Westinghouse Non-Proprietary Class 3

WCAP-14824
Revision 1

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

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

P. A. Grendys

April 1997

Work Performed Under Shop Order CPEP-139

Prepared by the Westinghouse Electric Corporation
for the Commonwealth Edison Company

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C. H. Boyd, Manager
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PREFACE

This report has been technically reviewed and verified by:

T. J. Laubham

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1 INTRODUCTION

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

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

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. 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]. Credible surveillance data is available for two capsules (Capsules U and X) for Byron Unit 1. 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:

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

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:

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

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, 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 ΔT (temperature) developed during cooldown results in a higher 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 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 K_{Ia} for the 1/4T crack during heatup is lower than the 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

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:

$$ART = Initial RT_{NDT} + \Delta RT_{NDT} + Margin \tag{3}$$

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.

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

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

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

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

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 ΔRT_{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 x 10^{18} n/cm^2. This fluence value was calculated from the surveillance Capsule X analysis presented in WCAP-13880[9].

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

The Ratio Procedure, as documented in paragraph (c)(2)(ii)(B) of 10 CFR Part 50.61, was used to adjust the measured values of ΔRT_{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. Sample calculations to determine the ART values for the Weld Metal 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		
Letter Report FDRT/ SRPLO-009(94) January 1994	0.03	0.75		
	0.05	0.73		
	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	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				
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:		0.057	0.614 --->	**0.06**	**0.61**	**Best Est. CF = 82**
Standard Deviation:		0.043	0.095			**Byron 1 & 2 Ratio = 3.0**

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.

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	0.06	0.61	82.0	-30

NOTES:

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

(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. ΔRT_{NDT}		FF* ΔRT_{NDT}	FF^2
Inter. Shell Forging 5P-5933 (Tangential)	U	3.72×10^{18}	0.727	0		0	0.529
	X	1.39×10^{19}	1.091	30		32.73	1.19
Inter. Shell Forging 5P-5933 (Axial)	U	3.72×10^{18}	0.727	0		0	0.529
	X	1.39×10^{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	3.72×10^{18}	0.727	0	0	0.00	0.529
	X	1.39×10^{19}	1.091	35	105[e]	114.56	1.19
Byron 2 Weld Metal WF-447[c]	U	3.996×10^{18}	0.746	0	0	0.00	0.557
	W	1.[illegible]11x10^{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 ΔRT_{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 ΔRT_{NDT} values were obtained from the surveillance Capsule W analysis (WCAP-14064[11]) using the ENDF/B-V scattering cross sections.

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

(e) Adjusted ΔRT_{NDT} per Ratio Procedure of 10 CFR 50.61. Ratio = 3.0. See Table 2.

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for the plate, and 8.5°F for the plate (half the value) when surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule is not used, and equal to 14°F when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

TABLE 5

Calculation of Adjusted Reference Temperatures (ART) at 12 EFPY for the Limiting Byron Unit 1 Reactor Vessel Material

Intermediate Shell Forging 5P-5933 (based on credible surveillance capsule data)

Parameter	Values	
Operating Time	12 EFPY	
Material	Intermediate Shell Forging 5P-5933	
Location	1/4T	3/4T
Chemistry Factor, CF (°F)	19.0	19.0
Fluence, f (10^{19} n/cm^2) (b)	0.486	0.175
Fluence Factor, FF	0.799	0.538
ΔRT_{NDT} = CF x ff (°F)	15.2	10.2
Initial RT_{NDT}, I (°F)	40	40
Margin, M (°F)	15.2	10.2
Adjusted Reference Temperature (ART), (°F) per Regulatory Guide 1.99, Revision 2	**70**	**60**

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.

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])	92	58
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. This approved 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. 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

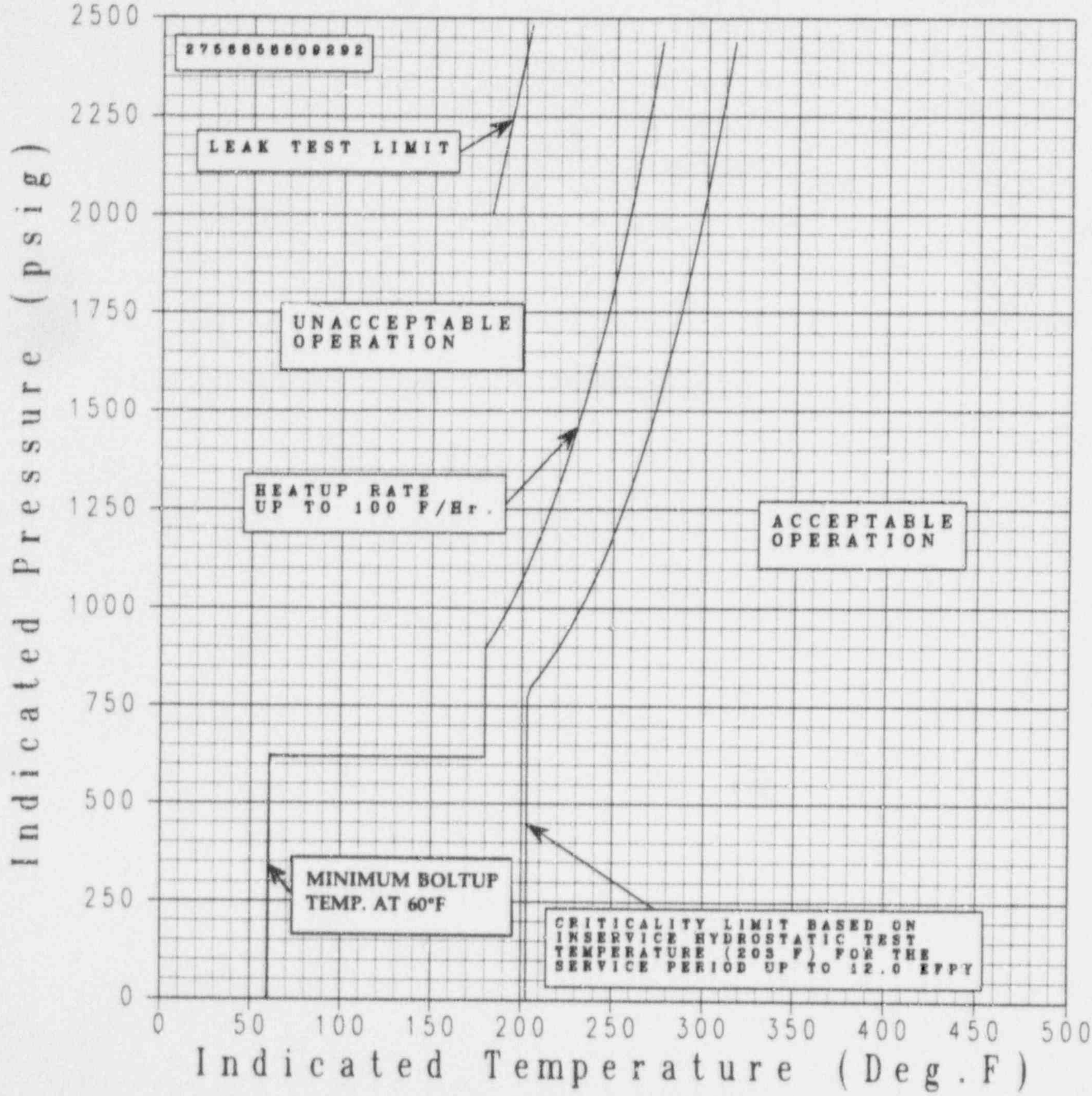


FIGURE 1 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors)

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

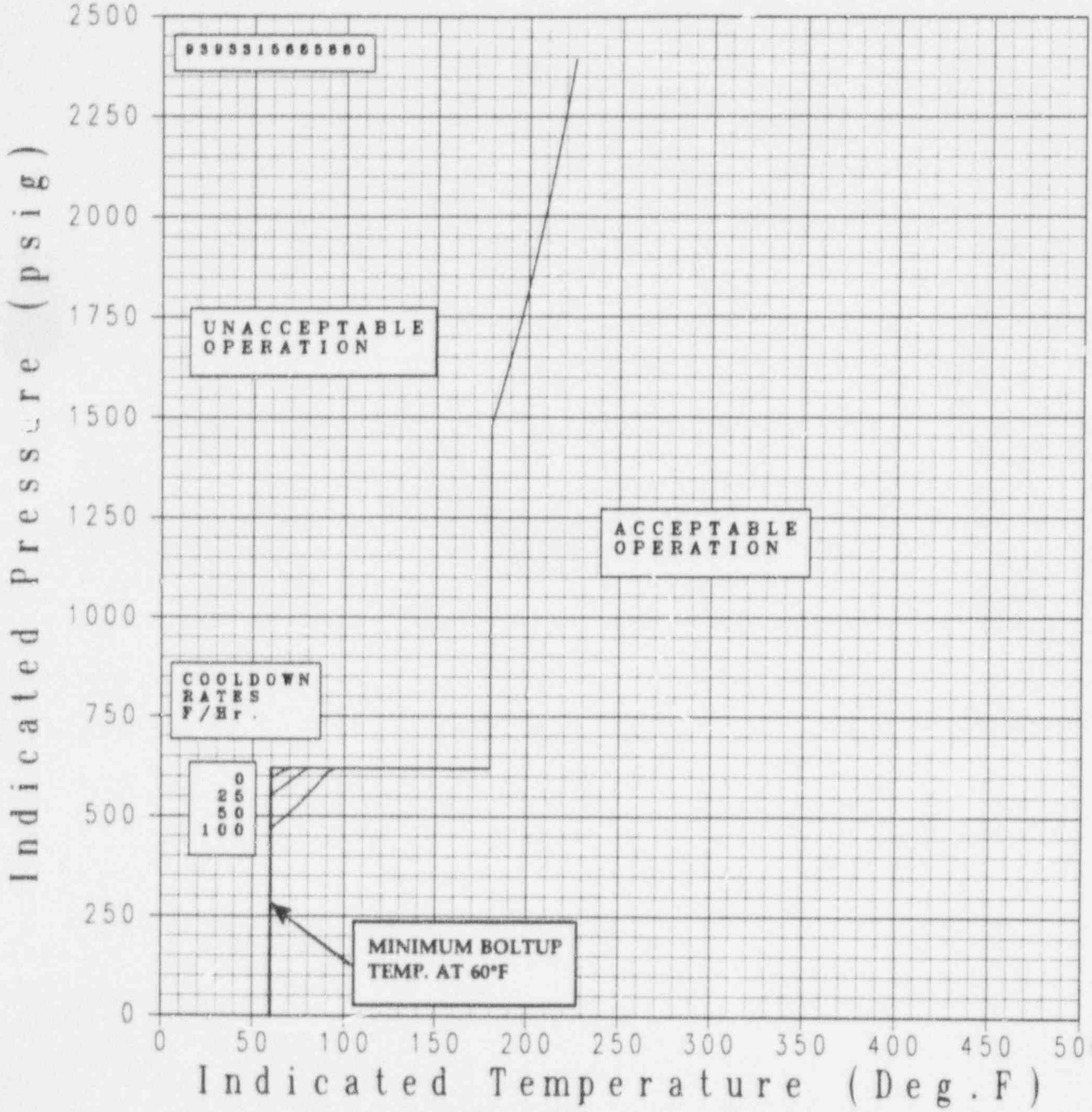


FIGURE 2 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors)

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

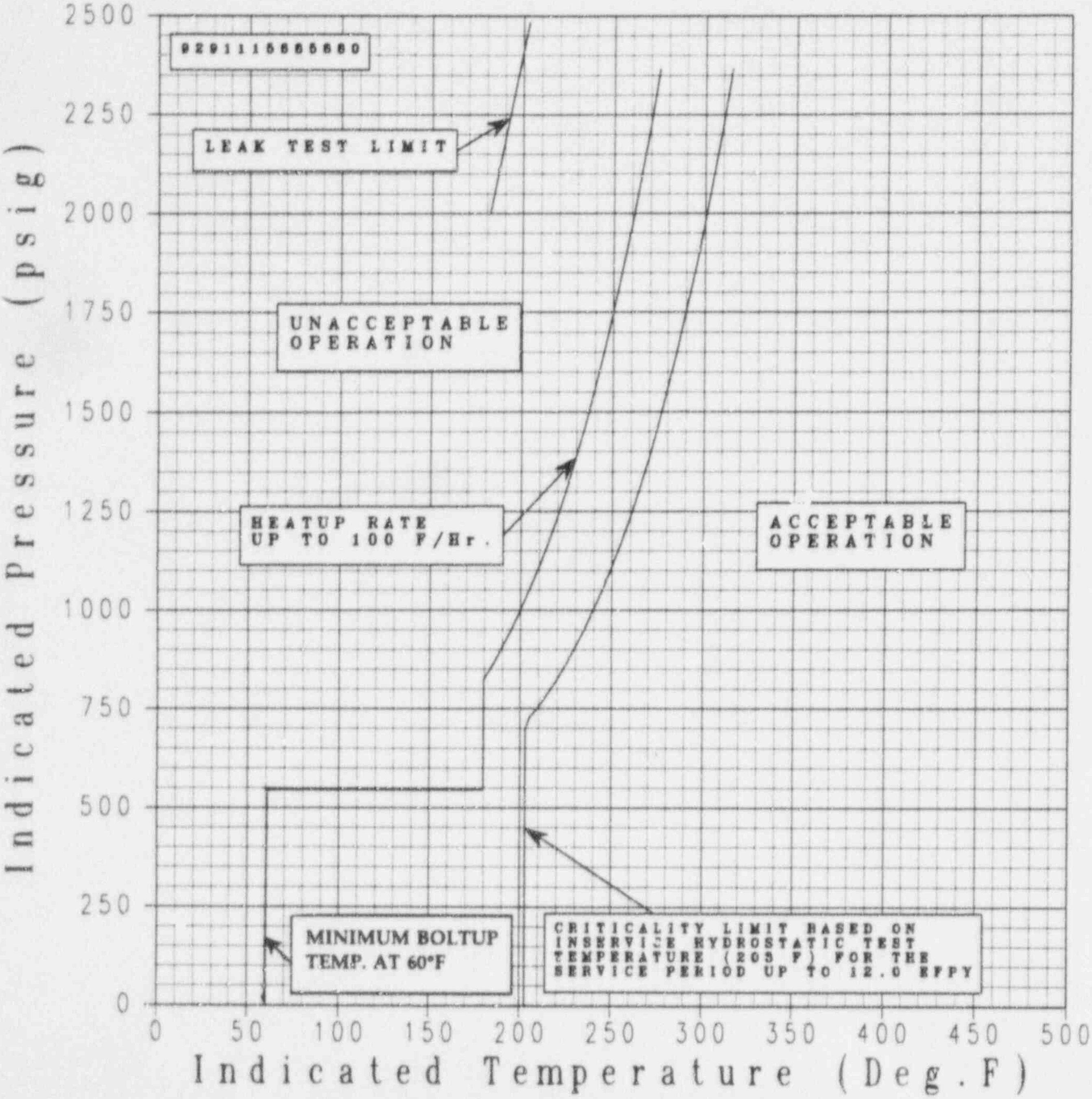


FIGURE 3 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors; Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and the Reactor Vessel Beltline Region)

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

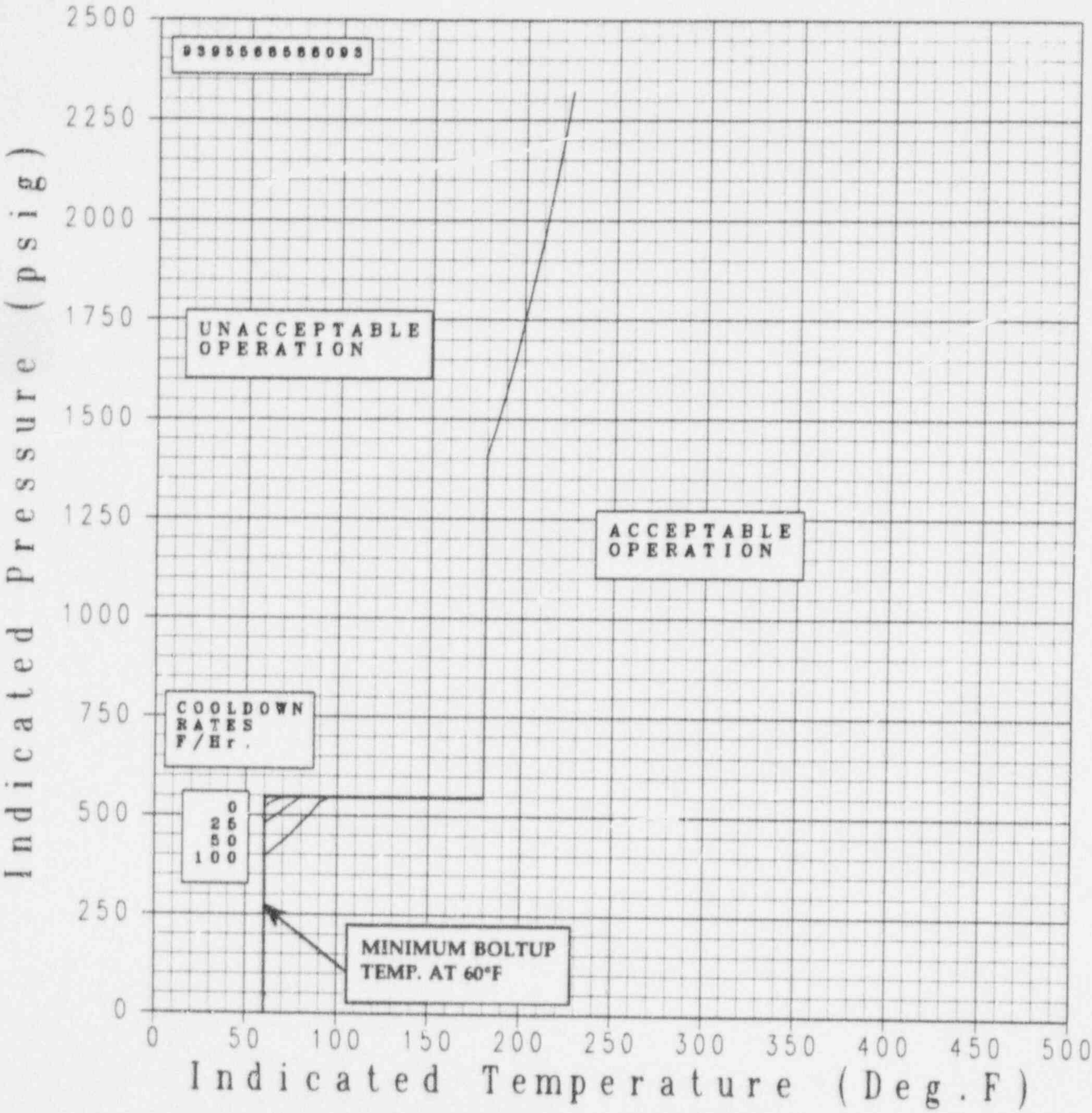


FIGURE 4 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 12 EFPY (Without Margins for Instrumentation Errors; Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and the Reactor Vessel Beltline Region)

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	152	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	533	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)

6 REFERENCES

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

2 Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

3 WCAP-9517, "Commonwealth Edison Co. Byron Station Unit 1 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, July 1979.

4 WCAP-10398, "Commonwealth Edison Co. Byron Station Unit 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, December 1983.

5 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

6 1992 Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, Appendix G, "Vessels".

7 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

8 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

9 WCAP-13880, "Analysis of Capsule X from the Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", P. A. Peter, et al., January 1994.

10 WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Reevaluation", E. P. Lippincott, April 1994.

11 WCAP-14064, "Analysis of Capsule W from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program", P. A. Peter, et al., November 1994.

12 WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves", W. S. Hazelton, et al., April 1975.

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

14 Babcock & Wilcox drawing numbers 184557E, Rev. 2; 185266E, Rev. 2; 185297E, Rev. 2; 185328E, Rev. 2; "Reactor Vessel Longitudinal Section".

15 Nuclear Safety Advisory Letter, NSAL-93-005A, "Cold Overpressure Mitigation System (COMS) Nonconservatism", L. R. Hardwick and H. A Sepp, 3/10/93.

16 WCAP-14063, "Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation", P. A. Peter, November 1994.

17 WCAP-13881, "Evaluation of Pressurized Thermal Shock for Byron Unit 1", P. A. Peter, January 1994.

18 WCAP-14054, "Evaluation of Pressurized Thermal Shock for Byron Unit 2", P. A. Peter, August 1995.

19 WCAP-14242, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1", P. A. Peter, March 1995.

20 WCAP-14229, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", P. A. Peter, March 1995.

21 WCAP-11651, "Analysis of Capsule U From The Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, et al., November 1987.

22 WCAP-12431, "Analysis of Capsule U from the Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program", E. Terek, et al., October 1989.

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), the initial RT_{NDT} of the welds differ. However, the surveillance materials have performed similarly, and it is shift data that is used in the integration of

data. As can be seen in Table 4 (page 10 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°
1.290×10^{19}	1.947×10^{19}	2.159×10^{19}	1.705×10^{19}	1.939×10^{19}

Byron Unit 2

0°	15°	25°	35°	45°
1.353×10^{19}	1.979×10^{19}	2.192×10^{19}	1.772×10^{19}	2.026×10^{19}

10. *What are the vessel inlet temperatures?*

    Byron 1: 558.4°F

    Byron 2: 558.4°F

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 1) all the surveillance capsule data (base metal & weld metal) for Byron Unit 1, and 2) weld metal (only) for Byron Unit 2. The credibility determination will use the Byron Unit 2 weld metal data for the Byron 1 heatup/cooldown pressure temperature limit curves. Therefore, it must be determined to be credible.

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: Intermediate shell forging 49D329/49C297-1-1
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 calculated RT_{PTS} values presented in WCAP-13881 (Byron 1 PTS) and the information provided in the Byron Unit 2 material selection documents, 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 ΔRT_{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.

TABLE A-2[c]

Byron Units 1 & 2 Surveillance Capsule Data Calculation of Best-Fit Line as Described in Position 2.1 of Regulatory Guide 1.99, Revision 2

Material	Capsule	f[a]	FF[b] (x)	Measured ΔRT_{NDT} (y)	FF x ΔRT_{NDT} (xy)	FF^2 (x^2)
Byron Unit 1 Inter. Shell Forging 5P-5933 (Axial)	U	3.72×10^{18}	0.727	0	0	0.529
	X	[illegible]	1.091	30	32.73	1.190
Byron Unit 1 Inter. Shell Forging 5P-5933 (Tangential)	U	3.72×10^{18}	0.727	0	0	0.529
	X	1.39×10^{19}	1.091	30	32.73	1.190
	$\Sigma^{n}_{i=1}$		3.636	60	65.46	3.44
Byron Unit 1 Weld Metal	U	3.72×10^{18}	0.727	0	0.00	0.529
	X	1.39×10^{19}	1.091	35	38.185	1.190
Byron Unit 2 Weld Metal	U	3.996×10^{18}	0.746	0	0.00	0.557
	W	1.211×10^{19}	1.053	30	31.600	1.110
	$\Sigma^{n}_{i=1}$		[illegible]	65	69.785	3.386

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 ΔRT_{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.

Per the 27th Edition of the CRC Standard Mathematical Tables (page 497), for a straight line fit by the method of least squares, the values b_o and b_1 are obtained by solving the normal equations:

$$n\, b_o + b_1\, \Sigma x_i = \Sigma y_i$$

and

$$b_o\, \Sigma x_i + b_1\, \Sigma x_i^2 = \Sigma x_i y_i$$

These equations can be re-written as follows:

$$\sum_{i=1}^{n} y_i = an + b\sum_{i=1}^{n} x_i$$

and

$$\sum_{i=1}^{n} x_i y_i = a\sum_{i=1}^{n} x_i + b\sum_{i=1}^{n} x_i^2$$

<u>Byron 1 & 2 Weld Metal:</u>
Based on the data provided in Table A-2 the equations become:

1. 65.0 = 4a + 3.617b or a = 16.25 - 0.9043b and
2. 69.785 = 3.617a + 3.386b

Thus, by substituting Eq. 1 into Eq. 2, b = 95.71. Now, enter b (= 95.71) into Eq. 1 and a = -70.30. Therefore, the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 95.71\,(X) - 70.30$$

The error in predicting a value Y corresponding to a given X value is: e = Y - Y'

<u>Byron 1 Base Metal:</u>
Based on the data provided in Table A-2 the equations become:

1. 60.0 = 4a + 3.636b or a = 15.0 - 0.909b and
2. 65.46 = 3.636a + 3.44b

Thus, by substituting Eq. 1 into Eq. 2, b = 80.89. Now, enter b (= 80.89) into Eq. 1 and a = -58.53. Therefore, the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 80.89\,(X) - 58.53$$

The error in predicting a value Y corresponding to a given X value is: e = Y - Y'

TABLE A-3

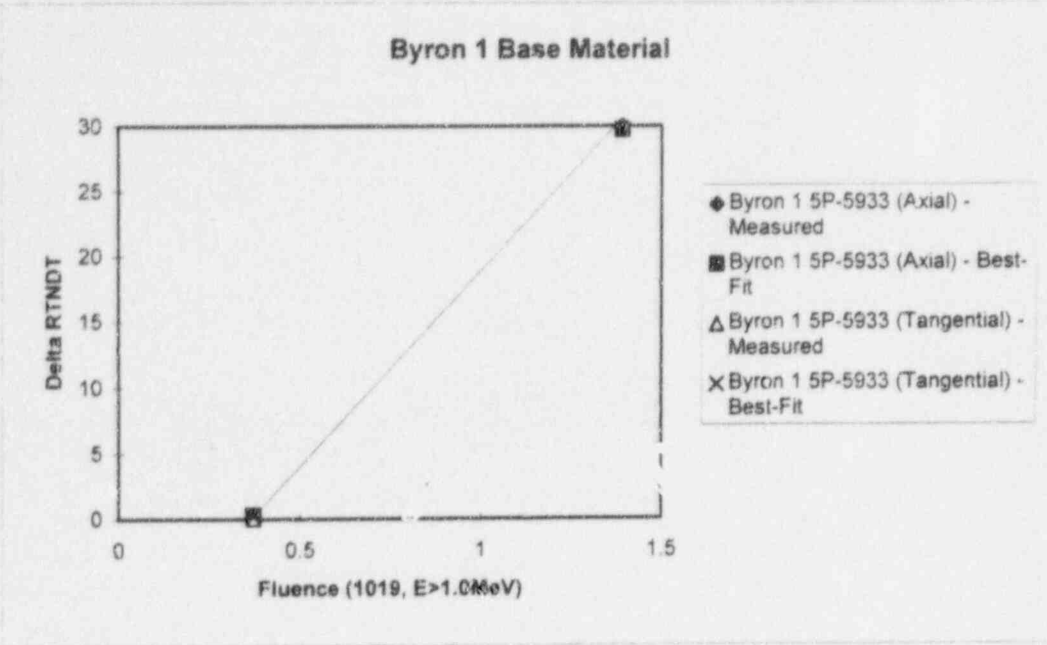
Best Fit Evaluation for Byron 1 & 2 Surveillance Materials

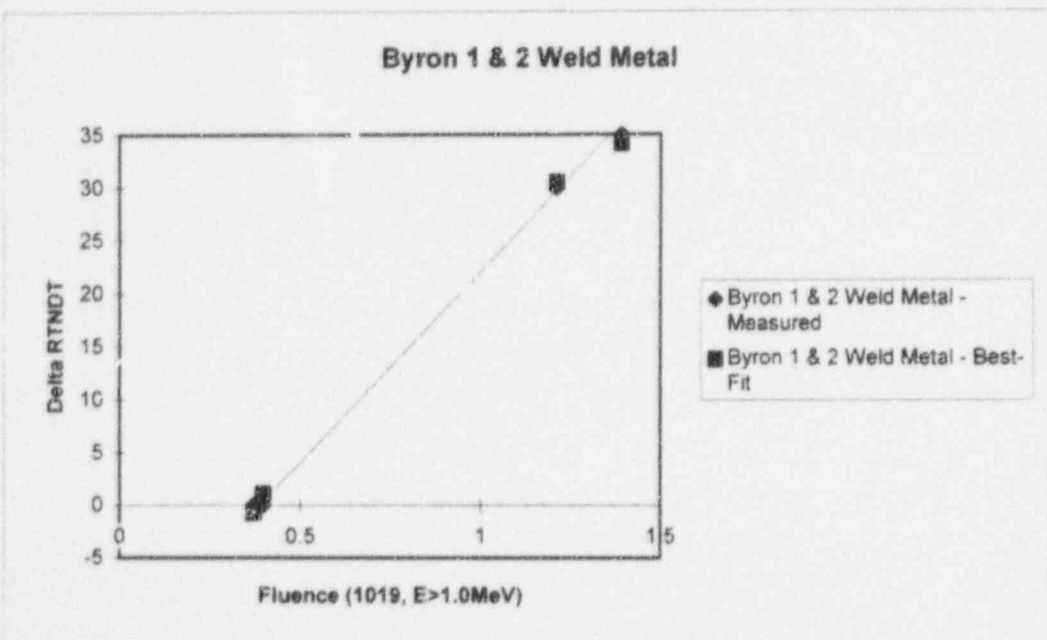
Base Material	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Byron 1 & 2 Weld Metal	0.727	0	-0.72	-0.00
	1.091	35	-0.00	35.00
	0.746	0	-0.00	0.00
	1.053	30	-0.00	30.00
Byron Unit 1 Inter. Shell Forging 5P-5933 (Axial)	0.727	0	0.28	-0.28
	1.091	30	29.72	0.28
Byron Unit 1 Inter. Shell Forging 5P-5933 (Tangential)	0.727	0	0.28	-0.28
	1.091	30	29.72	0.28

<u>Weld Metal:</u>
The scatter of ΔRT_{NDT} values 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.

<u>Base Material:</u>
The scatter of ΔRT_{NDT} values about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 17°F for base metal. As shown above, the error is within 17°F of the best-fit line. Therefore, this criteria is met for the Byron Unit 1 surveillance base metal.

See the following scatter plots for the Byron Unit 1 base material and the Byron 1 and 2 weld metal.


Byron 1 Base Material
Delta RTNDT
Fluence (1019, E>1.0MeV)
0
0.5
1
1.5
0
5
10
15
20
25
30
Byron 1 5P-5933 (Axial) - Measured
Byron 1 5P-5933 (Axial) - Best-Fit
Byron 1 5P-5933 (Tangential) - Measured
Byron 1 5P-5933 (Tangential) - Best-Fit


Byron 1 & 2 Weld Metal
Delta RTNDT
Fluence (1019, E>1.0MeV)
0
0.5
1
1.5
-5
0
5
10
15
20
25
30
35
Byron 1 & 2 Weld Metal - Measured
Byron 1 & 2 Weld Metal - Best-Fit

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. 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.

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. 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 FDRT/SRPLO-009(94)	66.37[b]	57.15[b]	70[c]	60[c]	--[a]
Byron 2 Curves at 16 EFPY WCAP-14063	43.5	33.2	92.6	75.8	Current curves/PTS evaluation are NOT conservative. Using weld metal integration will be more restrictive. Byron Unit 2 curves to be regenerated and documented in WCAP-14881

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-14881, "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	82.0°F	61.8°F

The vessel material data used in the latest PTS evaluation reports[17,18] was used in this evaluation. (No new material property values were calculated.) However, 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)	ΔRT_{PTS} (°F)	RT_{PTS} (°F)
32 EFPY							
Intermediate 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	**82.0**	2.159	1.209	-30	**56**	**99.1**	**125.1**
Using surv. capsule data[(c)]	**61.8**	2.159	1.209	-30	**28**	**74.7**	**72.7**
48 EFPY							
Intermediate 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	**82.0**	3.238	1.309	-30	**56**	**107.3**	**133.3**
Using surv. capsule data[(c)]	**61.8**	3.238	1.309	-30	**28**	**80.9**	**78.9**

NOTES:

(a) 2.159×10^{19} n/cm² (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)	ΔRT_{PTS} (°F)	RT_{PTS} (°F)
32 EFPY							
Lower Shell Forging MK 24-3	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	17	24.0	21.0
Inter. Shell Forging MK 24-2	20.0	2.192	1.213	-20	24.3	24.3	28.6
Circ. Weld Metal WF447	**82.0**	2.192	1.213	10	**56**	**99.5**	**165.5**
Using surv. capsule data[(c)]	**61.8**	2.192	1.213	10	**28**	**75.0**	**113.0**
48 EFPY							
Lower Shell Forging MK 24-3	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	17	26.0	23.0
Inter. Shell Forging MK 24-2	20.0	3.288	1.312	-20	26.2	26.2	32.4
Circ. Weld Metal WF447	**82.0**	3.288	1.312	10	**56**	**107.6**	**173.6**
Using surv. capsule data[(c)]	**61.8**	3.288	1.312	10	**28**	**81.1**	**119.1**

NOTES:

(a) 2.192×10^{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:

$$2.192\times10^{19} + \frac{(2.192\times10^{19} - 3.174\times10^{18})}{32 - 4.634 \text{ EFPY}}*(48 - 32 \text{ EFPY}) = 3.288\times10^{19} \text{ n/cm}^2$$

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

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

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 off 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°
1.321×10^{19}	1.984×10^{19}	2.239×10^{19}	1.86E$\times 10^{19}$	2.162×10^{19}

Braidwood Unit 2

0°	15°	25°	35°	45°
1.299×10^{19}	1.924×10^{19}	2.199×10^{19}	1.861×10^{19}	2.174×10^{19}

10. *What vessel inlet temperatures do the plants operate?*
    Braidwood 1: 558.4°F
    Braidwood 2: 558.4°F

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 the Braidwood Units 1 and 2 weld metal (only) to show that Braidwood 1 & 2 can share weld metal data and determine an integrated weld metal chemistry factor.

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 calculated RT_{PTS} values presented in WCAP-14242 (Braidwood 1 PTS) and the information provided in the Braidwood Unit 2 material selection documents, 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 ΔRT_{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.

TABLE B-2[c]

Braidwood Units 1 & 2 Surveillance Capsule Data Calculation of Best-Fit Line as Described in Position 2.1 of Regulatory Guide 1.99, Revision 2

Material	**Capsule**	**f[a]**	**FF[b] (x)**	**Measured ΔRT_{NDT} (y)**	**FF x ΔRT_{NDT} (xy)**	**FF^2 (x^2)**
Braidwood Unit 1 Weld Metal	U	0.3814×10^{19}	0.733	10	7.333	0.538
	X	1.144×10^{19}	1.038	25	25.95	1.077
Braidwood Unit 2 Weld Metal	U	0.3933×10^{19}	0.741	0	0.00	0.550
	X	1.126×10^{19}	1.033	20	20.66	1.067
	$\Sigma^n_{i=1}$		3.545	55	53.943	3.232
	Chemistry Factor[d] = 53.943 ÷ 3.232 = **16.7**					

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 ΔRT_{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 = $\sum(FF * RT_{NDT}) \div \sum(FF^2)$

Per the 27th Edition of the CRC Standard Mathematical Tables (page 497), for a straight line fit by the method of least squares, the values b_o and b_1 are obtained by solving the normal equations

$$n\, b_o + b_1\, \Sigma x_i = \Sigma y_i$$

and

$$b_o\, \Sigma x_i + b_1\, \Sigma x_i^2 = \Sigma x_i y_i$$

These equations can be re-written as follows:

$$\sum_{i=1}^{n} y_i = an + b\sum_{i=1}^{n} x_i$$

and

$$\sum_{i=1}^{n} x_i y_i = a\sum_{i=1}^{n} x_i + b\sum_{i=1}^{n} x_i^2$$

Braidwood 1 & 2 Weld Metal:

Based on the data provided in Table B-2, the equations become:

1.) $55.0 = 4a + 3.545b$ or $a = 13.75 - 0.8863b$ and

2.) $53.943 = 3.545a + 3.232b$

Thus, by substituting Eq. 1 into Eq. 2, b = 57.69. Now, enter b (= 57.69) into Eq. 1 and a = -37.38. Therefore, the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 57.69\,(X) - 37.38$$

The error in predicting a value Y corresponding to a given X value is: $e = Y - Y'$

TABLE B-3

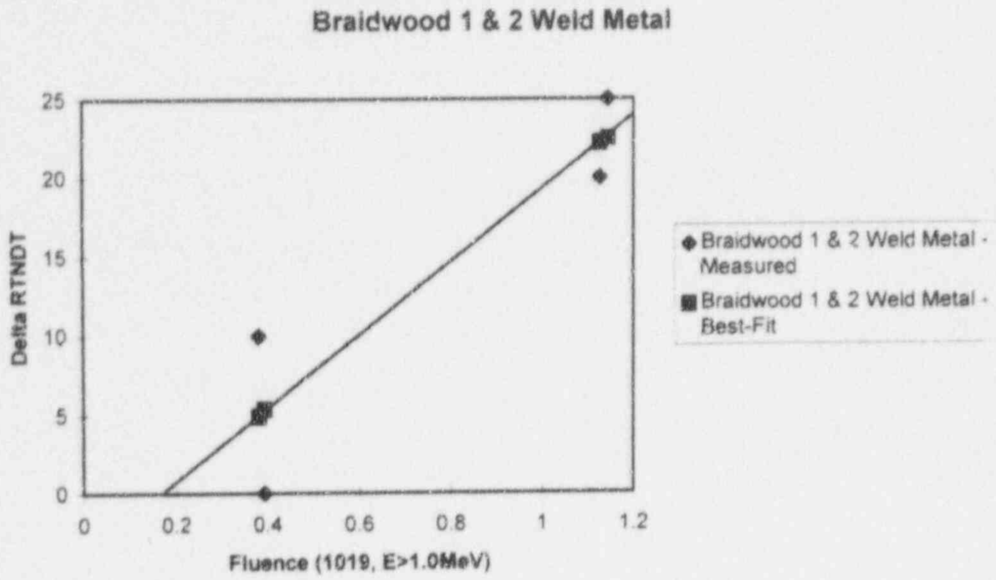
Best Fit Evaluation for Byron 1 & 2 and Braidwood 1 & 2 Weld Metal

Base Material	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Braidwood 1 & 2 Weld Metal	0.733	10	-0.00	10.00
	1.038	25	-0.00	25.00
	0.741	0	-0.00	0.00
	1.033	20	-0.00	20.00

Weld Metal:

The scatter of ΔRT_{NDT} values 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.

See the following plot of ΔRT_{NDT} versus fluence.



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

The Braidwood Unit 1 & 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. 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.

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

Braidwood Units 1 & 2 did not incorporate correlation monitor material in their surveillance program. 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	0.668
	0.029	0.686
	0.034	0.616
	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
	0.027	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.

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 WCAP-14243	76.6	65.4	69.7	60.6	Current curves/PTS evaluation are conservative. New Applicability Date: **27.9 EFPY**
Braidwood 2 Curves at 16 EFPY 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. New Applicability Date: **7.4 EFPY**

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.

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.

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 (σ_i) is 0°F when the initial RT_{NDT} is a measured value (as is the case for the Byron units). Additionally, the term σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

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

$$\mathbf{ART = I + (CF * FF) + (CF * FF)}$$

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

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

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

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

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

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

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

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

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

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 vessel material data used in the latest PTS evaluation reports[19,20] was used in this evaluation. (No new material property values were calculated.)

TABLE B-4

RT_{PTS} Values for Braidwood Unit 1

Material	CF (°F)	f	FF(a)	$RT_{NDT(U)}$ (°F)	M (°F)	ΔRT_{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	31.68	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) = $f^{(0.28 - 0.10*\log f)}$
(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)	ΔRT_{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	16.15	16.15	2.3
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	17	17.46	4.5
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.

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.

It is planned that neutron fluence evaluations for the Byron Units 1 & 2 and Braidwood Units 1 & 2, pressure vessels will be updated to incorporate the use of ENDF/B-VI cross-section libraries at the time of the next scheduled surveillance capsule withdrawal for each of the units. Based on recent experience in updating vessel fluence evaluations to the ENDF/B-VI methodology for reactors of similar design to Byron and Braidwood it is anticipated that best estimate neutron fluence values will be impacted by less than 7% compared to those over those previously reported. Likewise, application of the ENDF/B-VI methodology to the re-evaluation of neutron dosimetry from previously withdrawn surveillance capsules will change reported capsule exposures by an amount less than the uncertainties quoted with the prior dosimetry analyses.

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.

TABLE C-1

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

Irradiation Time	Fluence [n/cm²]		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	Fluence [n/cm^2]		Lead Factor	
[EFPY]	31.5 Caps	29.0 Caps	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] 31.5 Caps	29.0 Caps	Lead Factor 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 [EFPY]	Fluence [n/cm^2] 31.5 Caps	Fluence [n/cm^2] 29.0 Caps	Lead Factor 31.5 Caps	Lead Factor 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