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Revision 1

September 2011

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RECORD OF REVISIONS

Revision 0: Original Issue

Revision 1: The purpose of this revision is to update the reactor vessel integrity evaluations contained in this document due to updated neutron fluence projections. Furthermore, due to updated surveillance capsule fluence values, the surveillance capsule credibility evaluation has also been updated since the Capsule X analysis and is included in Appendix A of this report. Note that these reactor vessel integrity calculations, along with the credibility evaluation contained within this document, supersede the previous respective evaluations. Change bars were not used in this document to record the changes between Revisions 0 and 1 since this revision should be considered an entirely new document based on the nature of the updates.

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EXECUTIVE SUMMARY

The purpose of this supplement is to determine the Reference Temperature for Pressurized Thermal Shock (RT_{PTS}) values and Upper-Shelf Energy (USE) values for the Beaver Valley Unit 2 reactor vessel beltline and extended beltline materials. This analysis will be based upon the results of the latest surveillance capsule X evaluation and the implementation of the Extended Power Uprate program. These calculations are performed for End-of-License Extension (EOLE) at 54 Effective Full Power Years (EFPY). Furthermore, as part of this analysis, the current pressure-temperature limits along with the surveillance capsule withdrawal schedule and credibility evaluation will be assessed based on the updated fluence results.

The limiting plate material in the Beaver Valley Unit 2 reactor vessel is the upper shell plate B9003-2 with a projected EOLE RT_{PTS} value of 161.6°F for 54 EFPY. This value is below the screening criterion of 270°F for forgings/plates in 10 CFR 50 Part 61. The limiting weld material in the Beaver Valley Unit 2 reactor vessel is the upper shell longitudinal weld (heat BOHB (E-8018)) with an EOLE RT_{PTS} value of 130.4°F. This RT_{PTS} value is well below the screening criteria value of 270°F for axial welds at EOLE (54 EFPY). All of the beltline and extended beltline materials maintain USE above 50 ft-lbs at EOLE.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) that causes 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 the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The predicted decrease in Upper Shelf Energy (USE) is determined as a function of fluence and copper content using either 1) Figure 2 of Regulatory Guide 1.99, Revision 2, Position 1.2, or 2) Surveillance program test results and Figure 2 of Regulatory Guide 1.99, Revision 2, Position 2.2 [Reference 1]. Both methods require the use of the 1/4T (1/4 vessel thickness) vessel fluence.

The purpose of this report is to determine the Reference Temperature for Pressurized Thermal Shock (RT_{PTS}) and USE values for the Beaver Valley Power Station Unit 2 (BVPS-2) reactor vessel using the results of the surveillance Capsule X evaluation and the implementation of the Extended Power Uprate (EPU) Program. The results presented in this report are for End-of-License Extension (EOLE) at 54 Effective Full Power Years (EFPY). Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS} and USE. Section 4.0 provides the reactor vessel beltline and extended beltline region material properties for the BVPS-2 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0. The results of the RT_{PTS} and USE calculations are presented in Section 6.0. The current pressure-temperature limit curve applicability is presented in Section 7.0. The recommended surveillance capsule withdrawal schedule is presented in Section 8.0. The conclusion and references for the PTS and USE evaluations follow in Sections 9.0 and 10.0, respectively.

The surveillance capsule credibility analysis, based on the results of the surveillance Capsule X analysis and the updated capsule fluence values, is presented in Appendix A of this report.

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 revised PTS Rule, 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996 [Reference 2].

This amendment to the PTS Rule makes the following changes:

- The rule incorporates 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 currently described in Regulatory Guide 1.99, Revision 2 [Reference 1].

- The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value of the reference temperature for EOL fluence, RT_{PTS}.

- 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 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, 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

  - 300°F for circumferential weld materials

All available surveillance data must be considered in the evaluation. All credible plant-specific surveillance data must also be used in the evaluation.

3 METHODOLOGY FOR CALCULATION OF RT_{PTS} AND USE

3.1 RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL or EOLE fluence 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

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

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 should not exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{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 Table 1 for welds and Table 2 for base metal (plates or forgings) of the PTS Rule. Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of

CF is determined in Equation 5.

"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 or EOLE fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL or EOLE fluence values for determining ΔRT_{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 is determined from Equation 5.

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

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. This is done by multiplying them by the ratio of the chemistry factor for the vessel material to that of the surveillance weld.

3.2 USE

Per Regulatory Guide 1.99, Revision 2, the Charpy V-notch USE is assumed to decrease as a function of fluence and copper content when surveillance data is not used, as indicated in Figure 2 of the guide. Linear interpolation is permitted. In addition, if surveillance data is to be used, the decrease in USE may be obtained by plotting the reduced plant surveillance data on Figure 2 of the guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line should be used in preference to the existing graph. The USE can be predicted using the corresponding 1/4T fluence projection, the copper content of the beltline materials and/or the results of the capsules tested to date using Figure 2 in Regulatory Guide 1.99, Revision 2.

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 BVPS-2 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 and 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." In addition to the beltline regions, materials that exceed 1 x 10^{17} n/cm^2 (E>1.0 MeV) are subject to the guidelines provided in Appendix H of 10 CFR 50 [Reference 3]. In accordance with 10 CFR 50, Appendix H, any materials exceeding 1x10^{17} n/cm^2 (E>1.0 MeV) must be monitored to evaluate the changes in fracture toughness. Reactor vessel materials not traditionally regarded as plant limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at 54 EFPY.

Material property values were obtained from material test certifications from the original fabrication as well as the additional material chemistry tests performed as part of the BVPS-2 surveillance capsule testing program [Reference 4]. The average copper and nickel values were calculated for each beltline and extended beltline region material using all of the available material chemistry information. A summary of the pertinent chemical and mechanical properties of the beltline and extended beltline region forgings/plates and weld material of the BVPS-2 reactor vessel is provided in Tables 4-1 and 4-2.

Table 4-1 BVPS-2 Reactor Vessel Beltline Material Properties[(a)]

Material Description	Material ID	Heat Number	Wt % Cu	Wt % Ni	Initial RT_{NDT}[(b)] (°F)	Initial USE (ft-lbs)
Intermediate Shell Plate	B9004-1	---	0.065	0.55	60	83
Intermediate Shell Plate	B9004-2	---	0.06	0.57	40	79
Lower Shell Plate	B9005-1	---	0.08	0.58	28	82
Lower Shell Plate	B9005-2	---	0.07	0.57	33	78
Intermediate to Lower Shell Girth Weld	101-171	83642	0.046	0.086	-30	145
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	0.046	0.086	-30	145
Lower Shell Longitudinal Welds	101-142 A&B	83642	0.046	0.086	-30	145
Surveillance Weld	---	83642	0.065	0.065	---	---

Notes:

a) Materials information taken from WCAP-16527-NP [Reference 4] and WCAP-16528-NP [Reference 5].

b) All initial RT_{NDT} values are measured values.

Table 4-2 BVPS-2 Reactor Vessel Extended Beltline Material Properties[(a)]

Material Description	Material ID	Heat Number (Lot Number)	Wt % Cu	Wt % Ni	Initial RT_{NDT}[(b)] (°F)	Initial USE (ft-lbs)
Upper Shell	B9003-1	A9406-1	0.13	0.60	50	98
	B9003-2	B4431-2	0.12	0.60	60	80
	B9003-3	A9406-2	0.13	0.60	50	98
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.156[(d)]	0.059[(d)]	-50	96
		51912 (3536)	0.156[(d)]	0.059[(d)]	-70	114
		EAIB	0.02	0.98	10 (Gen)[(c)]	118
		IAGA	0.03	0.98	-30	160
		BOHB	0.05	1.00	10 (Gen)[(c)]	97
		BAOED	0.02	1.00	-50	150
Upper Shell to Intermediate Shell Girth Weld	103-121	4P5174 (1122)	0.09	1.00[(g)]	-50	70
		51922 (3489)	0.05	1.00[(g)]	-56 (Gen)[(c)]	102
		AAGC	0.03	0.98	-70	111
		KOIB	0.03	0.97	-60	110
Inlet Nozzles	B9011-1	2V2436-01-002	0.11	0.85	60	105 (68.25)[(e)]
	B9011-2	2V2437-02-001	0.13[(f)]	0.88	60 (Gen)[(c)]	116 (75.4)[(e)]
	B9011-3	2V2445-02-003	0.13[(f)]	0.84	70	122 (79.3)[(e)]
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174 (1122)	0.09	1.00[(g)]	-50	70
		LOHB	0.03	1.03	-60	137
		HABJC	0.02	1.02	-70	162
		BABBD	0.02	1.04	-70	142
		FABGC	0.03	1.02	-80	119
		EOBC	0.02	0.96	-60	127
		FAAFC	0.07	1.04	-60	119
		CCJC	0.02	0.99	-60	109
		FAGB	0.02	1.06	-30	114
		BAOED	0.02	1.00	-50	150
Outlet Nozzles	B9012-1	AV8080-2E9558	0.13[(f)]	0.72	-10	146
	B9012-2	AV8120-2E9560	0.13[(f)]	0.74	-10	128
	B9012-3	AV8097-2E9559	0.13[(f)]	0.70	-10	127
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	0.02	1.04	-70	142
		FAAFC	0.07	1.04	-60	119
		HAAEC	0.03	1.03	-80	178
		HABJC	0.02	1.02	-70	162
		HAGB	0.02	1.04	-40	194
		GACJC	0.03	1.00	-80	113
		JAHB	0.03	0.97	-40	149

Notes:

a) All of the materials data is obtained from Combustion Engineering report MISC-PENG-ER-021 [Reference 6] except as noted.
b) Initial RT_{NDT} values are measured values, unless otherwise noted.
c) The generic initial RT_{NDT} values were determined in accordance with NUREG-0800 [Reference 8] and 10 CFR 50.61 [Reference 2].
d) Chemistry obtained from CE Report NPSD-1039, Revision 2 [Reference 7].
e) Value in parenthesis is the 65% value per NUREG-0800 [Reference 8].
f) The Cu wt% was not available from the CMTR so in accordance with Regulatory Guide 1.99, Rev. 2 [Reference 1], a standard deviation analysis (average + standard deviation) was done to determine the value based on Westinghouse 508 Class 2 Nozzle Forgings (178 data points).
g) Default Wt % Ni content per Regulatory Guide 1.99, Revision 2.

5 NEUTRON FLUENCE VALUES

The maximum neutron exposures at the pressure vessel clad/base metal interface at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core major axes are presented in Table 5-1. The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the BVPS-2 reactor vessel are shown in Table 5-2 for the beltline and extended beltline materials. The calculated fast neutron fluence (E > 1.0 MeV) values at the radial and azimuthal center of the surveillance capsule positions, 17° and 20°, are presented in Table 5-3. The fluence projections were determined using ENDF/B-VI cross sections and are based on the results of the Capsule X radiation analysis and comply with Reg. Guide 1.190 [Reference 9].

These fluence data tabulations include fuel-cycle-specific calculated neutron exposures at the end of Cycle 14 as well as future projections to the end of Cycle 15 and for several intervals extending to 60 EFPY.

Neutron exposure projections beyond the end of Cycle 15 were based on the spatial power distributions and associated plant characteristics of Cycles 14 and 15 in conjunction with an uprated core power level of 2900 MWt.

Table 5-1 Maximum Calculated Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for BVPS-2

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]			
		0°	15°	30°	45°
1	1.25	1.94×10^{18}	1.08×10^{18}	7.99×10^{17}	5.46×10^{17}
2	2.26	3.00×10^{18}	1.75×10^{18}	1.30×10^{18}	8.87×10^{17}
3	3.50	4.33×10^{18}	2.64×10^{18}	2.00×10^{18}	1.39×10^{18}
4	4.76	5.80×10^{18}	3.56×10^{18}	2.67×10^{18}	1.82×10^{18}
5	5.99	7.25×10^{18}	4.44×10^{18}	3.28×10^{18}	2.23×10^{18}
6	7.24	8.50×10^{18}	5.27×10^{18}	3.99×10^{18}	2.74×10^{18}
7	8.51	9.83×10^{18}	6.11×10^{18}	4.68×10^{18}	3.26×10^{18}
8	9.84	1.11×10^{19}	6.94×10^{18}	5.34×10^{18}	3.74×10^{18}
9	11.06	1.24×10^{19}	7.67×10^{18}	5.89×10^{18}	4.17×10^{18}
10	12.56	1.38×10^{19}	8.56×10^{18}	6.59×10^{18}	4.67×10^{18}
11	14.00	1.52×10^{19}	9.45×10^{18}	7.24×10^{18}	5.11×10^{18}
12	15.41	1.66×10^{19}	1.03×10^{19}	7.84×10^{18}	5.55×10^{18}
13	16.82	1.82×10^{19}	1.11×10^{19}	8.46×10^{18}	5.98×10^{18}
14	18.18	1.95×10^{19}	1.19×10^{19}	9.06×10^{18}	6.40×10^{18}
15	19.44	2.06×10^{19}	1.26×10^{19}	9.60×10^{18}	6.80×10^{18}
Future	32.00	3.21×10^{19}	1.97×10^{19}	1.51×10^{19}	1.08×10^{19}
Future	48.00	4.67×10^{19}	2.88×10^{19}	2.21×10^{19}	1.58×10^{19}
Future	54.00	5.21×10^{19}	3.22×10^{19}	2.47×10^{19}	1.77×10^{19}
Future	60.00	5.76×10^{19}	3.56×10^{19}	2.73×10^{19}	1.96×10^{19}

Table 5-2 Calculated Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for BVPS-2 for the Beltline and Extended Beltline Regions

Material	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
	32 EFPY	**48 EFPY**	**54 EFPY**	**60 EFPY**
Beltline Materials				
Intermediate Shell Plates	3.19×10^{19}	4.64×10^{19}	5.18×10^{19}	5.73×10^{19}
Lower Shell Plates	3.21×10^{19}	4.67×10^{19}	5.21×10^{19}	5.76×10^{19}
Intermediate to Lower Shell Girth Weld	3.18×10^{19}	4.63×10^{19}	5.18×10^{19}	5.72×10^{19}
Intermediate Shell Longitudinal Welds	1.07×10^{19}	1.57×10^{19}	1.76×10^{19}	1.95×10^{19}
Lower Shell Longitudinal Welds	1.08×10^{19}	1.58×10^{19}	1.77×10^{19}	1.96×10^{19}
Extended Beltline Materials				
Upper Shell Plates	2.97×10^{18}	4.56×10^{18}	5.15×10^{18}	5.75×10^{18}
Upper to Intermediate Shell Girth Weld	2.97×10^{18}	4.56×10^{18}	5.15×10^{18}	5.75×10^{18}
Inlet Nozzle to Upper Shell Weld – Lowest Extent	1.70×10^{17}	2.63×10^{17}	2.98×10^{17}	3.33×10^{17}
Outlet Nozzle to Upper Shell Weld – Lowest Extent	8.60×10^{16}	1.34×10^{17}	1.51×10^{17}	1.69×10^{17}
Lower Shell to Lower Closure Head Weld[(a)]	4.11×10^{15}	6.19×10^{15}	6.97×10^{15}	7.74×10^{15}

Notes:

a) Extended beltline materials are currently interpreted to be the reactor vessel materials that will be exposed to a neutron fluence greater than or equal to 1×10^{17} n/cm^2 (E > 1.0 MeV) at EOLE. Since the fluence for the lower shell to closure head weld material is less than 1×10^{17} n/cm^2, this material has been omitted from the calculations contained in this report.

Table 5-3 Calculated Fluence (E > 1.0 MeV) at the Surveillance Capsule Locations for BVPS-2

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]	
		17°	20°
1	1.25	6.15×10^{18}	5.25×10^{18}
2	2.26	1.02×10^{19}	8.74×10^{18}
3	3.50	1.57×10^{19}	1.37×10^{19}
4	4.76	2.12×10^{19}	1.85×10^{19}
5	5.99	2.64×10^{19}	2.29×10^{19}
6	7.24	3.14×10^{19}	2.74×10^{19}
7	8.51	3.63×10^{19}	3.17×10^{19}
8	9.84	4.13×10^{19}	3.61×10^{19}
9	11.06	4.55×10^{19}	3.98×10^{19}
10	12.56	5.09×10^{19}	4.45×10^{19}
11	14.00	5.63×10^{19}	4.91×10^{19}
12	15.41	6.11×10^{19}	5.32×10^{19}
13	16.82	6.63×10^{19}	5.76×10^{19}
14	18.18	7.10×10^{19}	6.16×10^{19}
15	19.44	7.53×10^{19}	6.53×10^{19}
Future	32.00	1.18×10^{20}	1.03×10^{20}
Future	48.00	1.73×10^{20}	1.50×10^{20}
Future	54.00	1.94×10^{20}	1.68×10^{20}
Future	60.00	2.14×10^{20}	1.85×10^{20}

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

6.1 BVPS-2 RT_{PTS} CALCULATIONS FOR 54 EFPY

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline and extended beltline region materials of the BVPS-2 reactor vessel for fluence values at EOLE (54 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For BVPS-2, the related surveillance program results have been included in this PTS evaluation. Specifically, the BVPS-2 plant-specific surveillance capsule data for the intermediate shell (IS) plate B9004-2 and weld metal (heat 83642) is provided and applied as follows:

1) There have been four capsules removed from the BVPS-2 reactor vessel.

2) The data for the surveillance program plate material is deemed non-credible. The data was used with a σ_Δ margin of 17°F.

3) The data for the Unit 2 surveillance program weld material is deemed credible. The data was used with a σ_Δ margin of 14°F.

4) The surveillance capsule materials are representative of the actual vessel plate and all of the beltline welds (weld heat 83642).

5) The resulting RT_{PTS} values for intermediate shell plate B9004-2 remains below the screening criteria at 54 EFPY based on Positions 1.1 and 2.1 of Regulatory Guide 1.99, Revision 2. The resulting RT_{PTS} values for all other materials remains below the PTS Rule screening criteria at 54 EFPY.

Chemistry factor values for the BVPS-2 beltline region materials based on Position 1.1 and 2.1 from Regulatory Guide 1.99, Revision 2, are presented in Table 6-1. Additionally, chemistry factor values for the BVPS-2 extended beltline materials based on Position 1.1 and 2.1 from Regulatory Guide 1.99, Revision 2, are presented in Table 6-2. Tables 6-3 and 6-4 contain the RT_{PTS} calculations for all beltline and extended beltline region materials at 54 EFPY.

6.2 BVPS-2 UPPER-SHELF ENERGY CALCULATIONS FOR 54 EFPY

Surveillance data exists for plate B9004-2 and weld heat 83642 for BVPS-2. Each of the measured drops in USE for each of these material heats is plotted on Figure 2 of Regulatory Guide 1.99, Revision 2 with a horizontal line drawn parallel to the existing lines as the upper bound of all data. Figure 6-1 was used in the determination of the percent decrease in USE for the beltline and extended beltline materials. Tables 6-5 and 6-6 document the USE values for all of the materials at 54 EFPY. All of the beltline or extended beltline material USE values maintain 50 ft-lbs or greater at 54 EFPY.

Table 6-1 BVPS-2 Beltline Material Chemistry Factor Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material Description	Material ID	Heat Number	Chemistry Factor (°F)	
			Position 1.1	Position 2.1
Intermediate Shell Plate	B9004-1	---	40.5	---
Intermediate Shell Plate	B9004-2	---	37	51.4
Lower Shell Plate	B9005-1	---	51	---
Lower Shell Plate	B9005-2	---	44	---
Intermediate to Lower Shell Girth Weld	101-142 A&B	83642	34.4	12.5
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	34.4	12.5
Lower Shell Longitudinal Welds	101-171	83642	34.4	12.5
Surveillance Weld	---	83642	38	---

Table 6-2 BVPS-2 Extended Beltline Material Chemistry Factor Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1

Material Description	Material ID	Heat Number (Lot Number)	Position 1.1 Chemistry Factor (°F)
Upper Shell Plates	B9003-1	A9406-1	91.0
	B9003-2	B4431-2	83.0
	B9003-3	A9406-2	91.0
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	73.71
		51912 (3536)	73.71
		EAIB	27.0
		IAGA	41.0
		BOHB	68.0
		BAOED	27.0
Upper to Intermediate Shell Girth Weld	103-121	4P5174 (1122)	122.0
		51922 (3489)	68.0
		AAGC	41.0
		KOIB	41.0
Inlet Nozzles	B9011-1	2V2436-01-002	77.0
	B9011-2	2V2437-02-001	96.0
	B9011-3	2V2445-02-003	96.0
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174 (1122)	122.0
		LOHB	41.0
		HABJC	27.0
		BABBD	27.0
		FABGC	41.0
		EOBC	27.0
		FAAFC	95.0
		CCJC	27.0
		FAGB	27.0
		BAOED	27.0
Outlet Nozzles	B9012-1	AV8080-2E9558	94.0
	B9012-2	AV8120-2E9560	94.5
	B9012-3	AV8097-2E9559	93.5
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	27.0
		FAAFC	95.0
		HAAEC	41.0
		HABJC	27.0
		HAGB	27.0
		GACJC	41.0
		JAHB	41.0

Table 6-3 RT_{PTS} Values for BVPS-2 Beltline Region Materials at 54 EFPY

Material Description	Material ID	Heat Number	Surface Neutron Fluence ($x10^{19}$ n/cm^2)	Fluence Factor, FF[(a)]	Chemistry Factor (°F)	Initial RT_{NDT}[(b)] (°F)	ΔRT_{PTS}[(c)] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[(d)] (°F)	RT_{PTS}[(e)] (°F)
Intermediate Shell Plate	B9004-1	---	5.18	1.4092	40.5	60	57.1	0	17	34	151.1
Intermediate Shell Plate	B9004-2	---	5.18	1.4092	37	40	52.1	0	17	34	126.1
→ Using non-credible surveillance data[(f)]			5.18	1.4092	51.4	40	72.4	0	17	34	146.4
Lower Shell Plate	B9005-1	---	5.21	1.4104	51	28	71.9	0	17	34	133.9
Lower Shell Plate	B9005-2	---	5.21	1.4104	44	33	62.1	0	17	34	129.1
Intermediate to Lower Shell Girth Weld	101-171	83642	5.18	1.4092	34.4	-30	48.5	0	24.2	48.5	67.0
→ Using credible surveillance data[(f)]			5.18	1.4092	12.5	-30	17.6	0	8.8	17.6	5.2
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	1.76	1.1554	34.4	-30	39.7	0	19.9	39.7	49.5
→ Using credible surveillance data[(f)]			1.76	1.1554	12.5	-30	14.4	0	7.2	14.4	-1.1
Lower Shell Longitudinal Welds	101-142 A&B	83642	1.77	1.1569	34.4	-30	39.8	0	19.9	39.8	49.6
→ Using credible surveillance data[(f)]			1.77	1.1569	12.5	-30	14.5	0	7.2	14.5	-1.1

Notes:

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

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

c) ΔRT_{PTS} = CF * FF.

d) $M = 2 * (\sigma_U^2 + \sigma_\Delta^2)^{1/2}$.

e) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin.

f) The BVPS-2 surveillance weld metal is the same weld heat as the BVPS-2 beltline welds (heat 83642). The BVPS-2 surveillance weld data is credible (see Appendix A); therefore, the reduced σ_Δ term of 14°F was utilized for BVPS-2 weld heat 83642. The BVPS-2 surveillance plate material is representative of the BVPS-2 intermediate shell plate B9004-2. The surveillance plate material is non-credible (see Appendix A); therefore, the higher σ_Δ term of 17°F was utilized for BVPS-2 plate B9004-2.

Table 6-4 RT_{PTS} Values for BVPS-2 Extended Beltline Region Materials at 54 EFPY

Material Description	Material ID	Heat Number (Lot Number)	Surface Neutron Fluence ($x10^{19}$ n/cm^2)	Fluence Factor, FF[(a)]	Chemistry Factor (°F)	Initial RT_{NDT}[(b)] (°F)	ΔRT_{PTS}[(d)] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[(e)] (°F)	RT_{PTS}[(f)] (°F)
Upper Shell Plates	B9003-1	A9406-1	0.515	0.8147	91.0	50	74.1	0	17	34	158.1
	B9003-2	B4431-2	0.515	0.8147	83.0	60	67.6	0	17	34	**161.6**
	B9003-3	A9406-2	0.515	0.8147	91.0	50	74.1	0	17	34	158.1
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.515	0.8147	73.71	-50	60.1	0	28	56	66.1
		51912 (3536)	0.515	0.8147	73.71	-70	60.1	0	28	56	46.1
		EAIB	0.515	0.8147	27.0	10[(c)]	22.0	17	11.0	40.5	72.5
		IAGA	0.515	0.8147	41.0	-30	33.4	0	16.7	33.4	36.8
		BOHB	0.515	0.8147	68.0	10[(c)]	55.4	17	27.7	65.0	**130.4**
		BAOED	0.515	0.8147	27.0	-50	22.0	0	11.0	22.0	-6.0
Upper to Intermediate Shell Girth Weld	103-121	4P5174	0.515	0.8147	122.0	-50	99.4	0	28	56.0	105.4
		51922	0.515	0.8147	68.0	-56[(c)]	55.4	17	27.7	65.0	64.4
		AAGC	0.515	0.8147	41.0	-70	33.4	0	16.7	33.4	-3.2
		KOIB	0.515	0.8147	41.0	-60	33.4	0	16.7	33.4	6.8
Inlet Nozzles	B9011-1	2V2436-01-002	0.0298	0.2188	77.0	60	16.8	0	8.4	16.8	93.7
	B9011-2	2V2437-02-001	0.0298	0.2188	96.0	60[(c)]	21.0	17	10.5	40.0	121.0
	B9011-3	2V2445-02-003	0.0298	0.2188	96.0	70	21.0	0	10.5	21.0	112.0
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174	0.0298	0.2188	122.0	-50	26.7	0	13.3	26.7	3.4
		LOHB	0.0298	0.2188	41.0	-60	9.0	0	4.5	9.0	-42.1
		HABJC	0.0298	0.2188	27.0	-70	5.9	0	3.0	5.9	-58.2
		BABBD	0.0298	0.2188	27.0	-70	5.9	0	3.0	5.9	-58.2
		FABGC	0.0298	0.2188	41.0	-80	9.0	0	4.5	9.0	-62.1
		EOBC	0.0298	0.2188	27.0	-60	5.9	0	3.0	5.9	-48.2

Table 6-4 RT_{PTS} Values for BVPS-2 Extended Beltline Region Materials at 54 EFPY

Material Description	Material ID	Heat Number (Lot Number)	Surface Neutron Fluence ($x10^{19}$ n/cm^2)	Fluence Factor, FF$^{(a)}$	Chemistry Factor (°F)	Initial $RT_{NDT}^{(b)}$ (°F)	$\Delta RT_{PTS}^{(d)}$ (°F)	σ_U (°F)	σ_Δ (°F)	Margin$^{(e)}$ (°F)	$RT_{PTS}^{(f)}$ (°F)
		FAAFC	0.0298	0.2188	95.0	-60	20.8	0	10.4	20.8	-18.4
		CCJC	0.0298	0.2188	27.0	-60	5.9	0	3.0	5.9	-48.2
		FAGB	0.0298	0.2188	27.0	-30	5.9	0	3.0	5.9	-18.2
		BAOED	0.0298	0.2188	27.0	-50	5.9	0	3.0	5.9	-38.2
Outlet Nozzles	B9012-1	AV8080-2E9558	0.0151	0.1440	94.0	-10	13.5	0	6.8	13.5	17.1
	B9012-2	AV8120-2E9560	0.0151	0.1440	94.5	-10	13.6	0	6.8	13.6	17.2
	B9012-3	AV8097-2E9559	0.0151	0.1440	93.5	-10	13.5	0	6.7	13.5	16.9
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	0.0151	0.1440	27.0	-70	3.9	0	1.9	3.9	-62.2
		FAAFC	0.0151	0.1440	95.0	-60	13.7	0	6.8	13.7	-32.6
		HAAEC	0.0151	0.1440	41.0	-80	5.9	0	3.0	5.9	-68.2
		HABJC	0.0151	0.1440	27.0	-70	3.9	0	1.9	3.9	-62.2
		HAGB	0.0151	0.1440	27.0	-40	3.9	0	1.9	3.9	-32.2
		GACJC	0.0151	0.1440	41.0	-80	5.9	0	3.0	5.9	-68.2
		JAHB	0.0151	0.1440	41.0	-40	5.9	0	3.0	5.9	-28.2

Notes:

a) FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$.
b) Initial RT_{NDT} values are measured values, unless otherwise noted.
c) Initial RT_{NDT} values are generic.
d) ΔRT_{PTS} = CF * FF.
e) $M = 2 *(\sigma_U^2 + \sigma_\Delta^2)^{1/2}$.
f) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin.

Table 6-5 BVPS-2 Beltline Materials Projected USE Values at 54 EFPY

Material Description	Material ID	Heat Number	Wt % Cu	1/4T EOLE Fluence ($x10^{19}$ n/cm^2)	Initial USE (ft-lbs)	Projected USE Decrease[a] (%)	Projected EOLE USE (ft-lbs)
Intermediate Shell Plate	B9004-1	---	0.065	3.229	83	25	62.3
Intermediate Shell Plate	B9004-2	---	0.06	3.229	79	13[b]	68.7
Lower Shell Plate	B9005-1	---	0.08	3.248	82	25	61.5
Lower Shell Plate	B9005-2	---	0.07	3.248	78	25	58.5
Intermediate to Lower Shell Girth Weld	101-171	83642	0.046	3.229	145	5.8[c]	136.6
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	0.046	1.097	145	4.6[c]	138.3
Lower Shell Longitudinal Welds	101-142 A&B	83642	0.046	1.103	145	4.6[c]	138.3

Notes:

a) Unless otherwise noted, percent USE decreases are based on the closest Cu Wt. % chemistry line (rounding up) on Figure 2 of Regulatory Guide 1.99, Revision 2.
b) Based on BVPS-2 surveillance plate results for B9004-2 [Reference 4].
c) Based on BVPS-2 surveillance weld results for heat 83642 [Reference 4].

Table 6-6 BVPS-2 Extended Beltline Materials Projected USE Values at 54 EFPY

Material Description	Material ID	Heat Number (Lot Number)	Wt % Cu	1/4T EOLE Fluence ($x10^{19}$ n/cm^2)	Initial USE (ft-lbs)	Projected USE Decrease[a,b] (%)	Projected EOLE USE (ft-lbs)
Upper Shell Plates	B9003-1	A9406-1	0.13	0.321	98	18.5	79.9
	B9003-2	B4431-2	0.12	0.321	80	18.5	65.2
	B9003-3	A9406-2	0.13	0.321	98	18.5	79.9
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.156	0.321	96	26	71.0
		51912 (3536)	0.156	0.321	114	26	84.4
		EAIB	0.02	0.321	118	14.5	100.9
		IAGA	0.03	0.321	160	14.5	136.8
		BOHB	0.05	0.321	97	14.5	82.9
		BAOED	0.02	0.321	150	14.5	128.3
Upper to Intermediate Shell Girth Weld	103-121	4P5174	0.09	0.321	70	18.5	57.1
		51922	0.05	0.321	102	14.5	87.2
		AAGC	0.03	0.321	111	14.5	94.9
		KOIB	0.03	0.321	110	14.5	94.1
Inlet Nozzles	B9011-1	2V2436-01-002	0.11	0.019	68.25	9.5	61.8
	B9011-2	2V2437-02-001	0.13	0.019	75.4	9.5	68.2
	B9011-3	2V2445-02-003	0.13	0.019	79.3	9.5	71.8
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174	0.09	0.019	70	9.5	63.4
		LOHB	0.03	0.019	137	7.5	126.7
		HABJC	0.02	0.019	162	7.5	149.9
		BABBD	0.02	0.019	142	7.5	131.4
		FABGC	0.03	0.019	119	7.5	110.1
		EOBC	0.02	0.019	127	7.5	117.5
		FAAFC	0.07	0.019	119	9.5	107.7
		CCJC	0.02	0.019	109	7.5	100.8
		FAGB	0.02	0.019	114	7.5	105.5
		BAOED	0.02	0.019	150	7.5	138.8
Outlet Nozzles	B9012-1	AV8080-2E9558	0.13	0.009	146	9.5	132.1
	B9012-2	AV8120-2E9560	0.13	0.009	128	9.5	115.8
	B9012-3	AV8097-2E9559	0.13	0.009	127	9.5	114.9
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	0.02	0.009	142	7.5	131.4
		FAAFC	0.07	0.009	119	9.5	107.7
		HAAEC	0.03	0.009	178	7.5	164.7
		HABJC	0.02	0.009	162	7.5	149.9
		HAGB	0.02	0.009	194	7.5	179.5
		GACJC	0.03	0.009	113	7.5	104.5
		JAHB	0.03	0.009	149	7.5	137.8

Notes:

a) The lower line in Figure 2 of Regulatory Guide 1.99, Revision 2, was used as the bounding value when Wt % Cu values were below this limit for plates and/or welds.

b) The minimum fluence value (2 x 10^{17} n/cm^2) displayed on Figure 2 of Regulatory Guide 1.99, Revision 2 was conservatively used to determine the projected USE decrease values for the inlet/outlet nozzle forgings and welds.

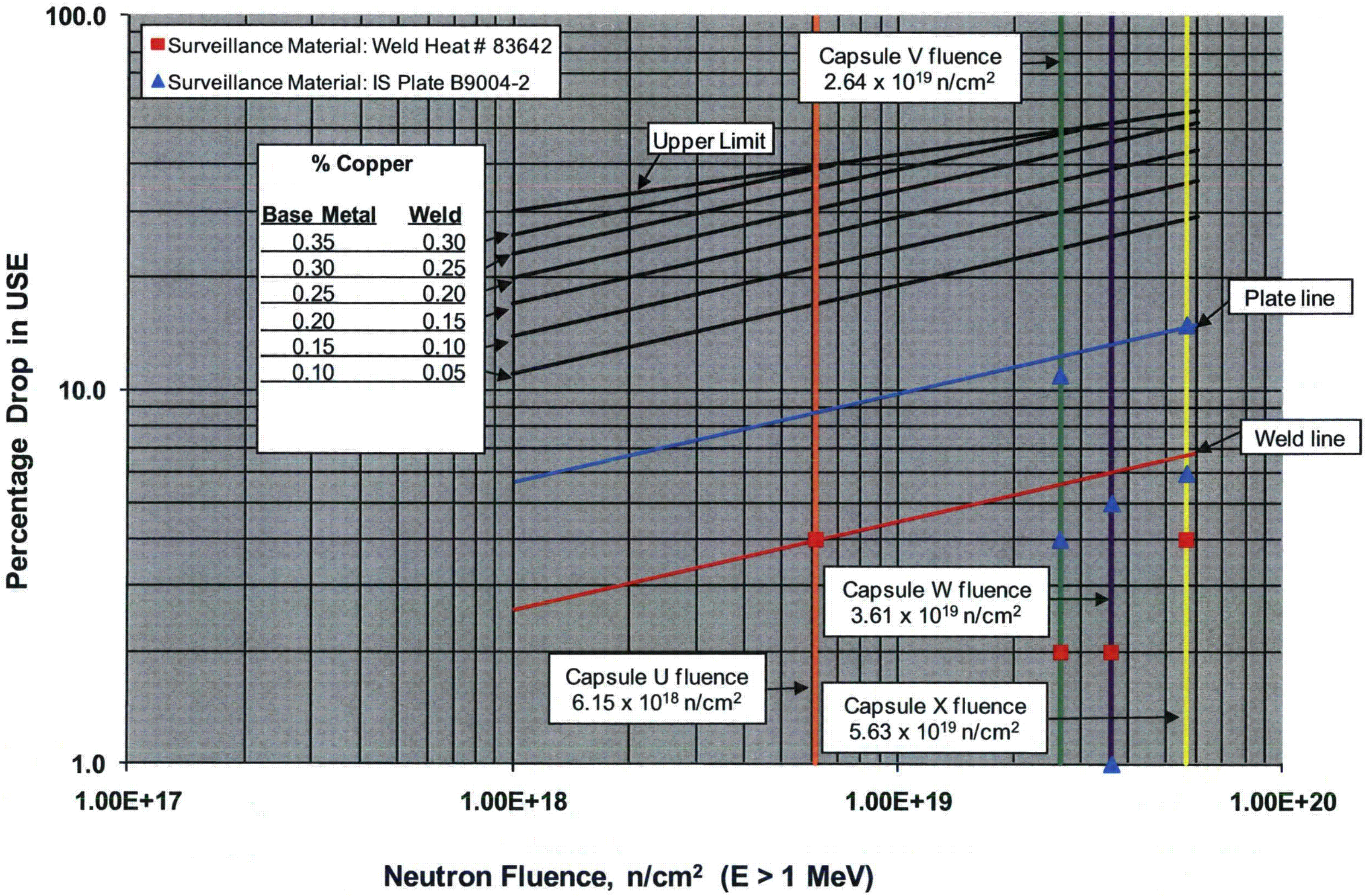

Figure 6-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE as a Function of Copper and Fluence for BVPS-2

7 PRESSURE-TEMPERATURE LIMIT CURVES APPLICABILITY

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} corresponding to the limiting material in the beltline region of the reactor pressure vessel. The most limiting RT_{NDT} of the material in the core (beltline) region of the reactor pressure vessel is determined by using the unirradiated reactor pressure vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}). RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature (P-T) limit curves are determined in accordance with Appendix G to Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code [Reference 10] as specified by CFR Part 50, Appendix G [Reference 11].

The BVPS-2 P-T limit curves for normal heatup and cooldown of the primary reactor coolant system were previously developed in WCAP-16528-NP, Revision 1 [Reference 5], for 22 EFPY, 30 EFPY, 40, and 54 EFPY. Currently, BVPS-2 is operating to their 22 EFPY curves. The existing 22 EFPY P-T limit curves are based on the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material.

The BVPS-2 22 EFPY P-T limit curves were developed by calculating ART values utilizing the maximum clad/base metal fluence across all beltline materials. The limiting ART values used in the development of the 22 EFPY P-T limit curves correspond to the intermediate shell plate B9004-1 (Position 1.1 – without using surveillance data).

Taking into account the updated fluence values, as shown in Section 5 of this report, as well as the updated Position 2.1 chemistry factor values in Section 6, the intermediate shell plate B9004-1 continues to be the limiting material for the current BVPS-2 P-T limit curves. Additionally, the BVPS-2 updated vessel and surveillance capsule fluence values do not reduce the existing 22 EFPY applicability term for which the P-T limit curves were originally developed; therefore, the existing P-T limit curves remain valid as documented in Reference 5 for BVPS-2.

8 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 [Reference 12] and is recommended for future capsules to be withdrawn from the BVPS-2 reactor vessel.

Table 8-1 Recommended Surveillance Capsule Withdrawal Schedule for BVPS-2

Capsule	Capsule Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence, f[(a)] [n/cm², E > 1.0 MeV]
U	343°	3.17	1.25	6.15×10^{18}
V	107°	3.64	5.99	2.64×10^{19}
W	110°	3.25	9.84	3.61×10^{19}
X	287°	3.69	14.00	5.63×10^{19}
Y[(c)]	290°	3.17	(c)	6.53×10^{19}[(c)]
Z[(c)]	340°	3.17	(c)	6.53×10^{19}[(c)]
A[(d)]	107°	3.58	32[(d)]	3.40×10^{19}[(d)]

Notes:

a) Updated in recent fluence analysis; see Section 5 of this report.

b) Effective Full Power Years (EFPY) from plant startup.

c) Either Capsule Y or Z is to be withdrawn in accordance with the recommendations in ASTM E185-82 for a 60-year and 80-year license renewal. Accumulated fluence value through EOC 15. It is recommended that one of these capsules be pulled at a time when the capsule fluence exceeds one times the projected peak 80-year EOL fluence, but before the capsule fluence reaches two times the projected 60-year EOL fluence. Therefore, based on the current information, Capsule Y or Z should be withdrawn between 20.5 EFPY and 32.4 EFPY. In order to be consistent with NUREG-1929 [Reference 13], this capsule should be pulled at the refueling outage closest to 26.1 EFPY, which corresponds with a peak vessel fluence level of 8.5×10^{19} n/cm^2 (E > 1.0 MeV). The second capsule will remain in the vessel to provide fluence monitoring or future testing. The withdrawal of this capsule will be revisited at a later time.

d) Supplemental Capsule A contains BVPS-1 initially unirradiated material specimens, as well as previously irradiated material specimens from BVPS-1, St. Lucie, and Fort Calhoun. Supplemental Capsule A was inserted in the vacant Capsule V location at the EOC 8. Accumulated fluence value through EOC 15. It is to be removed and tested when it reaches a fluence value equivalent to the peak 80-year vessel fluence for BVPS-1. This is expected to occur at approximately 32 EFPY.

9 CONCLUSION

- All of the beltline and extended beltline region materials in the BVPS-2 reactor vessel have EOLE RT_{PTS} values well below the screening criteria values of 270°F for forgings/plates and 300°F for circumferential welds at EOLE (54 EFPY).

- All of the USE values for the beltline and extended beltline materials are greater than 50 ft-lbs at EOLE (54 EFPY).

- The current P-T limit curves remain valid through 22 EFPY for BVPS-2.

- Four surveillance capsules have been withdrawn and tested from BVPS-2. With the withdrawal of these four capsules, BVPS-2 has satisfied the current requirements for their 40-year license.

10 REFERENCES

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

2. Code of Federal Regulations, 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, effective January 18, 1996.

3. Code of Federal Regulations, 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

4. WCAP-16527-NP, Revision 0, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," B.N. Burgos, et. al., March 2006.

5. WCAP-16528-NP, Revision 1, "Beaver Valley Power Station Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," N. R. Jurcevich, June 2008.

6. Combustion Engineering Report MISC-PENG-ER-021, Revision 00, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the Beaver Valley Unit 2 Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," J.P. Molkenthin, et. al., October 1995.

7. "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CEOG Report CE NPSD-1039, Revision 2, ABB Combustion Engineering, June 1997.

8. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, MTEB 5-2 and 5-3, June 1987.

9. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

10. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."

11. Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

13. NUREG-1929, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2 (Docket Nos. 50-334 and 50-412) FirstEnergy Nuclear Operating Company," October 2009 (ADAMS Accession Number ML093020276).

APPENDIX A
BEAVER VALLEY UNIT 2 SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

INTRODUCTION

Regulatory Guide 1.99, Revision 2 [Reference A-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date, there have been four surveillance capsules removed and tested from the BVPS-2 reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with Regulatory Guide 1.99, Revision 2, the credibility of the surveillance data will be judged based on five criteria.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the BVPS-2 reactor vessel surveillance data in order to determine if that surveillance data is credible.

EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," [Reference A-2] as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The BVPS-2 reactor vessel beltline region consists of the following materials:

1. Intermediate Shell Plates B9004-1 and B9004-2
2. Lower Shell Plates B9005-1 and B9005-2
3. Intermediate Shell Longitudinal Welds 101-124 A&B (Heat 83642)
4. Intermediate to Lower Shell Girth Weld 101-171 (Heat 83642)
5. Lower Shell Longitudinal Welds 101-142 A&B (Heat 83642)

Per WCAP-9615, Revision 1 [Reference A-3], at the time the BVPS-2 surveillance program was being developed, selection of the surveillance material was based on an evaluation of initial toughness (characterized by the reference temperature, RT_{NDT}, and upper-shelf energies), the predicted effect of chemical composition (residual copper and phosphorus) and neutron fluence on the toughness (RT_{NDT} shift) during reactor operation. Intermediate shell plate B9004-2 was selected as the surveillance base metal since it had one of the highest adjusted EOL RT_{NDT} of the four beltline region plates. Weld heat 83642 was selected because it is the same heat used in the fabrication of all of the longitudinal and girth beltline welds.

Based on the discussion above, Criterion 1 is met for the BVPS-2 surveillance program.

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

Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions are presented in Section 5 and Appendix C of the latest surveillance capsule report, WCAP-16527-NP, Revision 0 [Reference A-4].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the BVPS-2 surveillance materials unambiguously.

Hence, Criterion 2 is met for the BVPS-2 surveillance program.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Reference A-5].

The functional form of the least-squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the plate.

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

Following the NRC Case 1 guidelines, the BVPS-2 surveillance plate and weld metal (heat 83642) will be evaluated using the BVPS-2 data. This evaluation is contained in Table A-1. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values. In addition, only BVPS-2 data is being considered; therefore, no temperature adjustment is required.

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

Material	Capsule	Capsule f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate B9004-2 (Longitudinal)	U	0.615	0.864	24.0	20.73	0.746
	V	2.64	1.260	56.0	70.54	1.587
	W	3.61	1.334	71.0	94.68	1.778
	X	5.63	1.425	98.0	139.65	2.031
Intermediate Shell Plate B9004-2 (Transverse)	U	0.615	0.864	17.7	15.29	0.746
	V	2.64	1.260	46.1	58.07	1.587
	W	3.61	1.334	63.4	84.55	1.778
	X	5.63	1.425	104.1	148.34	2.031
				SUM:	631.87	12.284
	$CF_{IS\ Plate\ B9004-2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (631.87) \div (12.284) = 51.4$°F					
Beaver Valley Unit 2 Weld Metal (Heat 83642)	U	0.615	0.864	4.1	3.54	0.746
	V	2.64	1.260	25.7	32.37	1.587
	W	3.61	1.334	6.0	8.00	1.778
	X	5.63	1.425	22.9	32.63	2.031
				SUM :	76.55	6.142
	$CF_{83642} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (76.55) \div (6.142) = 12.5$°F					

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table A-2.

Table A-2 Best-Fit Evaluation for Beaver Valley Unit 2 Surveillance Materials Only

Material	Capsule	CF (Slope$_{best-fit}$) (°F)	Capsule f (x10^{19} n/cm^2, E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT} (°F)	<17°F (Base Metal) <28°F (Weld)
Intermediate Shell Plate B9004-2 (Longitudinal)	U	51.4	0.615	0.864	24.0	44.4	20.4	No
	V	51.4	2.64	1.260	56.0	64.8	8.8	Yes
	W	51.4	3.61	1.334	71.0	68.6	2.4	Yes
	X	51.4	5.63	1.425	98.0	73.3	24.7	No
Intermediate Shell Plate B9004-2 (Transverse)	U	51.4	0.615	0.864	17.7	44.4	26.7	No
	V	51.4	2.64	1.260	46.1	64.8	18.7	No
	W	51.4	3.61	1.334	63.4	68.6	5.2	Yes
	X	51.4	5.63	1.425	104.1	73.3	30.8	No
Beaver Valley Unit 2 Weld Metal (Heat 83642)	U	12.5	0.615	0.864	4.1	10.8	6.7	Yes
	V	12.5	2.64	1.260	25.7	15.7	10.0	Yes
	W	12.5	3.61	1.334	6.0	16.6	10.6	Yes
	X	12.5	5.63	1.425	22.9	17.8	5.1	Yes

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal. Table A-2 indicates that only three of the eight surveillance data points fall within the +/- 1σ of 17°F scatter band for surveillance base metals; therefore, the intermediate shell plate B9004-2 data is deemed "non-credible" per the third criterion.

Note that the original Capsule X report [Reference A-4] deemed this material "credible" when considering an increased allowable scatter band; however, the reduced margin term that would be used in subsequent reactor vessel integrity evaluations is not needed for this BVPS-2 material. Therefore, the "non-credible" conclusion in this updated evaluation is considered a conservative approach and will be utilized for the BVPS-2 intermediate shell plate B9004-2.

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 28°F for weld metal. Table A-2 indicates that all four surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material (heat 83642) is deemed "credible" per the third criterion.

Note that although the intermediate shell plate B9004-2 did not meet Criterion 3, it may still be used in determining the upper-shelf energy decrease in accordance with Regulatory Guide 1.99, Revision 2, Position 2.2.

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 BVPS-2 capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pad. The location of the specimens with respect to the reactor vessel beltline ensures that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F.

Hence, Criterion 4 is met for the BVPS-2 surveillance program.

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

The BVPS-2 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the BVPS-2 surveillance program.

CONCLUSION:

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2, Section B, the BVPS-2 surveillance plate data is deemed non-credible, whereas the surveillance weld data is deemed credible.

REFERENCES

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

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

A-3 WCAP-9615, Revision 1, "Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Peter, June 1995.

A-4 WCAP-16527-NP, Revision 0, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," B. N. Burgos et al., March 2006.

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

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