Westinghouse Non-Proprietary Class 3

WCAP - 15569
Revision 0

Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1

Westinghouse Electric Company LLC

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

FENOC-01-039

March 2, 2001

Mr. Denny Weakland
FirstEnergy Nuclear Operating Co.
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Unit 2
Subject: Corrected Page to WCAP-15571

Dear Mr. Weakland:

Attached is the changed page for WCAP-15569, Revision 0 which provides the alternate statement of WCAP-15571" in the next to last line instead of the previous "WCAP-15571, Revision 1". The Reference list on page 8-1 of WCAP-15569, Revision 0 is correct as is.

We regret any inconvenience incurred by you on this matter. Contact Tom Laubham (412-374-6788) or me if there is anything else we can do to support you in this area.

Sincerely,

WESTINGHOUSE ELECTRIC COMPANY

for/ John DeBlasio
Edward A. Dzenis
Customer Project Manager

Attachment

cc: Bill Kline – FENOC
Mark Musulin – FENOC
Mike Testa - FENOC

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Beaver Valley Unit 1 reactor vessel using the results of the surveillance Capsule Y evaluation. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Beaver Valley Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-15571[1]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15569

Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1

C. Brown
E. Terek

November 2000

Prepared by the Westinghouse Electric Company LLC
for the First Energy Company

Approved: ______________________
D. M. Trombola, Manager
Mechanical Systems Integration

Approved: ______________________
C. . Boyd, Manager
Equipment & Materials Technology

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Beaver Valley Unit 1 reactor vessel beltline materials based upon the results of the Surveillance Capsule Y evaluation. The conclusion of this report is that all the beltline materials in the Beaver Valley Unit 1 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, and 300°F for circumferential welds at EOL (28 EFPY). For Life Extension (45 EFPY) RT_{PTS}, all but the lower shell plate, B6903-1, met this criteria. Specifically, the lower shell plate, B6903-1, was the most limiting material with 28 and 45 EFPY PTS values of 259°F and 274°F, respectively. It is expected to reach the screening criteria at approximately 38.5 EFPY.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Beaver Valley Unit 1 reactor vessel using the results of the surveillance Capsule Y evaluation. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Beaver Valley Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-15571, Revision 1[1]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule, 10 CFR Part 50.61[2], was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is also described in Regulatory Guide 1.99, Revision 2[3].

2. The rule is restructured to improve clarity, with the Requirements section giving only the requirements for the value for the reference temperature for end of license (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and
  300°F for circumferential weld materials.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence at the clad/base metal interface for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \tag{1}$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \tag{3}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF, when using credible surveillance data, is determined using Equation 5.

The EOL Fluence (f) is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}.

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF for surveillance materials is determined from Equation 5.

$$CF = \frac{\sum[A_i * fi^{(0.28-0.10\log fi)}]}{\sum[fi^{(0.56-0.20\log fi)}]} \tag{5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}. For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{PTS,\ measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{PTS} = \Delta RT_{PTS,\ measured} + 1.0*(T_{capsule} - T_{plant})$$

Note that the temperature adjust methodology has been reinforce by the NRC at the NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Beaver Valley Unit 1 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as, "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the Beaver Valley Unit 1 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from WCAP-15570, Table 4-10[4]. The best estimate copper and nickel content is also documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for Beaver Valley Unit 1 reactor vessel beltline material properties are also shown in Table 1.

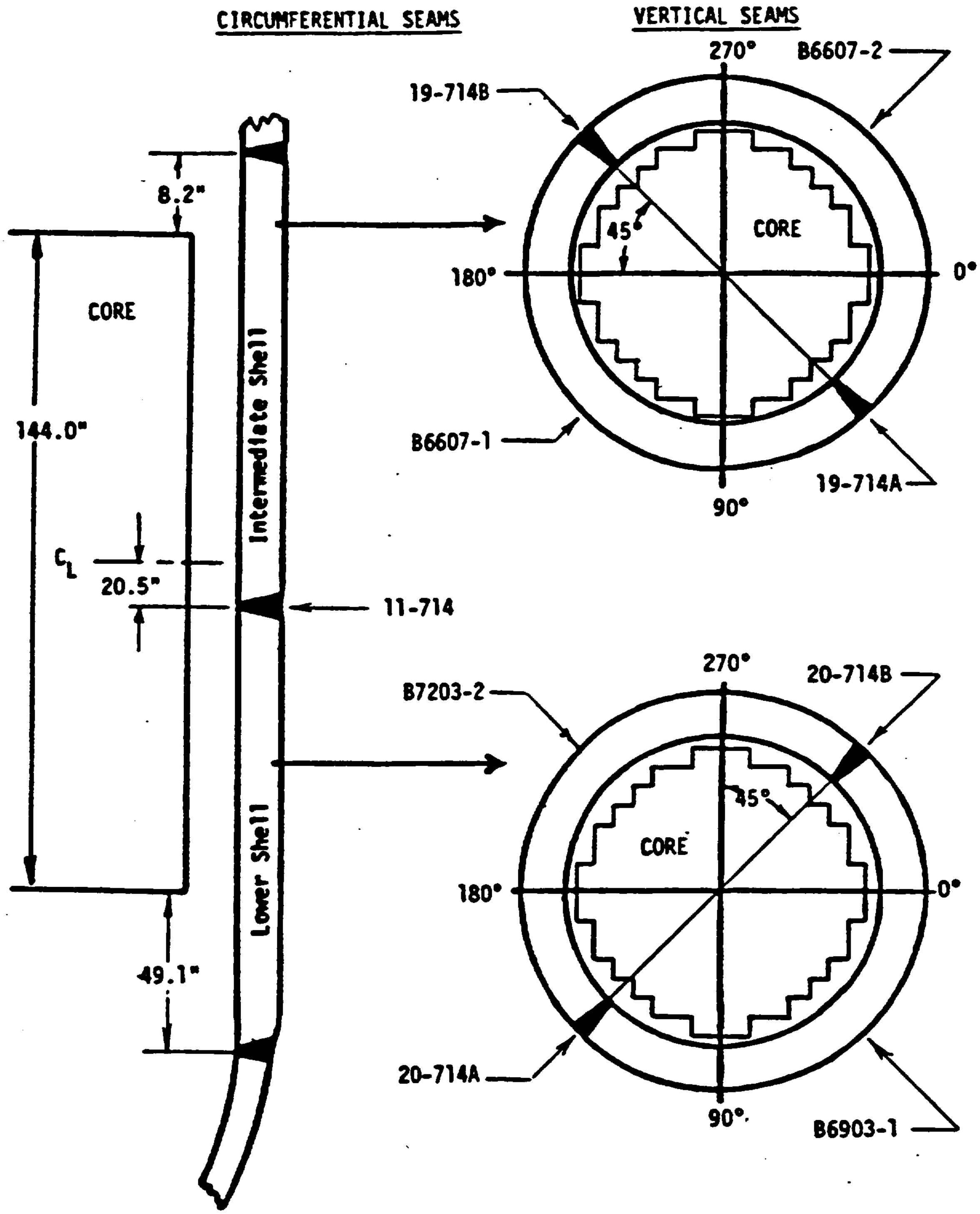

Figure 1: Identification and Location of Beltline Region Materials for the Beaver Valley Unit 1 Reactor Vessel

Table 1
Beaver Valley Unit 1 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}*
Intermediate Shell Plate B6607-1	0.14	0.62	43
Intermediate Shell Plate B6607-2	0.14	0.62	73
Lower Shell Plate (surveillance program) B6903-1	0.21	0.54	27
Lower Shell Plate B7203-2	0.14	0.57	20
Intermediate to Lower Shell Weld 11-714	0.27	0.07	-56
St. Lucie Surveillance Weld	0.23	0.07	---
Intermediate Longitudinal Weld 19-714 A&B	0.28	0.63	-56
Beaver Valley Surveillance Weld	0.26	0.61	---
Lower Longitudinal Weld 20-714 A&B	0.34	0.61	-56
Fort Calhoun Surveillance Weld	0.35	0.60	---

* This value is determined from all available data.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the clad base metal interface of the Beaver Valley Unit 1 reactor vessel for 28 and 45 EFPY are shown in Table 2. These values were projected using the results of the Capsule Y analysis. See Section 6.0 of the Capsule Y analysis report, WCAP-15571[1].

TABLE 2
Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for Beaver Valley Unit 1 at 28 (EOL) and 45 (Life Extension) EFPY

Material	Location	28 EFPY Fluence	45 EFPY Fluence
Intermediate Shell Plate B6607-1	0°	3.54×10^{19} n/cm^2	5.85×10^{19} n/cm^2
Intermediate Shell Plate B6607-2	0°	3.54×10^{19} n/cm^2	5.85×10^{19} n/cm^2
Lower Shell Plate (surveillance program) B6903-1	0°	3.54×10^{19} n/cm^2	5.85×10^{19} n/cm^2
Lower Shell Plate B7203-2	0°	3.54×10^{19} n/cm^2	5.85×10^{19} n/cm^2
Intermediate to Lower Shell Weld 11-714	0°	3.53×10^{19} n/cm^2	5.82×10^{19} n/cm^2
Intermediate Longitudinal Weld 19-714 A&B	45°	7.08×10^{18} n/cm^2	1.13×10^{19} n/cm^2
Lower Longitudinal Weld 20-714 A&B	45°	7.08×10^{18} n/cm^2	1.13×10^{19} n/cm^2

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

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Beaver Valley Unit 1 reactor vessel for fluence values at the EOL (28 EFPY) and life extension (45 EFPY).

Per 10 CFR Part 50.61, Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. The Beaver Valley Unit 1 surveillance program data has evaluated and shown to not be credible in WCAP-15571[1]. The related surveillance program results have been included in this PTS evaluation.

As presented in Table 3, chemistry factor values for Beaver Valley Unit 1 based on average copper and nickel weight percent values were calculated using Tables 1 and 2 from 10 CFR 50.61[2]. Additionally, chemistry factor values based on credible surveillance capsule data are calculated in Table 4 for Beaver Valley, and Table 5 for St. Lucie and Fort Calhoun. Tables 6 and 7 contain the RT_{PTS} calculations for all beltline region materials at EOL (28 EFPY) and life extension (45 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material	Cu wt. %	Ni wt. %	Chemistry Factor, °F
Intermediate Shell Plate B6607-1	0.14	0.62	100.5
Intermediate Shell Plate B6607-2	0.14	0.62	100.5
Lower Shell Plate (surveillance program) B6903-1	0.21	0.54	147.2
Lower Shell Plate B7203-2	0.14	0.57	98.7
Intermediate to Lower Shell Weld 11-714	0.27	0.07	124.3
Intermediate Longitudinal Weld 19-714 A&B	0.28	0.63	191.7
Lower Longitudinal Weld 20-714 A&B	0.34	0.61	210.5
Beaver Valley Unit 1 Surveillance Weld (Ht. # 305424)	0.26	0.61	181.6
St. Lucie Unit 1 Surveillance Weld (Ht. # 90136)	0.23	0.07	106.6
Ft. Calhoun Unit 1 Surveillance Weld (Ht. # 305414)	0.35	0.60	212.0

TABLE 4
Calculation of Chemistry Factors using Surveillance Capsule Data Per Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Plate B6903-1[d] (Longitudinal)	V	.323	.689	128.49	88.53	.475
	U	.646	.878	118.93	104.42	.771
	W	.986	.996	148.52	147.93	.992
	Y	2.15	1.21	142.18	172.04	1.464
Lower Shell Plate B6903-1[d] (Transverse)	V	.323	.689	137.81	94.95	.475
	U	.646	.878	131.84	115.76	.771
	W	.986	.996	179.99	179.27	.992
	Y	2.15	1.21	166.93	201.99	1.464
				SUM:	1104.89	7.404
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (1104.89) \div (7.404) = 149.2$ °F						
Beaver Valley Surveillance Weld Metal 305424[d]	V	.323	.689	169.3	116.7	.475
	U	.646	.878	176.3	154.8	.771
	W	.986	.996	199.0	198.2	.992
	Y	2.15	1.21	189.4	229.2	1.464
				SUM:	689.8	3.702
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (689.8) \div (3.702) = 188.8$°F						

Notes:
(a) f = Calculated fluence from the Beaver Valley Unit 1 capsule Y dosimetry analysis results, (x 10^{19} n/cm^2, E > 1.0 MeV).
(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.
(c) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 1.06.
(d) Data not credible.

TABLE 5

Circ. Weld CF Based on St. Lucie Unit 1 and Fort Calhoun Unit 1 S/P Data

Material	Capsule	Capsule $f^{(a)}$	$T_{capsule}$	$FF^{(b)}$	$\Delta RT_{NDT}{}^{(c)}$	Temp. Adjusted $\Delta RT_{NDT}{}^{(d)}$	Ratio * Temp. Adjusted $\Delta RT_{NDT}{}^{(d)}$	$FF*\Delta RT_{NDT}$	FF^2
St. Lucie Surveillance Weld Metal Ht. 90136	97°	.627	546.7	.869	72.3	74.9	87.6	76.1	.755
	104°	.909	546.7	.973	67.4	70.0	81.9	79.7	.947
	284°	1.41	546.7	1.10	68.0	70.6	82.6	90.9	1.21
							SUM:	246.7	2.91
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (246.7) \div (2.91) = 84.8°F$									
Fort Calhoun Surveillance Weld Metal Ht. 305414	W-225	.553	527	.834	238	220.9	219.4	183.0	.696
	W-265	.771	534	.927	221	210.9	209.4	194.1	.859
	W-275	1.28	538	1.07	219	212.9	211.4	226.2	1.14
							SUM:	603.3	2.695
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (603.3) \div (2.695) = 223.9°F$									

Notes:

(a) f = Calculated fluence, (x 10^{19} n/cm^2, E > 1.0 MeV), Reference 5 and 6.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb. shift values taken from Reference 5 and 6.

(d) Temperature adjusted 30 ft-lb. Values = $\Delta RT_{NDT,measured} + 1.0(T_{capsule} - T_{plant})$

(e) Ratio adjustment for St. Lucie ($CF_{VW}/CF_{SW} * \Delta RT_{NDT}$) = 124.3/106.6 = 1.17

(f) Ratio adjustment for Ft. Calhoun ($CF_{VW}/CF_{SW} * \Delta RT_{NDT}$) = 210.5/212.0 = 0.993

TABLE 6
RT_{PTS} Calculation for Beaver Valley Unit 1 Beltline Region Materials at EOL (28 EFPY)

Material	Fluence (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Plate B6607-1	3.54	1.329	100.5	133.6	34	43	211
Intermediate Shell Plate B6607-2	3.54	1.329	100.5	133.6	34	73	241
Lower Shell Plate B7203-2	3.54	1.329	98.7	131.2	34	20	185
Lower Shell Plate B6903-1	3.54	1.329	147.2	195.6	34	27	257
→ Using S/C Data[e]	3.54	1.329	149.2	198.3	34	27	259
Inter. Shell Circ. Weld 19-714A/B	0.708	0.903	191.7	173.1	65.5	-56	183
→ Using S/C Data[e]	0.708	0.903	188.8	170.5	65.5	-56	180
Lower Shell Long. Weld 20-714A/B	0.708	0.903	210.5	190.1	65.5	-56	200
→ Using S/C Data[f]	0.708	0.903	223.9	202.2	65.5	-56	212
Circumferential Weld 11-714	3.53	1.329	124.3	165.2	65.5	-56	175
→ Using S/C Data[d]	3.53	1.329	84.8	112.7	44	-56	101

Notes:

(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(c) $\Delta RT_{PTS} = CF * FF$
(d) Based on credible St. Lucie Unit 1 surveillance data.
(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_Δ
(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_Δ

TABLE 7

RT_{PTS} Calculation for Beaver Valley Unit 1 Beltline Region Materials at Life Extension (45 EFPY)

Material	Fluence (n/cm2, E>1.0 MeV)	FF	CF (°F)	ΔRTPTS(c) (°F)	Margin (°F)	$RT_{NDT(U)}$(a) (°F)	RT_{PTS}(b) (°F)
Intermediate Shell Plate B6607-1	5.85	1.43	100.5	143.7	34	43	221
Intermediate Shell Plate B6607-2	5.85	1.43	100.5	143.7	34	73	251
Lower Shell Plate B7203-2	5.85	1.43	98.7	141.1	34	20	195
Lower Shell Plate B6903-1	5.85	1.43	147.2	210.5	34	27	**272**
→ Using S/C Data[e]	5.85	1.43	149.2	213.4	34	27	**274**[g]
Inter. Shell Circ. Weld 19-714A/B	1.13	1.03	191.7	197.5	65.5	-56	207
→ Using S/C Data[e]	1.13	1.03	188.8	194.5	65.5	-56	204
Lower Shell Long. Weld 20-714A/B	1.13	1.03	210.5	216.8	65.5	-56	226
→ Using S/C Data[f]	1.13	1.03	223.9	230.6	65.5	-56	240
Circumferential Weld 11-714	5.82	1.43	124.3	177.7	65.5	-56	187
→ Using S/C Data[d]	5.82	1.43	84.8	121.3	44	-56	109

Notes:

(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(c) ΔRT_{PTS} = CF * FF
(d) Based on credible St. Lucie Unit 1 surveillance data.
(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_Δ.
(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_Δ.
(g) Material is expected to have reached the screening criteria of 270° by 38.5 EFPY.

7 CONCLUSIONS

As shown in Tables 6 and 7, all of the beltline region materials in the Beaver Valley Unit 1 reactor vessel have EOL (28 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates and longitudinal welds, and 300°F for circumferential welds. For Life Extension (45 EFPY) RT_{PTS}, all but the lower shell plate, B6903-1, and the corresponding surveillance material, meet this criteria. Specifically, the lower shell plate, B6903-1 was the most limiting material with 28 and 45 EFPY PTS values of 259°F and 274°F, respectively.

8 REFERENCES

1 WCAP-15571, Revision 0, Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", C. Brown, et al., August, 2000.

2 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

4 WCAP-15570, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", C. Brown, September, 2000.

5 WCAP-15446, "Analysis of Capsule 284° from the Florida Power and Light Company St. Lucie Unit 1 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, September 2000.

6 BAW-2226-00 (BWNT Document No. 77-2226-00), "Analysis of Capsule W-275 Omaha Public Power District Fort Calhoun Station Unit No. 1", M. J. DeVan, July 1994.