Westinghouse Non-Proprietary Class 3

WCAP-15103
Revision 0

Evaluation of Pressurized Thermal Shock for V. C. Summer Unit 1

Westinghouse Energy Systems

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Evaluation of Pressurized Thermal Shock for V. C. Summer Unit 1

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the V. C. Summer Unit 1 reactor vessel beltline based upon the results of the Surveillance Capsule W evaluation. The conclusion of this report is that all the beltline materials in the V. C. Summer Unit 1 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY).

1 INTRODUCTION

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

The purpose of this report is to determine the RT_{PTS} values for the V. C. Summer Unit 1 reactor vessel using the results of the surveillance Capsule W evaluation. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the V. C. Summer Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-15101[5]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) recently amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule[1], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

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

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

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

The PTS Rule requirements consist of the following:

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

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

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

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

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

3 METHOD FOR CALCULATION OF RT_{PTS}

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

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

Where,

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

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

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

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

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

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

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

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

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

For welds:

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

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

σ_Δ not to exceed one half of ΔRT_{NDT}

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

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

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

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

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

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

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

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

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

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

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 V. C. Summer Unit 1 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat-affected zones and plates 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". Figure 1 identifies and indicates the location of all beltline region materials for the V. C. Summer Unit 1 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from V.C. Summer response to NRC Generic Letter 92-01, Revision 1, Supplement 1[6] and WCAP-12867[4]. The best estimate copper and nickel content is documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for V. C. Summer Unit 1 reactor vessel beltline material properties are also shown in Table 1.

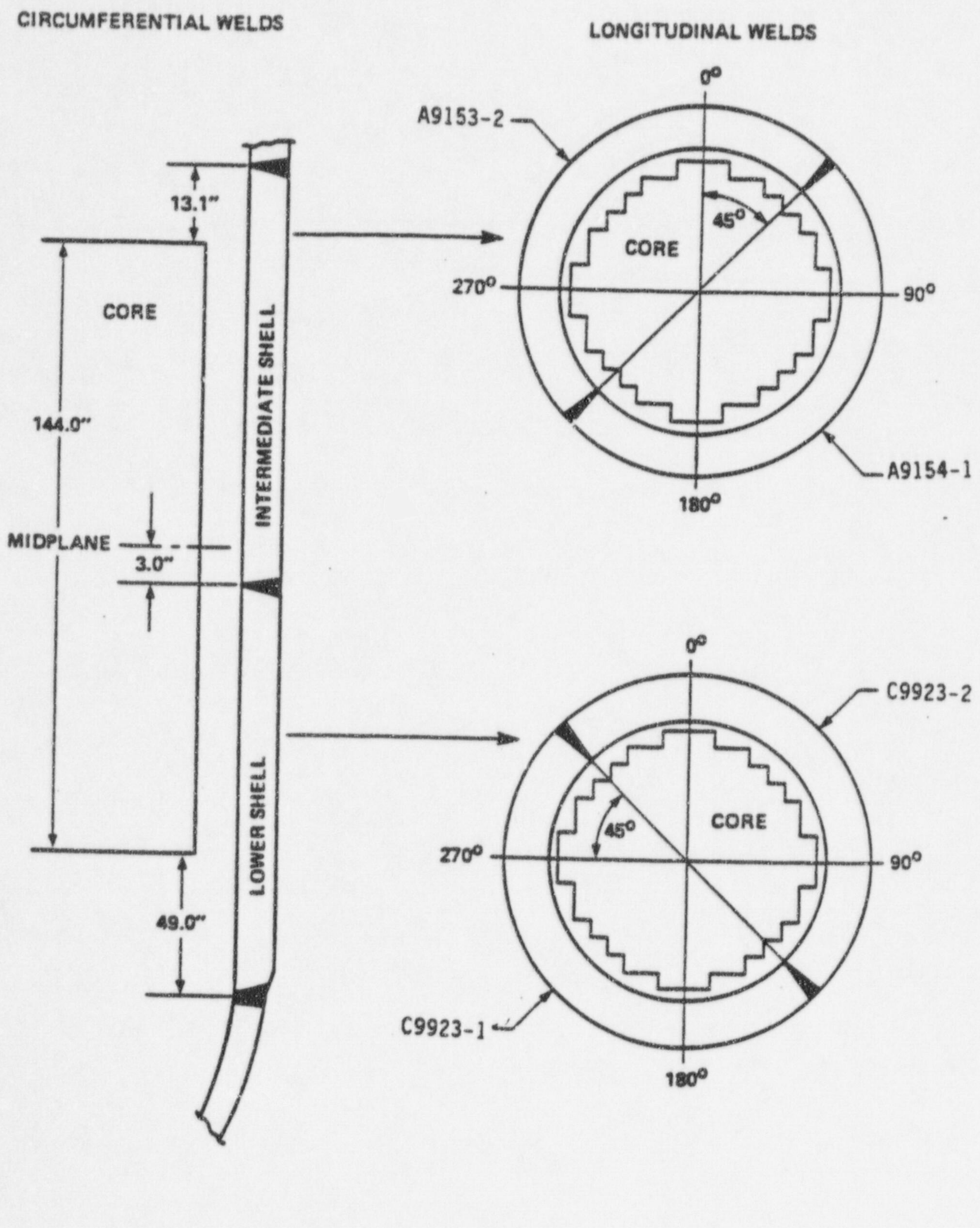

Figure 1: Identification and Location of Beltline Region Materials for the V. C. Summer Unit 1 Reactor Vessel

Table 1

V. C. Summer Unit 1 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 5297-1[b]	n/a	n/a	10°F[b]
Vessel Flange 5301-V-1	n/a	n/a	0°F[b]
Intermediate Shell Plate A9154-1	0.10	0.51	30°F
Intermediate Shell Plate A9153-2	0.09	0.45	-20°F
Lower Shell Plate C9923-1	0.08	0.41	10°F
Lower Shell Plate C9923-2	0.08	0.41	10°F
Intermediate Shell Longitudinal Welds, Seams BC & BD	0.05	0.91	-44°F
Lower Shell Longitudinal Welds, Seams BA & BB	0.05	0.91	-44°F
Intermediate to Lower Shell Plate Circumferential AB	0.05	0.91	-44°F
Surveillance Program Weld Metal	0.04	0.95	- - -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data per WCAP-12867[4].

(b) In the past the closure head flange was reported as Heat A9231 with a IRT_{NDT} of -20°F. Based on a review of Westinghouse files, the correct data is Heat # 5297-V1 with a IRT_{NDT} of 10°F. Also, the vessel flange was reported a IRT_{NDT} of 10°F., however, again , based on a review of Westinghouse files, the correct IRT_{NDT} of 0°F.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the V. C. Summer Unit 1 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the results of the Capsule W radiation analysis. See Section 6.0 of the Capsule W dosimetry analysis report, WCAP-15101[5] (48 EFPY values were obtained from interpolation of fluences between 32 EFPY and 54 EFPY).

TABLE 2
Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for V. C. Summer Unit 1 at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Intermediate Shell Plate A9154-1	0°	3.84×10^{19} n/cm^2	5.70×10^{19} n/cm^2
Intermediate Shell Plate A9154-2	0°	3.84×10^{19} n/cm^2	5.70×10^{19} n/cm^2
Lower Shell Plate C9923-1	0°	3.84×10^{19} n/cm^2	5.70×10^{19} n/cm^2
Lower Shell Plate C9923-2	0°	3.84×10^{19} n/cm^2	5.70×10^{19} n/cm^2
Intermediate Shell Longitudinal Weld Seam BC& BD	45°	1.43×10^{19} n/cm^2	2.15×10^{19} n/cm^2
Lower Shell Longitudinal Weld Seams BA &BB	45°	1.43×10^{19} n/cm^2	2.15×10^{19} n/cm^2
Intermediate to Lower Shell Plate Circumferential	0°	3.84×10^{19} n/cm^2	5.70×10^{19} n/cm^2

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

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the V. C. Summer Unit 1 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For V. C. Summer Unit 1, the related surveillance program results have been included in this PTS evaluation. (See Reference 5 for the credibility evaluation of the V.C. Summer Unit 1 surveillance data.)

As presented in Table 3, chemistry factor values for V. C. Summer Unit 1 based on average copper and nickel weight percent were calculated using Tables 1 and 2 from 10 CFR 50.61[1]. Additionally, chemistry factor values based on credible surveillance capsule data are calculated in Table 4. Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

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

Material	**Ni, wt %**	**Chemistry Factor, °F**
Intermediate Shell Plate A9154-1 Given Cu wt% = 0.10	0.51	65.0°F
Intermediate Shell Plate A9153-2 Given Cu wt% = 0.09	0.45	58.0°F
Intermediate Shell Plate C9923-1 Given Cu wt% = 0.08	0.41	51.0°F
Lower Shell Plate C9923-2 Given Cu wt% = 0.08	0.41	51.0°F
Intermediate Shell Longitudinal Welds, BC&BD Given Cu wt% = 0.05	0.91	68.0°F
Lower Shell Longitudinal Welds BA & BB Given Cu wt% = 0.05	0.91	68.0°F
Inter. to Lower Shell Circ. Weld Seam AB Given Cu wt% = 0.05	0.91	68.0°F
Surveillance Program Weld Metal Given Cu wt% = 0.04	0.95	54.0°F

TABLE 4

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

Material	Capsule	Capsule f(a)	FF(b)	ΔRT_{NDT} (c)	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell	U	0.654	0.881	36.0	31.7	0.776
Plate A9154-1	V	1.538	1.119	52.6	58.9	1.252
	X	2.543	1.250	37.7	47.1	1.563
(Longitudinal)	W	4.664	1.388	65.7	91.2	1.927
Intermediate Shell	U	0.654	0.881	14.5	12.8	0.776
Plate A9154-1	V	1.538	1.119	32.4	36.3	1.252
	X	2.543	1.250	26.0	32.5	1.563
(Transverse)	W	4.664	1.388	57.8	80.2	1.927
				SUM	390.7	11.036
	$CF_{A9154\text{-}1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (390.7) \div (11.036) = 35.4°F$					
Surveillance Weld	U	0.654	0.881	28.0(d)	24.7	0.776
Metal	V	1.538	1.119	58.6(d)	65.6	1.252
	X	2.543	1.250	28.3(d)	35.4	1.563
	W	4.664	1.388	54.4(d)	75.5	1.927
				SUM	201.2	5.518
	$CF_{Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (201.2) \div (5.518) = 36.5°F$					

Notes:

(a) f = Measured fluence from capsule W dosimetry analysis results[5], (x 10^{19} n/cm^2, E > 1.0 MeV).

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values.

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 1.26. ($CF_{VW} \div CF_{SW}$ = 68 °F ÷ 54°F = 1.26)

TABLE 5

RT_{PTS} Calculation for V. C. Summer Unit 1 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence (n/cm^2, E>1.0MeV)	CF (°F)	FF	$RT_{NDT(U)}$ [(a)]	ΔRT_{PTS} [(c)]	Margin	RT_{PTS} [(b)]
Intermediate Shell Plate A9154-1	3.84	65	1.35	30	88	34	152
Using Surveillance Capsule Data	3.84	35.4	1.35	30	48	17	95
Intermediate Shell Plate A9153-2	3.84	58	1.35	-20	78	34	92
Lower Shell Plate C9923-1	3.84	51	1.35	10	69	34	113
Lower Shell Plate C9923-2	3.84	51	1.35	10	69	34	113
Intermediate Shell & Lower Long. Weld Materials BC, BD and BA, BB (45° Azimuth)	1.43	68	1.10	-44	75	56	87
Using Surveillance Capsule Data	1.43	36.5	1.10	-44	40	28	24
Intermediate to Lower Shell Circumferential Weld Seam A5	3.84	68	1.35	-44	92	56	104
Using Surveillance Capsule Data	3.84	36.5	1.35	-44	49	28	33

Notes:

(a) Initial RT_{NDT} values are measured values.

(b) RT_{PTS} = Initial $RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) $\Delta RT_{PTS} = CF * FF$

TABLE 6

RT_{PTS} Calculation for V. C. Summer Unit 1 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence (n/cm^2, E>1.0MeV)	CF (°F)	FF	$RT_{NDT(U)}$ [a]	ΔRT_{PTS} [c]	Margin	RT_{PTS} [b]
Intermediate Shell Plate A9154-1	5.70	65	1.43	30	93	34	157
Using Surveillance Capsule Data	5.70	35.4	1.43	30	51	17	98
Intermediate Shell Plate A9153-2	5.70	58	1.43	-20	83	34	97
Lower Shell Plate C9923-1	5.70	51	1.43	10	73	34	117
Lower Shell Plate C9923-2	5.70	51	1.43	10	73	34	117
Intermediate Shell & Lower Long. Weld Materials BC, BD and BA, BB (45° Azimuth)	2.15	68	1.21	-44	82	56	94
Using Surveillance Capsule Data	2.15	36.5	1.21	-44	44	28	28
Intermediate to Lower Shell Circumferential Weld Seam A5	5.70	68	1.43	-44	97	56	109
Using Surveillance Capsule Data	5.70	36.5	1.43	-44	52	28	36

Notes:

(a) Initial RT_{NDT} values are measured values.

(b) RT_{PTS} = Initial $RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) $\Delta RT_{PTS} = CF * FF$

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the V. C. Summer Unit 1 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds.

8 REFERENCES

1 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.

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

3 WCAP-9234, "South Carolina Electric and Gas Company Virgil C. Summer Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, January 1978.

4 WCAP-12867, "Analysis of Capsule X from the South Carolina Electric and Gas Company V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program", J. M. Chicots, et. al., March, 1991.

5 WCAP-15101, "Analysis of Capsule W from the South Carolina Electric and Gas Company V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., September 1998.

6 South Carolina Electric and Gas Letter, G.J. Taylor to U.S. NRC, Dated 11/7/95, Subject, "Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Response to G.L. 92-01, Revision 1, Supplement 1".