the Westinghouse surveillance capsules, the impact of internal heat generation is small and the specimen temperatures are maintained very close to the downcomer water temperature.

d) Irradiation Temperature

The vessel inlet temperatures for the Byron and Braidwood units for each of their operating fuel cycles are listed as follows:

| Fuel Cycle | Byron 1 | Byron 2 | Braidwood 1 | Braidwood 2 |
|---|---|---|---|---|
| 1 | 557 | 557 | 557 | 557 |
| 2 | 557 | 551 | 551 | 551 |
| 3 | 551 | 551 | 551 | 551 |
| 4 | 551 | 551 | 551 | 551 |
| 5 | 551 | 551 | 551-554 | 551 |
| 6 | 551 | 551 | 554 | 551 |
| 7 | 551 | 551 | | |
| 8 | 551 | | | |

Note: Byron Units 1 & 2, and Braidwood Units 1 & 2 capsules U were withdrawn at the end of cycle 1. Byron 1 capsule X was withdrawn at the end of cycle 5 and Byron 2 capsule W was withdrawn at the end of cycle 4. Braidwood 1 & 2 capsules X were both withdrawn at the end of cycle 4.

For Braidwood Units 1 and 2, the vessel inlet temperatures were identical from one unit to the other during the cycles that the capsules in question were being irradiated. Thus, the irradition conditions were identical.

For Byron Units 1 and 2, on the other hand, there exists a 6°F temperature difference in cycle 2. Since Byron 2 capsule W was pulled at the end of the fourth cycle, the time weighted temperature difference over the four cycles that the capsules (i.e. Capsules X and W) were irradiated is 1.5°F, or conservatively 2°F. For the purpose of evaluating surveillance weld data credibility in Table A-4 and for the purpose of evaluating weld chemistry factor normalized to the Byron Unit 1 and Byron Unit 2 reactor vessels in Table 4 (page 11), the effects of this adjustment to the measured $\Delta RT_{NDT}$ by 2°F (conservatively applied to raising the lower temperature Byron 2 capsule W measured $\Delta RT_{NDT}$) is very small. The effect of the 2°F adjustment , and even the effect of applying the entire single cycle 6°F temperature difference to the Byron 2 capsule W measured $\Delta RT_{NDT}$, is completely compensated for with the application of the conservative ratio of 3.0, relative to the actual ratio of 2.5. Temperature difference and chemistry factor ratios will be re-evaluated at all future scheduled capsule evaluations.

Table C-1

BEST ESTIMATE FAST NEUTRON FLUENCE (E > 1.0 MeV) PROJECTIONS
AT SURVEILLANCE CAPSULE LOCATIONS - BYRON UNIT 1

| Irradiation Time | Fluence [n/cm$^2$] | | Lead Factor | |
|---|---|---|---|---|
| [EFPY] | 31.5 Caps | 29.0 Caps | 31.5 Caps | 29.0 Caps |
| 5.64 | 1.443e+19 | 1.365e+19 | 3.79 | 3.58 |
| 8.00 | 2.046e+19 | 1.935e+19 | 3.79 | 3.58 |
| 10.00 | 2.558e+19 | 2.419e+19 | 3.79 | 3.58 |
| 12.00 | 3.070e+19 | 2.902e+19 | 3.79 | 3.58 |
| 14.00 | 3.581e+19 | 3.386e+19 | 3.79 | 3.58 |
| 16.00 | 4.093e+19 | 3.870e+19 | 3.79 | 3.58 |
| 18.00 | 4.604e+19 | 4.353e+19 | 3.79 | 3.58 |
| 20.00 | 5.116e+19 | 4.837e+19 | 3.79 | 3.58 |
| 22.00 | 5.628e+19 | 5.321e+19 | 3.79 | 3.58 |
| 24.00 | 6.139e+19 | 5.804e+19 | 3.79 | 3.58 |
| 26.00 | 6.651e+19 | 6.288e+19 | 3.79 | 3.58 |
| 28.00 | 7.162e+19 | 6.772e+19 | 3.79 | 3.58 |
| 30.00 | 7.674e+19 | 7.256e+19 | 3.79 | 3.58 |
| 32.00 | 8.186e+19 | 7.739e+19 | 3.79 | 3.58 |

Table C-2

BEST ESTIMATE FAST NEUTRON FLUENCE (E > 1.0 MeV) PROJECTIONS
AT SURVEILLANCE CAPSULE LOCATIONS - BYRON UNIT 2

| Irradiation Time [EFPY] | Fluence [n/cm$^2$] 31.5 Caps | 29.0 Caps | Lead Factor 31.5 Caps | 29.0 Caps |
|---|---|---|---|---|
| 4.63 | 1.235e+19 | 1.154e+19 | 3.89 | 3.64 |
| 6.00 | 1.598e+19 | 1.494e+19 | 3.89 | 3.64 |
| 8.00 | 2.131e+19 | 1.992e+19 | 3.89 | 3.64 |
| 10.00 | 2.664e+19 | 2.491e+19 | 3.89 | 3.64 |
| 12.00 | 3.197e+19 | 2.989e+19 | 3.89 | 3.64 |
| 14.00 | 3.730e+19 | 3.487e+19 | 3.89 | 3.64 |
| 16.00 | 4.262e+19 | 3.985e+19 | 3.89 | 3.64 |
| 18.00 | 4.795e+19 | 4.483e+19 | 3.89 | 3.64 |
| 20.00 | 5.328e+19 | 4.981e+19 | 3.89 | 3.64 |
| 22.00 | 5.861e+19 | 5.479e+19 | 3.89 | 3.64 |
| 24.00 | 6.394e+19 | 5.977e+19 | 3.89 | 3.64 |
| 26.00 | 6.927e+19 | 6.475e+19 | 3.89 | 3.64 |
| 28.00 | 7.459e+19 | 6.973e+19 | 3.89 | 3.64 |
| 30.00 | 7.992e+19 | 7.472e+19 | 3.89 | 3.64 |
| 32.00 | 8.525e+19 | 7.970e+19 | 3.89 | 3.64 |

Table C-3

BEST ESTIMATE FAST NEUTRON FLUENCE (E > 1.0 MeV) PROJECTIONS
AT SURVEILLANCE CAPSULE LOCATIONS - BRAIDWOOD UNIT 1

| Irradiation Time [EFPY] | Fluence [n/cm$^2$] | | Lead Factor | |
|---|---|---|---|---|
| | 31.5 Caps | 29.0 Caps | 31.5 Caps | 29.0 Caps |
| 4.23 | 1.193e+19 | 1.105e+19 | 4.02 | 3.73 |
| 6.00 | 1.690e+19 | 1.565e+19 | 4.02 | 3.73 |
| 8.00 | 2.254e+19 | 2.087e+19 | 4.02 | 3.73 |
| 10.00 | 2.817e+19 | 2.609e+19 | 4.02 | 3.73 |
| 12.00 | 3.380e+19 | 3.130e+19 | 4.02 | 3.73 |
| 14.00 | 3.944e+19 | 3.652e+19 | 4.02 | 3.73 |
| 16.00 | 4.507e+19 | 4.174e+19 | 4.02 | 3.73 |
| 18.00 | 5.070e+19 | 4.696e+19 | 4.02 | 3.73 |
| 20.00 | 5.634e+19 | 5.217e+19 | 4.02 | 3.73 |
| 22.00 | 6.197e+19 | 5.739e+19 | 4.02 | 3.73 |
| 24.00 | 6.761e+19 | 6.261e+19 | 4.02 | 3.73 |
| 26.00 | 7.324e+19 | 6.783e+19 | 4.02 | 3.73 |
| 28.00 | 7.887e+19 | 7.304e+19 | 4.02 | 3.73 |
| 30.00 | 8.451e+19 | 7.826e+19 | 4.02 | 3.73 |
| 32.00 | 9.014e+19 | 8.348e+19 | 4.02 | 3.73 |

Table C-4

BEST ESTIMATE FAST NEUTRON FLUENCE (E > 1.0 MeV) PROJECTIONS AT SURVEILLANCE CAPSULE LOCATIONS - BRAIDWOOD UNIT 2

| Irradiation Time | Fluence [n/cm$^2$] | | Lead Factor | |
|---|---|---|---|---|
| [EFPY] | 31.5 Caps | 29.0 Caps | 31.5 Caps | 29.0 Caps |
| 4.21 | 1.163e+19 | 1.072e+19 | 4.02 | 3.70 |
| 6.00 | 1.656e+19 | 1.526e+19 | 4.02 | 3.70 |
| 8.00 | 2.208e+19 | 2.034e+19 | 4.02 | 3.70 |
| 10.00 | 2.760e+19 | 2.543e+19 | 4.02 | 3.70 |
| 12.00 | 3.312e+19 | 3.051e+19 | 4.02 | 3.70 |
| 14.00 | 3.864e+19 | 3.560e+19 | 4.02 | 3.70 |
| 16.00 | 4.416e+19 | 4.068e+19 | 4.02 | 3.70 |
| 18.00 | 4.968e+19 | 4.577e+19 | 4.02 | 3.70 |
| 20.00 | 5.520e+19 | 5.085e+19 | 4.02 | 3.70 |
| 22.00 | 6.072e+19 | 5.594e+19 | 4.02 | 3.70 |
| 24.00 | 6.625e+19 | 6.102e+19 | 4.02 | 3.70 |
| 26.00 | 7.177e+19 | 6.611e+19 | 4.02 | 3.70 |
| 28.00 | 7.729e+19 | 7.119e+19 | 4.02 | 3.70 |
| 30.00 | 8.281e+19 | 7.628e+19 | 4.02 | 3.70 |
| 32.00 | 8.833e+19 | 8.136e+19 | 4.02 | 3.70 |