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Revision 2

September 2011

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Westinghouse

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RECORD OF REVISIONS

Revision 0: Original Issue

Revision 1: The purpose of this revision is to address CAPS Issue 08-059-M009, which resulted in corrections made to Pages 6-1 and 6-5. In addition, editorial changes were made, including an update to Reference 4 to refer to Revision 1 of WCAP-15571.

Revision 2: 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 and sister plant data, the surveillance capsule credibility evaluation has also been updated since the Capsule Y 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 1 and 2 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 Power Station Unit 1 (BVPS-1) reactor vessel beltline and extended beltline materials. This analysis will be based on the results of the latest surveillance capsule Y evaluation, sister plant surveillance data, and the implementation of the Extended Power Uprate (EPU) program. These calculations are performed for End-of-License Extension (EOLE) at 50 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 BVPS-1 beltline is the lower shell plate B6903-1 with a projected EOLE RT_{PTS} value of 277.0°F using the BVPS-1 surveillance capsule data for 50 EFPY (equivalent to a fluence of 5.57×10^{19} n/cm^2 (E > 1.0 MeV)). This value is slightly above the screening criterion of 270°F for forgings/plates in 10 CFR 50 Part 61. The screening limit of 270°F for lower shell plate B6903-1 will be reached at a fluence level of 4.407×10^{19} n/cm^2 (E > 1.0 MeV), which is equivalent to 39.6 EFPY. The limiting weld material in the BVPS-1 reactor vessel beltline is the lower shell longitudinal weld (heat 305414) with an EOLE RT_{PTS} value of 231.6°F using Fort Calhoun surveillance capsule sister plant data. This RT_{PTS} value is well below the screening criteria value of 270°F for axial welds at EOLE (50 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 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 RT_{PTS} and USE values for the BVPS-1 reactor vessel using the results of the surveillance Capsule Y evaluation, sister plant data, and the implementation of the EPU Program. The results presented in this report are for EOLE at 50 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-1 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 reactor vessel integrity evaluations follow in Sections 9.0 and 10.0, respectively.

The surveillance capsule credibility analysis, based on the results of the surveillance Capsule Y 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}.

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

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.

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

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 Regulatory Guide 1.99, Revision 2. 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 Regulatory Guide 1.99, Revision 2, 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 PTS evaluation, a review of the latest plant-specific material properties for the BVPS-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 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 50 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-1 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-1 reactor vessel is provided in Tables 4-1 and 4-2.

Table 4-1 BVPS-1 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	B6607-1	---	0.14	0.62	43	94
Intermediate Shell Plate	B6607-2	---	0.14	0.62	73	83
Lower Shell Plate	B6903-1	---	0.21	0.54	27	83
Lower Shell Plate	B7203-2	---	0.14	0.57	20	85
Intermediate to Lower Shell Girth Weld	11-714	90136	0.27	0.07	-56	144
Intermediate Shell Longitudinal Welds	19-714 A&B	305424	0.28	0.63	-56	112
Lower Shell Longitudinal Welds	20-714 A&B	305414	0.34	0.61	-56	>100
Surveillance Weld	---	305424	0.26	0.61	---	---

Notes:

a) Materials information taken from WCAP-16799-NP [Reference 5] and WCAP-15571 [Reference 4].

b) The Initial RT_{NDT} values are measured values for the plates while the weld values are generic.

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

Material Description	Material ID	Heat Number (Lot Number)	Wt % Cu	Wt % Ni	Initial RT_{NDT}[(e)] (°F)	Initial USE (ft-lbs)
Upper Shell Forging	B6604	123V339VA1	0.12[(b)]	0.68	40	155 (101)[(c)]
Upper to Intermediate Shell Girth Weld	10-714	305414 (3951)	0.337[(d)]	0.609[(d)]	-56 (Gen)	97[(f)]
		305414 (3958)	0.337[(d)]	0.609[(d)]	-56 (Gen)	97[(f)]
		AOFJ	0.03	0.93	10 (Gen)	111
		FOIJ	0.03	0.94	10 (Gen)	104
		EODJ	0.02	1.04	10 (Gen)	156
		HOCJ	0.02	0.93	10 (Gen)	160
Inlet Nozzles	B6608-1	95443-1	0.10	0.82	60 (Gen)	82.5
	B6608-2	95460-1	0.10	0.82	60 (Gen)	94
	B6608-3	95712-1	0.08	0.79	60 (Gen)	97
Inlet Nozzle Welds	1-717B 1-717D 1-717F	EODJ	0.02	1.04	10 (Gen)	156
		FOIJ	0.03	0.94	10 (Gen)	104
		HOCJ	0.02	0.93	10 (Gen)	160
		DBIJ	0.02	0.97	10 (Gen)	123
		EOEJ	0.01	1.03	10 (Gen)	152
		ICJJ	0.03	0.99	10 (Gen)	123
		JACJ	0.04	0.97	10 (Gen)	116
Outlet Nozzles	B6605-1	95415-1	0.13[(g)]	0.77	60 (Gen)	93
	B6605-2	95415-2	0.13[(g)]	0.77	60 (Gen)	112.5
	B6605-3	95444-1	0.09	0.79	60 (Gen)	103
Outlet Nozzle Welds	1-717A 1-717C 1-717E	ICJJ	0.03	0.99	10 (Gen)	123
		IOBJ	0.02	0.97	10 (Gen)	122
		JACJ	0.04	0.97	10 (Gen)	116
		HOCJ	0.02	0.93	10 (Gen)	160
		EODJ	0.02	1.04	10 (Gen)	156
		FOIJ	0.03	0.94	10 (Gen)	104

Notes:

a) All of the materials data is obtained from Combustion Engineering report MISC-PENG-ER-022 [Reference 6] except as noted.
b) The Cu wt % was not available from the CMTR so in accordance with Regulatory Guide 1.99, Revision 2, a standard deviation analysis (average + standard deviation) was done to determine the value based on Westinghouse 508 Class 2 Shell Forgings (55 data points).
c) Value in parenthesis is the 65% value per NUREG-0800 [Reference 8].
d) Chemistry obtained from CE Report NPSD-1039, Revision 2 [Reference 7].
e) The initial RT_{NDT} value for the upper shell forging is a measured value. The generic initial RT_{NDT} values for the remaining materials were determined in accordance with NUREG-0800 [Reference 8] and 10 CFR 50.61 [Reference 2].
f) The USE for Linde flux type 1092 welds documented in CEN-622-A. [Reference 9].
g) The Cu wt % was not available from the CMTR, so in accordance with Regulatory Guide 1.99, Revision 2, a standard deviation analysis (average + standard deviation) was done to determine the value based on Westinghouse 508 Class 2 Nozzle Forgings (178 data points).

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-1 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, 15°, 25°, 35°, and 45°, 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 Y radiation analysis and comply with Reg. Guide 1.190 [Reference 10].

These fluence data tabulations include fuel-cycle-specific calculated neutron exposures at the end of Cycle 20 (the last completed at BVPS-1), as well as future projections to the end of Cycle 21 (the current operating cycle) and for several intervals extending to 60 EFPY.

Neutron exposure projections beyond the end of Cycle 21 were based on the spatial power distributions and associated plant characteristics of Cycles 20 and 21 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-1

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]			
		0°	15°	30°	45°
1	1.16	1.86×10^{18}	9.03×10^{17}	4.88×10^{17}	3.25×10^{17}
2	1.88	2.98×10^{18}	1.45×10^{18}	7.92×10^{17}	5.30×10^{17}
3	2.67	4.43×10^{18}	2.15×10^{18}	1.16×10^{18}	7.71×10^{17}
4	3.59	5.68×10^{18}	2.78×10^{18}	1.49×10^{18}	9.88×10^{17}
5	4.78	7.26×10^{18}	3.56×10^{18}	1.91×10^{18}	1.27×10^{18}
6	5.89	8.40×10^{18}	4.21×10^{18}	2.31×10^{18}	1.53×10^{18}
7	7.14	9.97×10^{18}	5.03×10^{18}	2.74×10^{18}	1.82×10^{18}
8	8.24	1.13×10^{19}	5.76×10^{18}	3.13×10^{18}	2.08×10^{18}
9	9.62	1.29×10^{19}	6.60×10^{18}	3.62×10^{18}	2.42×10^{18}
10	10.81	1.39×10^{19}	7.17×10^{18}	4.00×10^{18}	2.70×10^{18}
11	11.78	1.47×10^{19}	7.62×10^{18}	4.32×10^{18}	2.93×10^{18}
12	12.92	1.57×10^{19}	8.19×10^{18}	4.71×10^{18}	3.19×10^{18}
13	14.29	1.70×10^{19}	8.88×10^{18}	5.16×10^{18}	3.50×10^{18}
14	15.61	1.83×10^{19}	9.47×10^{18}	5.50×10^{18}	3.74×10^{18}
15	16.94	1.93×10^{19}	1.00×10^{19}	5.86×10^{18}	4.00×10^{18}
16	18.38	2.07×10^{19}	1.08×10^{19}	6.28×10^{18}	4.29×10^{18}
17	19.61	2.18×10^{19}	1.13×10^{19}	6.61×10^{18}	4.52×10^{18}
18	20.99	2.33×10^{19}	1.21×10^{19}	7.02×10^{18}	4.79×10^{18}

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

19	22.46	2.47×10^{19}	1.29×10^{19}	7.46×10^{18}	5.12×10^{18}
20	23.81	2.62×10^{19}	1.36×10^{19}	7.87×10^{18}	5.41×10^{18}
21	25.15	2.78×10^{19}	1.43×10^{19}	8.25×10^{18}	5.68×10^{18}
Future	32.00	3.55×10^{19}	1.81×10^{19}	1.03×10^{19}	7.12×10^{18}
Future	48.00	5.36×10^{19}	2.70×10^{19}	1.50×10^{19}	1.05×10^{19}
Future	54.00	6.03×10^{19}	3.03×10^{19}	1.68×10^{19}	1.17×10^{19}
Future	60.00	6.71×10^{19}	3.36×10^{19}	1.85×10^{19}	1.30×10^{19}

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

Material	Neutron Fluence (E > 1.0 MeV) [n/cm^2]				
	32 EFPY	48 EFPY	50 EFPY$^{(a)}$	54 EFPY	60 EFPY
Beltline Materials					
Intermediate Shell Plates	3.54×10^{19}	5.35×10^{19}	5.57×10^{19}	6.02×10^{19}	6.70×10^{19}
Lower Shell Plates	3.55×10^{19}	5.35×10^{19}	5.57×10^{19}	6.02×10^{19}	6.70×10^{19}
Intermediate to Lower Shell Girth Weld	3.53×10^{19}	5.33×10^{19}	5.55×10^{19}	6.00×10^{19}	6.67×10^{19}
Intermediate Shell Longitudinal Welds	7.09×10^{18}	1.04×10^{19}	1.08×10^{19}	1.17×10^{19}	1.30×10^{19}
Lower Shell Longitudinal Welds	7.11×10^{18}	1.05×10^{19}	1.09×10^{19}	1.17×10^{19}	1.30×10^{19}
Extended Beltline Materials					
Upper Shell Forging	3.87×10^{18}	5.99×10^{18}	6.25×10^{18}	6.78×10^{18}	7.58×10^{18}
Upper to Intermediate Shell Girth Weld	3.87×10^{18}	5.99×10^{18}	6.25×10^{18}	6.78×10^{18}	7.58×10^{18}
Inlet Nozzle to Upper Shell Weld – Lowest Extent	1.00×10^{17}	1.56×10^{17}	1.63×10^{17}	1.77×10^{17}	1.98×10^{17}
Outlet Nozzle to Upper Shell Weld – Lowest Extent	6.93×10^{16}	1.08×10^{17}	1.13×10^{17}	1.23×10^{17}	1.37×10^{17}
Lower Shell to Lower Closure Head Weld$^{(b)}$	6.13×10^{15}	9.36×10^{15}	---	1.06×10^{16}	1.18×10^{16}

Notes:

a) Fluence values at 50 EFPY were linearly interpolated using the values at 48 and 54 EFPY for each of the reactor vessel materials.

b) 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-1

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
		15°	25°	35°	45°
1	1.16	2.99×10^{18}	1.94×10^{18}	1.31×10^{18}	1.01×10^{18}
2	1.88	4.89×10^{18}	3.19×10^{18}	2.15×10^{18}	1.67×10^{18}
3	2.67	7.25×10^{18}	4.70×10^{18}	3.16×10^{18}	2.44×10^{18}
4	3.59	9.33×10^{18}	6.04×10^{18}	4.04×10^{18}	3.12×10^{18}
5	4.78	1.20×10^{19}	7.72×10^{18}	5.16×10^{18}	4.00×10^{18}
6	5.89	1.41×10^{19}	9.30×10^{18}	6.22×10^{18}	4.81×10^{18}
7	7.14	1.68×10^{19}	1.10×10^{19}	7.36×10^{18}	5.71×10^{18}
8	8.24	1.92×10^{19}	1.26×10^{19}	8.38×10^{18}	6.49×10^{18}
9	9.62	2.21×10^{19}	1.46×10^{19}	9.74×10^{18}	7.59×10^{18}
10	10.81	2.39×10^{19}	1.60×10^{19}	1.08×10^{19}	8.42×10^{18}
11	11.78	2.54×10^{19}	1.72×10^{19}	1.17×10^{19}	9.18×10^{18}
12	12.92	2.72×10^{19}	1.87×10^{19}	1.27×10^{19}	9.96×10^{18}
13	14.29	2.95×10^{19}	2.05×10^{19}	1.40×10^{19}	1.09×10^{19}
14	15.61	3.14×10^{19}	2.18×10^{19}	1.49×10^{19}	1.17×10^{19}
15	16.94	3.32×10^{19}	2.32×10^{19}	1.59×10^{19}	1.25×10^{19}
16	18.38	3.56×10^{19}	2.48×10^{19}	1.70×10^{19}	1.34×10^{19}
17	19.61	3.76×10^{19}	2.61×10^{19}	1.79×10^{19}	1.41×10^{19}
18	20.99	4.01×10^{19}	2.77×10^{19}	1.90×10^{19}	1.49×10^{19}
19	22.46	4.26×10^{19}	2.95×10^{19}	2.03×10^{19}	1.60×10^{19}
20	23.81	4.51×10^{19}	3.11×10^{19}	2.14×10^{19}	1.69×10^{19}
21	25.15	4.76×10^{19}	3.26×10^{19}	2.25×10^{19}	1.78×10^{19}
Future	32.00	6.03×10^{19}	4.06×10^{19}	2.81×10^{19}	2.23×10^{19}
Future	48.00	8.99×10^{19}	5.93×10^{19}	4.13×10^{19}	3.29×10^{19}
Future	54.00	1.01×10^{20}	6.63×10^{19}	4.62×10^{19}	3.68×10^{19}
Future	60.00	1.12×10^{20}	7.34×10^{19}	5.11×10^{19}	4.08×10^{19}

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

6.1 BVPS-1 RT_{PTS} CALCULATIONS FOR 50 EFPY

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline and extended beltline region materials of the BVPS-1 reactor vessel for fluence values at EOLE (50 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For BVPS-1, the related surveillance program results have been included in this PTS evaluation. Specifically, the BVPS-1 plant-specific surveillance capsule data for the lower shell (LS) plate B6903-1 and weld metal (heat 305424) is provided and applied as follows:

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

2) The data for the BVPS-1 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 BVPS-1 surveillance program weld material is deemed non-credible. The data was used with a σ_Δ margin of 28°F.

4) The surveillance capsule materials are representative of the actual vessel plate (B6903-1) and intermediate shell longitudinal weld metal (weld heat 305424).

5) The resulting RT_{PTS} values for lower shell plate B6903-1 exceed the screening criteria at 50 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 remain below the PTS Rule screening criteria at 50 EFPY.

The BVPS-1 reactor vessel intermediate to lower shell girth weld and lower shell longitudinal welds were fabricated using weld heats 90136 and 305414, respectively. These weld heats are not contained in the BVPS-1 surveillance capsule program; however, the St. Lucie Unit 1 surveillance capsule program contains weld heat 90136 and the Fort Calhoun surveillance capsule program contains weld heat 305414. Therefore, the sister plant data from St. Lucie Unit 1 and Fort Calhoun are applied to the applicable BVPS-1 evaluations. The data for the St. Lucie surveillance program weld material (heat 90136) is deemed credible; whereas the data for the Fort Calhoun surveillance program weld material (heat 305414) is deemed non-credible. Appendix A of this report contains the credibility evaluation for these materials.

Chemistry factor values for the BVPS-1 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-1 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 50 EFPY, respectively.

6.2 BVPS-1 UPPER-SHELF ENERGY CALCULATIONS FOR 50 EFPY

Surveillance data exists for plate B6903-1 and weld heat 305424 for BVPS-1. 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 50 EFPY. All of the beltline and extended beltline material USE values maintain 50 ft-lbs or greater at 50 EFPY.

Table 6-1 BVPS-1 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	B6607-1	---	100.5	---
Intermediate Shell Plate	B6607-2	---	100.5	---
Lower Shell Plate	B6903-1	---	147.2	151.8
Lower Shell Plate	B7203-2	---	98.7	---
Intermediate to Lower Shell Girth Weld	11-714	90136	124.3	87.1
Intermediate Shell Longitudinal Welds	19-714 A&B	305424	191.7	192.3
Lower Shell Longitudinal Welds	20-714 A&B	305414	210.5	216.9
Surveillance Weld	---	305424	181.6	---

Table 6-2 BVPS-1 Extended 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 (Lot Number)	Chemistry Factor (°F)	
			Position 1.1	Position 2.1
Upper Shell Forging	B6604	123V339VA1	84.2	---
Upper to Intermediate Shell Girth Weld	10-714	305414 (3951 & 3958)	209.11	216.9
		AOFJ	41.0	---
		FOIJ	41.0	---
		EODJ	27.0	---
		HOCJ	27.0	---
Inlet Nozzles	B6608-1	95443-1	67.0	---
	B6608-2	95460-1	67.0	---
	B6608-3	95712-1	51.0	---
Inlet Nozzle Welds	1-717B 1-717D 1-717F	EODJ	27.0	---
		FOIJ	41.0	---
		HOCJ	27.0	---
		DBIJ	27.0	---
		EOEJ	20.0	---
		ICJJ	41.0	---
		JACJ	54.0	---
Outlet Nozzles	B6605-1	95415-1	95.25	---
	B6605-2	95415-2	95.25	---
	B6605-3	95444-1	58.0	---
Outlet Nozzle Welds	1-717A 1-717C 1-717E	ICJJ	41.0	---
		IOBJ	27.0	---
		JACJ	54.0	---
		HOCJ	27.0	---
		EODJ	27.0	---
		FOIJ	41.0	---

Table 6-3 RT_{PTS} Values for BVPS-1 Beltline Region Materials at 50 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	B6607-1	---	5.57	1.4231	100.5	43	143.0	0	17	34	220.0
Intermediate Shell Plate	B6607-2	---	5.57	1.4231	100.5	73	143.0	0	17	34	250.0
Lower Shell Plate	B6903-1	---	5.57	1.4231	147.2	27	209.5	0	17	34	270.5
→ Using non-credible surveillance data[(f)]			5.57	1.4231	151.8	27	216.0	0	17[(f)]	34	**277.0**
Lower Shell Plate	B7203-2	---	5.57	1.4231	98.7	20	140.5	0	17	34	194.5
Intermediate to Lower Shell Girth Weld	11-714	90136	5.55	1.4225	124.3	-56	176.8	17	28	65.5	186.3
→ Using credible surveillance data[(g)]			5.55	1.4225	87.1	-56	123.9	17	14[(g)]	44.0	111.9
Intermediate Shell Longitudinal Weld	19-714 A&B	305424	1.08	1.0224	191.7	-56	196.0	17	28	65.5	205.5
→ Using non-credible surveillance data[(f)]			1.08	1.0224	192.3	-56	196.6	17	28[(f)]	65.5	206.1
Lower Shell Longitudinal Weld	20-714 A&B	305414	1.09	1.0241	210.5	-56	215.6	17	28	65.5	225.1
→ Using non-credible surveillance data[(h)]			1.09	1.0241	216.9	-56	222.1	17	28[(h)]	65.5	**231.6**

Notes:

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

b) Initial RT_{NDT} values are measured values with the exception of the vessel welds.

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-1 surveillance weld metal is the same weld heat as the BVPS-1 intermediate shell longitudinal welds (heat 305424). The BVPS-1 surveillance weld data is non-credible (see Appendix A); therefore, the higher σ_Δ term of 28°F was utilized for BVPS-1 weld heat 305424. The BVPS-1 surveillance plate material is representative of the BVPS-1 lower shell plate B6903-1. The surveillance plate material is non-credible (see Appendix A); therefore, the higher σ_Δ term of 17°F was utilized for BVPS-1 plate B6903-1.

g) The St. Lucie Unit 1 surveillance weld metal is the same weld heat as the BVPS-1 intermediate to lower shell girth weld (heat 90136). The St. Lucie Unit 1 surveillance weld data is credible (see Appendix A); therefore, the reduced σ_Δ term of 14°F was utilized for BVPS-1 weld heat 90136.

h) The Fort Calhoun surveillance weld metal is the same weld heat as the BVPS-1 lower shell longitudinal welds (heat 305414). The Fort Calhoun surveillance weld data is non-credible (see Appendix A); therefore, the higher σ_Δ term of 28°F was utilized for BVPS-1 weld heat 305414.

Table 6-4 RT_{PTS} Values for BVPS-1 Extended Beltline Region Materials at 50 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}[(c)] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[(d)] (°F)	RT_{PTS}[(e)] (°F)
Upper Shell Forging	B6604	123V339VA1	0.625	0.8685	84.2	40	73.1	0	17	34	147.1
Upper to Intermediate Shell Girth Weld	10-714	305414 (3951 & 3958)	0.625	0.8685	209.11	-56	181.6	17	28	65.5	191.1
→ Using non-credible surveillance data[(f)]			0.625	0.8685	216.9	-56	188.4	17	28[(f)]	65.5	197.9
Upper to Intermediate Shell Girth Weld	10-714	AOFJ	0.625	0.8685	41.0	10	35.6	17	17.8	49.2	94.8
		FOIJ	0.625	0.8685	41.0	10	35.6	17	17.8	49.2	94.8
		EODJ	0.625	0.8685	27.0	10	23.4	17	11.7	41.3	74.8
		HOCJ	0.625	0.8685	27.0	10	23.4	17	11.7	41.3	74.8
Inlet Nozzles	B6608-1	95443-1	0.016	0.1513	67.0	60	10.1	17	5.1	35.5	105.6
	B6608-2	95460-1	0.016	0.1513	67.0	60	10.1	17	5.1	35.5	105.6
	B6608-3	95712-1	0.016	0.1513	51.0	60	7.7	17	3.9	34.9	102.6
Inlet Nozzle Welds	1-717 B 1-717 D 1-717 F	EODJ	0.016	0.1513	27.0	10	4.1	17	2.0	34.2	48.3
		FOIJ	0.016	0.1513	41.0	10	6.2	17	3.1	34.6	50.8
		HOCJ	0.016	0.1513	27.0	10	4.1	17	2.0	34.2	48.3
		DBIJ	0.016	0.1513	27.0	10	4.1	17	2.0	34.2	48.3
		EOEJ	0.016	0.1513	20.0	10	3.0	17	1.5	34.1	47.2
		ICJJ	0.016	0.1513	41.0	10	6.2	17	3.1	34.6	50.8
		JACJ	0.016	0.1513	54.0	10	8.2	17	4.1	35.0	53.1
Outlet Nozzles	B6605-1	95415-1	0.011	0.1191	95.25	60	11.3	17	5.7	35.8	107.2
	B6605-2	95415-2	0.011	0.1191	95.25	60	11.3	17	5.7	35.8	107.2
	B6605-3	95444-1	0.011	0.1191	58.0	60	6.9	17	3.5	34.7	101.6
Outlet Nozzle Welds	1-717 A 1-717 C 1-717 E	ICJJ	0.011	0.1191	41.0	10	4.9	17	2.4	34.3	49.2
		IOBJ	0.011	0.1191	27.0	10	3.2	17	1.6	34.2	47.4
		JACJ	0.011	0.1191	54.0	10	6.4	17	3.2	34.6	51.0
		HOCJ	0.011	0.1191	27.0	10	3.2	17	1.6	34.2	47.4
		EODJ	0.011	0.1191	27.0	10	3.2	17	1.6	34.2	47.4
		FOIJ	0.011	0.1191	41.0	10	4.9	17	2.4	34.3	49.2

Notes:

a) FF = fluence factor = $f^{(0.28 - 0.10 \log (f))}$.
b) Initial RT_{NDT} value for the upper shell forging is a measured value. All other values are generic.
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 Fort Calhoun surveillance weld metal is the same weld heat as the BVPS-1 upper to intermediate shell girth weld (heat 305414). The Fort Calhoun surveillance weld data is non-credible (see Appendix A); therefore, the higher σ_Δ term of 28°F was utilized for BVPS-1 weld heat 305414.

Table 6-5 BVPS-1 Beltline Materials Projected USE Values at 50 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	B6607-1	---	0.14	3.475	94	32.5	63.5
Intermediate Shell Plate	B6607-2	---	0.14	3.475	83	32.5	56.0
Lower Shell Plate	B6903-1	---	0.21	3.475	83	37[b]	52.3
Lower Shell Plate	B7203-2	---	0.14	3.475	85	32.5	57.4
Intermediate to Lower Shell Girth Weld	11-714	90136	0.27	3.462	144	52	69.1
Intermediate Shell Longitudinal Weld	19-714 A&B	305424	0.28	0.675	112	28[c]	80.6
Lower Shell Longitudinal Weld	20-714 A&B	305414	0.34	0.680	>100	40[d]	60.0

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 results from BVPS-1 surveillance plate B6903-1 [Reference 4].

c) Based on results from BVPS-1 surveillance weld heat 305424 [Reference 4].

d) Since this material's Cu Wt. % value is greater than the highest Cu Wt. % chemistry line on Figure 2 of Regulatory Guide 1.99, Revision 2, the upper limit line on Figure 2 was utilized to determine the percent USE decrease.

Table 6-6 BVPS-1 Extended Beltline Materials Projected USE Values at 50 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 Forging	B6604	123V339VA1	0.12	0.390	101	19	81.8
Upper to Intermediate Shell Girth Weld	10-714	305414 (3951 & 3958)	0.337	0.390	97	37[c]	61.1
		AOFJ	0.03	0.390	111	15	94.4
		FOIJ	0.03	0.390	104	15	88.4
		EODJ	0.02	0.390	156	15	132.6
		HOCJ	0.02	0.390	160	15	136.0
Inlet Nozzles	B6608-1	95443-1	0.10	0.010	82.5	7.5	76.3
	B6608-2	95460-1	0.10	0.010	94	7.5	87.0
	B6608-3	95712-1	0.08	0.010	97	7.5	89.7
Inlet Nozzle Welds	1-717 B 1-717 D 1-717 F	EODJ	0.02	0.010	156	7.5	144.3
		FOIJ	0.03	0.010	104	7.5	96.2
		HOCJ	0.02	0.010	160	7.5	148.0
		DBIJ	0.02	0.010	123	7.5	113.8
		EOEJ	0.01	0.010	152	7.5	140.6
		ICJJ	0.03	0.010	123	7.5	113.8
		JACJ	0.04	0.010	116	7.5	107.3
Outlet Nozzles	B6605-1	95415-1	0.13	0.007	93	9.5	84.2
	B6605-2	95415-2	0.13	0.007	112.5	9.5	101.8
	B6605-3	95444-1	0.09	0.007	103	7.5	95.3
Outlet Nozzle Welds	1-717 A 1-717 C 1-717 E	ICJJ	0.03	0.007	123	7.5	113.8
		IOBJ	0.02	0.007	122	7.5	112.9
		JACJ	0.04	0.007	116	7.5	107.3
		HOCJ	0.02	0.007	160	7.5	148.0
		EODJ	0.02	0.007	156	7.5	144.3
		FOIJ	0.03	0.007	104	7.5	96.2

Notes:

a) Percent USE decreases are based on the closest Cu Wt. % chemistry line (rounding up) of Figure 2 of Regulatory Guide 1.99, Revision 2 unless the actual Cu Wt. % value of the evaluated material is exactly the same as that of one of the Cu Wt % lines on Figure 2.

b) The minimum fluence value (2×10^{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.

c) Since this material's Cu Wt. % value is greater than the highest Cu Wt. % chemistry line of Figure 2 of Regulatory Guide 1.99, Revision 2, the upper limit line on Figure 2 was utilized to determine the percent USE decrease.

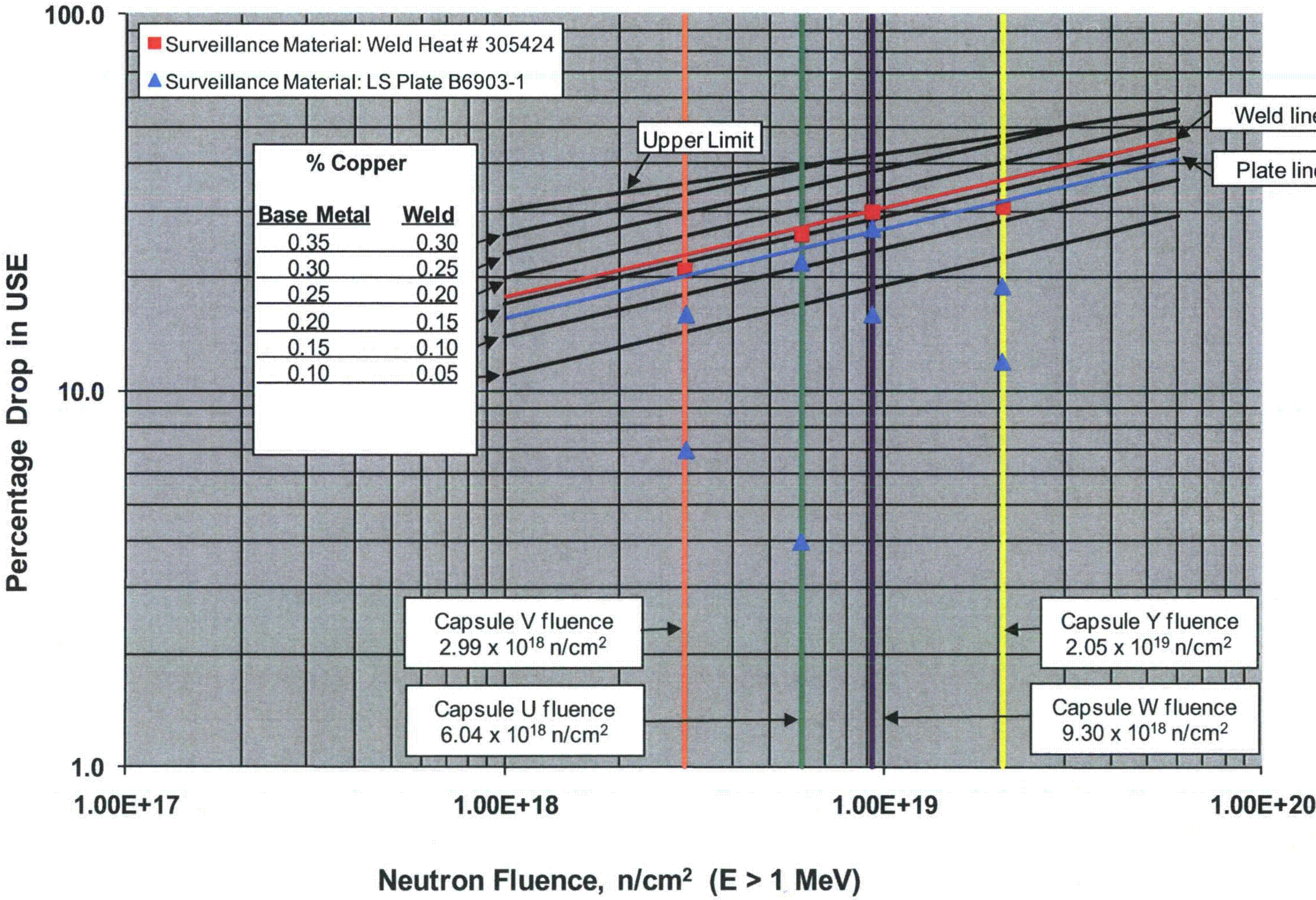

Figure 6-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in USE as a Function of Copper and Fluence for BVPS-1

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 11] as specified by CFR Part 50, Appendix G [Reference 12].

The BVPS-1 P-T limit curves for normal heatup and cooldown of the primary reactor coolant system were previously developed in WCAP-16799-NP, Revision 1 [Reference 5] through 30 EFPY. The existing 30 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-1 30 EFPY P-T limit curves were developed by calculating ART values utilizing the maximum clad/base metal fluence for the lower and intermediate shell plates as well as the lower to intermediate shell girth weld. The ART calculations for the intermediate and lower shell longitudinal welds utilized the peak clad/base metal interface fluence at the 45° azimuthal location. The limiting ART values used in the development of the 30 EFPY P-T limit curves correspond to the lower shell plate B6903-1 (Position 2.1 - using non-credible 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 lower shell plate B6903-1 (using non-credible surveillance data) continues to be the limiting material for the current BVPS-1 P-T limit curves. Additionally, the BVPS-1 updated vessel and surveillance capsule fluence values, as well as the revised sister-plant Position 2.1 CF values, do not reduce the existing 30 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-1.

8 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 [Reference 13] and is recommended for future capsules to be withdrawn from the BVPS-1 reactor vessel.

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

Capsule	Current (Original) Capsule Location	Lead Factor[a]	Withdrawal EFPY[b]	Fluence, f[a] [n/cm^2, E > 1.0 MeV]
V	165°	1.61	1.16	2.99×10^{18}
U	65°	1.06	3.59	6.04×10^{18}
W	245°	1.11	5.89	9.30×10^{18}
Y	295°	1.2	14.29	2.05×10^{19}
X[c]	285°	1.72	26.5[c]	5.01×10^{19}[c]
T[d]	65° (55°)	0.99	(d)	2.74×10^{19}[d]
S[e]	45°	0.64	(e)	1.78×10^{19}[e]
Z[f]	165° (305°)	1.24	36.6[f]	3.45×10^{19}[f]

Notes:

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

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

c) Capsule X is planned to be withdrawn at the EOC 22, which corresponds to 26.5 EFPY. This capsule will meet the requirements of ASTM E185-82 for the fifth capsule to be withdrawn for the 40-year EOL.

d) Capsule T was moved to the Capsule U location at the EOC 10. Accumulated fluence value through EOC 21. In order to achieve higher fluence data for this capsule, Capsule T should be relocated to the current Capsule Z location when Capsule Z is withdrawn from the vessel (See footnote (f)). This capsule will reach the projected 80-year EOL (corresponding to 68 EFPY) peak vessel fluence at approximately 56 EFPY after relocation to the Capsule V location; however, since the current regulations may change between now and then, it is recommended that the schedule for withdrawal of an 80-year license capsule be revisited at a later time.

e) Accumulated fluence value through EOC 21. In order to achieve higher fluence data for this capsule, Capsule S should be relocated to the Capsule X location when Capsule X is withdrawn from the vessel at 26.5 EFPY. This capsule will reach the projected 80-year EOL (corresponding to 68 EFPY) peak vessel fluence at approximately 58 EFPY after relocation to the Capsule X location; however, since the current regulations may change between now and then, it is recommended that the schedule for withdrawal of an 80-year license capsule be revisited at a later time.

f) Capsule Z was moved to the original Capsule V location at the EOC 10. Accumulated fluence value through EOC 21. Based on the current information, Capsule Z should be withdrawn after 36.6 EFPY, which corresponds to the peak vessel fluence at 60-year EOL (50 EFPY), 5.58×10^{19} n/cm^2 (E > 1.0 MeV).

9 CONCLUSION

- All of the beltline and extended beltline region materials in the BVPS-1 reactor vessel have EOLE RT_{PTS} values below the screening criteria values of 270°F for forgings/plates and 300°F for circumferential welds at EOLE (50 EFPY) with the exception of lower shell plate B6903-1. This plate has a 50 EFPY RT_{PTS} value of 277.0°F. Based on the fluence information provided in Section 5, the PTS screening criteria of 270°F is reached at a fluence value of 4.407×10^{19} n/cm^2 (E>1.0 MeV). This fluence value of 4.407×10^{19} n/cm^2 (E>1.0 MeV) equates to 39.6 EFPY for BVPS-1.

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

- The current P-T limit curves remain valid through 30 EFPY for BVPS-1.

- Four capsules have been withdrawn and tested from BVPS-1. Capsule X is scheduled to be withdrawn from BVPS-1 at EOC 22 and will satisfy the current requirements for the last capsule to be withdrawn for the BVPS-1 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-15571, Revision 1 "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," N.R. Jurcevich, April 2008.

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6. Combustion Engineering Report MISC-PENG-ER-022, Revision 00, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the Beaver Valley Unit 1 Reactor Pressure Vessel Plates, Forgings, Welds and Cladding," S.M. Schloss, et. al., October 1995.

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12. 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.

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APPENDIX A
BEAVER VALLEY UNIT 1 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-1 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-1 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-1 reactor vessel beltline region consists of the following materials:

1. Intermediate Shell Plates B6607-1 and B6607-2
2. Lower Shell Plates B6903-1 and B7203-2
3. Intermediate Shell Longitudinal Welds (Heat 305424)
4. Intermediate to Lower Shell Girth Weld (Heat 90136)
5. Lower Shell Longitudinal Welds (Heat 305414)

Per WCAP-8457 [Reference A-3], the BVPS-1 surveillance program was developed to the requirements of ASTM E185-73. Intermediate shell plate B6607-2 had the highest initial RT_{NDT} value; however, lower shell plate B6903-1 had the highest weight percent copper and the lowest initial USE values of all plates in the beltline region. At the time the surveillance program material was selected, it was believed that copper and phosphorus were the most important elements to radiation embrittlement. Hence, the lower shell plate B6903-1 was chosen as the most limiting material. Furthermore, an evaluation was performed at a later time to determine which plate between the intermediate shell plate B6607-2 and lower shell plate B6903-1 would be limiting during the reactor vessel's lifetime. Per WCAP-14543 [Reference A-4], intermediate shell plate B6607-2 was the limiting material during the time when the fluence values were less than 1.727×10^{19} n/cm^2 (E > 1.0 MeV). Therefore, since the fluence for the majority of the vessel's lifetime will be greater than 1.727×10^{19} n/cm^2 (E > 1.0 MeV), the most limiting surveillance plate material was properly selected for Beaver Valley Unit 1.

The initial RT_{NDT} of the weld metals contained in the beltline region were not known and are based on a generic value. In addition, the initial USE values for the weld metals were not available. However, the weld wire used in the intermediate shell longitudinal weld seams had one of the highest weight percent copper and the highest weight percent phosphorus. Hence, weld wire heat 305424 Linde 1092 (flux lot # 3889) was utilized in the surveillance program and is identical to the intermediate shell longitudinal welds.

Based on the discussion above, Criterion 1 is met for the BVPS-1 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-15571-NP, Revision 1 [Reference A-5].

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-1 surveillance materials unambiguously.

Hence, Criterion 2 is met for the Beaver Valley Unit 1 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-6].

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-7]. 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-1 surveillance plate and weld material. However, the other Beaver Valley Unit 1 beltline welds are contained in either the St. Lucie or the Fort Calhoun surveillance programs. These welds will be also be evaluated for credibility using the guidance for the appropriate case as explained in Reference A-7. The welds and their respective evaluation methods are described below:

1. Lower Shell Plate B6903-1 (Case 1) – This plate material will be evaluated using the NRC Case 1 guidelines as described above.

2. Heat 305424 (Case 1) – This weld heat pertains to the intermediate shell longitudinal welds in the BVPS-1 reactor vessel. NRC Case 1 per Reference A-7 is entitled "Surveillance data available from plant but no other source" and most closely represents the situation for BVPS-1 weld heat 305424.

3. Heat 90136 (Case 5) – This weld heat pertains to the intermediate to lower shell girth weld in the BVPS-1 reactor vessel. This weld heat is not contained in the BVPS-1 surveillance program; however, it is contained in the St. Lucie Unit 1 surveillance program. NRC Case 5 per Reference A-7 is entitled "Surveillance Data from Other Sources Only" and most closely represents the situation for BVPS-1 weld heat 90136.

4. Heat 305414 (Case 5) – This weld heat pertains to the lower shell longitudinal welds in the BVPS-1 reactor vessel. This weld heat is not contained in the BVPS-1 surveillance program; however, it is contained in the Fort Calhoun surveillance program. NRC Case 5 per Reference A-7 is entitled "Surveillance Data from Other Sources Only" and most closely represents the situation for BVPS-1 weld heat 305414.

Case 1: Lower Shell Plate B6903-1 and Weld Heat 305424

Following the NRC Case 1 guidelines, the BVPS-1 surveillance plate and weld metal (heat 305424) will be evaluated using the BVPS-1 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-1 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 1 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
Lower Shell Plate B6903-1 (Longitudinal)	V	0.299	0.669	128.49	86.01	0.448
	U	0.604	0.859	118.93	102.14	0.738
	W	0.930	0.980	148.52	145.50	0.960
	Y	2.05	1.196	142.18	169.98	1.429
Lower Shell Plate B6903-1 (Transverse)	V	0.299	0.669	137.81	92.25	0.448
	U	0.604	0.859	131.84	113.23	0.738
	W	0.930	0.980	179.99	176.33	0.960
	Y	2.05	1.196	166.93	199.58	1.429
				SUM:	1085.02	7.150
	$CF_{LS\ Plate\ B6903-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1085.02) \div (7.150) = 151.8°F$					
Beaver Valley Unit 1 Weld Metal (Heat 305424)	V	0.299	0.669	159.72	106.92	0.448
	U	0.604	0.859	166.32	142.84	0.738
	W	0.930	0.980	187.73	183.91	0.960
	Y	2.05	1.196	179.69	214.83	1.429
				SUM :	648.50	3.575
	$CF_{305424} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (648.50) \div (3.575) = 181.4°F$					

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 1 Surveillance Materials Only

Material	Capsule	CF ($Slope_{best-fit}$) (°F)	Capsule f ($x10^{19}$ n/cm², E > 1.0 MeV)	FF	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT} (°F)	<17°F (Base Metal) <28°F (Weld)
Lower Shell Plate B6903-1 (Longitudinal)	V	151.8	0.299	0.669	128.49	101.6	26.9	No
	U	151.8	0.604	0.859	118.93	130.3	11.4	Yes
	W	151.8	0.930	0.980	148.52	148.7	0.2	Yes
	Y	151.8	2.05	1.196	142.18	181.4	39.3	No
Lower Shell Plate B6903-1 (Transverse)	V	151.8	0.299	0.669	137.81	101.6	36.2	No
	U	151.8	0.604	0.859	131.84	130.3	1.5	Yes
	W	151.8	0.930	0.980	179.99	148.7	31.3	No
	Y	151.8	2.05	1.196	166.93	181.4	14.5	Yes
Beaver Valley Unit 1 Weld Metal (Heat 305424)	V	181.4	0.299	0.669	159.72	121.4	38.3	No
	U	181.4	0.604	0.859	166.32	155.8	10.5	Yes
	W	181.4	0.930	0.980	187.73	177.7	10.0	Yes
	Y	181.4	2.05	1.196	179.69	216.9	37.2	No

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 four of the eight surveillance data points fall within the +/- 1σ of 17°F scatter band for surveillance base metals; therefore, the lower shell plate B6903-1 data is deemed "non-credible" per the third criterion.

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 only two of the four surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance weld materials; therefore, the weld material (heat 305424) is deemed "non-credible" per the third criterion.

Note that although the lower shell plate B6903-1 and the weld material (heat 305424) did not meet Criterion 3, they may still be used in determining the upper-shelf energy decrease in accordance with Regulatory Guide 1.99, Revision 2, Position 2.2.

Case 5: Weld Heat 90136 (St. Lucie Unit 1 data)

Following the NRC Case 5 guidelines, the St. Lucie Unit 1 surveillance weld metal (heat 90136) will be evaluated for credibility. Weld heat 90136 pertains to the BVPS-1 intermediate to lower shell girth weld, and is not contained in the BVPS-1 surveillance program. No adjustments for irradiation temperature or chemistry are required since only the data scatter from a single source (St. Lucie Unit 1) is being considered here for credibility. This is performed below in Table A-3.

Table A-3 Calculation of Interim Chemistry Factor for Weld Heat 90136 Using St. Lucie Unit 1 Surveillance Data

Material	Capsule	Capsule f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Weld Metal Heat 90136 (St. Lucie Unit 1 data)	97°	0.5174	0.816	72.34	59.03	0.666
	104°	0.7885	0.933	67.4	62.91	0.871
	284°	1.243	1.061	68.0	72.12	1.125
				SUM :	194.06	2.662
	$CF_{90136} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (194.06) \div (2.662) = 72.9°F$					

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-4.

Table A-4 Best-Fit Evaluation for Weld Heat 90136 Using St. Lucie Unit 1 Data

Material	Capsule	CF ($Slope_{best\text{-}fit}$) (°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)	<28°F (Weld)
Weld Metal Heat 90136 (St. Lucie Unit 1 data)	97°	72.9	0.5174	0.816	72.34	59.5	12.9	Yes
	104°	72.9	0.7885	0.933	67.4	68.0	0.6	Yes
	284°	72.9	1.243	1.061	68.0	77.3	9.3	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 28°F for weld metal. Table A-4 indicates that all three surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance base metals; therefore, the St. Lucie Unit 1 weld metal heat 90136 data is deemed "credible" per the third criterion.

Therefore, the surveillance data from St. Lucie Unit 1 for weld heat 90136 may be applied to the BVPS-1 reactor vessel weld as credible data.

Case 5: Weld Heat 305414 (Fort Calhoun data)

Following the NRC Case 5 guidelines, the Fort Calhoun surveillance weld metal (heat 305414) will be evaluated for credibility. Weld heat 305414 pertains to the BVPS-1 lower shell longitudinal welds, and is not contained in the BVPS-1 surveillance program. No adjustments for irradiation temperature or chemistry are required since only the data scatter from a single source (Fort Calhoun) is being considered here for credibility. This is performed below in Table A-5.

Table A-5 Calculation of Interim Chemistry Factor for Weld Heat 305414 Using Fort Calhoun Surveillance Data

Material	Capsule	Capsule f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Weld Metal Heat 305414 (Fort Calhoun data)	W-225	0.488	0.800	210	167.98	0.640
	W-265	0.847	0.953	225	214.52	0.909
	W-275	1.54	1.119	219	245.15	1.253
				SUM :	627.66	2.802
	$CF_{305414} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (627.66) \div (2.802) = 224.0°F$					

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-6.

Table A-6 Best-Fit Evaluation for Weld Heat 305414 Using Fort Calhoun Data

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)	<28°F (Weld)
Weld Metal Heat 305414 (Fort Calhoun data)	W-225	224.0	0.488	0.800	210	179.2	30.8	No
	W-265	224.0	0.847	0.953	225	213.6	11.4	Yes
	W-275	224.0	1.54	1.119	219	250.8	31.8	No

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-6 indicates that only one of the three surveillance data points fall within the +/- 1σ of 28°F scatter band for surveillance base metals; therefore, the Fort Calhoun weld metal heat 305414 data is deemed "non-credible" per the third criterion.

Therefore, the surveillance data from Fort Calhoun for weld heat 305414 may be applied to the BVPS-1 reactor vessel weld as non-credible data.

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-1 capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the thermal shield. 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 such that the temperatures will not differ by more than 25°F.

Hence, Criterion 4 is met for the BVPS-1 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-1 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the BVPS-1 surveillance program.

CONCLUSION:

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2, Section B, the BVPS-1 surveillance weld and plate data, as well as the Fort Calhoun surveillance weld data for use at BVPS-1, are deemed non-credible. The St. Lucie surveillance weld data for use at BVPS-1 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-8457, Revision 0, "Duquesne Light Company Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program," J. A. Davidson et al., October 1974.

A-4 WCAP-14543, Revision 0, "Evaluation of Pressurized Thermal Shock for the Beaver Valley Unit 1 Reactor Vessel," P. A. Grendys, June 1996.

A-5 WCAP-15571-NP, Revision 1, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," N. R. Jurcevich, April 2008.

A-6 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-7 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.