Westinghouse Non-Proprietary Class 3

WCAP-15090
Revision 1

Evaluation of Pressurized Thermal Shock for Wolf Creek

Westinghouse Energy Systems

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PREFACE

This report has been technically reviewed and verified by: T.J.Laubham

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EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Wolf Creek reactor vessel beltline based upon the results of the Surveillance Capsule V evaluation. The surveillance program test results are deemed credible per the 10 CFR Part 50.61 criteria (see WCAP-15078, Appendix D). Hence, the RT_{PTS} values presented in this report were calculated per the methodology given in 10CFR50.61 utilizing surveillance data where applicable. Based upon these conservative assumptions, all of the beltline materials in the Wolf Creek reactor vessel have RT_{PTS} values below the screening criteria values of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (35 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 significant pressure or followed by a significant re-pressurization in the reactor vessel. A PTS concern arises if one of these transient acts on the beltline region of the 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 Wolf Creek reactor vessel using the results of the surveillance Capsule V evaluation. Section 2.0 discusses the PTS Rule and its requirements. Sections 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Wolf Creek reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0. 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 CRITERIA

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 revised 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.

The 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 currently described in Regulatory Guide 1.99, Revision 2[2].
2. The rule was restructured to improve clarity, with the requirements section giving only the requirement for the value of the reference temperature for End Of Life (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, forging and axial weld materials, and

  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 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.

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

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 calculation procedures. M is evaluated from Equation 2.

$$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2} \quad (2)$$

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 ΔRT_{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}.

In equation 4, calculate a bounding value of RT_{PTS} using Equation 3 with EOL 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 ΔRT_{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 Wolf Creek 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 Wolf Creek reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from WCAP-10015, Rev. 1[3], CE Report NPSD-1039, Rev. 2[4] and letter WCAP-15078[5]. The best estimate copper and nickel content is documented in Table 1 herein. These average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for the Wolf Creek reactor vessel beltline materials are also shown in Table 1.

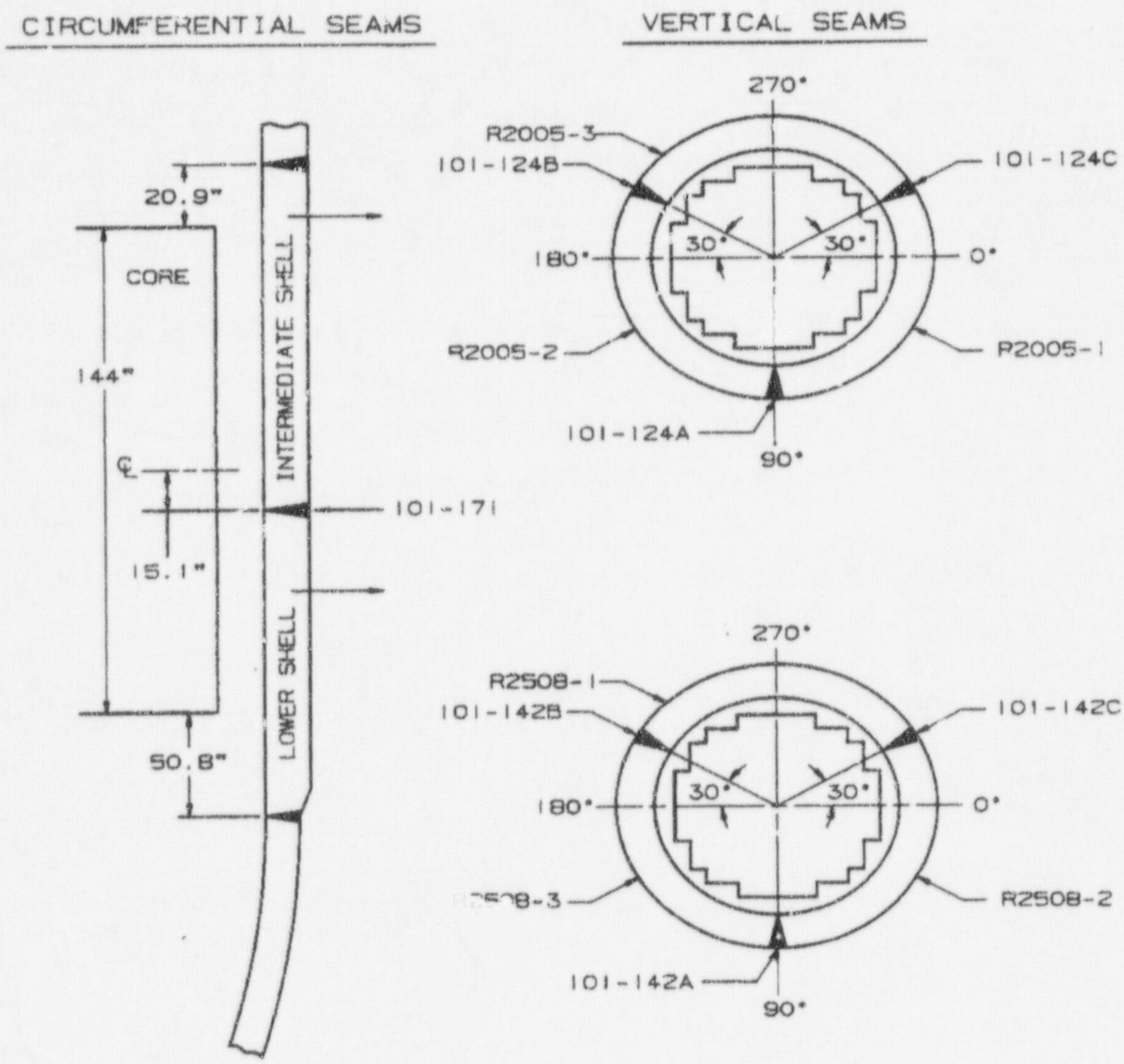

Figure 1
Identification and Location of Beltline Region Materials for Wolf Creek Reactor Vessel

Table 1
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Materials Description	Cu wt%	Ni wt%	Initial RT $_{NDT}$ (a) °F
Closure Head Flange R2504 - 1		0.66	20
Vessel Flange R2501 - 1		0.70	20
Intermediate Shell Plate R2005 - 1	0.04	0.66	-20
Intermediate Shell Plate R2005 - 2	0.04	0.64	-20
Intermediate Shell Plate R2005 - 3	0.05	0.63	-20
Lower Shell Plate R2508 - 1	0.09	0.67	0
Lower Shell Plate R2508 - 2	0.06	0.64	10
Lower Shell Plate R2508 - 3	0.09	0.58	40
Intermediate and Lower Shell Longitudinal Weld Seams (b)	0.04	0.08	-50
Intermediate to Lower Shell Weld Seam (b)	0.04	0.08	-50
Surveillance Weld Metal (b)	0.07	0.10	

Notes:

(a) Based on measured data.

(b) All vessel beltline weld seams were fabricate with weld wire heat number 90146. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate shell longitudinal weld seams 101-124A, B & C and lower shell longitudinal weld seams 101-142A , B & C were fabricated with Flux Type 0091 Lot Number 0842. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot Number 1061. Per Regulatory Guide 1.99, Revision 2, " weight-percent copper" and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld". The surveillance weld material was made with the same weld wire heat as all of the vessel beltline weld seams and is therefore representative of all of the beltline weld seams.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E>1.0 MeV) values at the inner surface of the Wolf Creek reactor vessel are shown in Table 2. These values were projected using the results of the Capsule V radiation analysis. See Section 6.0 of the Capsule V analysis report, WCAP -15078[5].

Table 2
Fluence (E> 1.0MeV) on the Pressure Vessel Clad/Base Interface for Wolf Creek @ 35 EFPY (EOL)

Material	Location(a)	Fluence (n/cm^2)
Intermediate Shell Plate R2005 - 1	30°	2.18E+19
Intermediate Shell Plate R2005 - 2	30°	2.18E+19
Intermediate Shell Plate R2005 - 3	30°	2.18E+19
Lower Shell Plate R2508 - 1	30°	2.18E+19
Lower Shell Plate R2508 - 2	30°	2.18E+19
Lower Shell Plate R2508 - 3	30°	2.18E+19
Intermediate Shell LongitudinalWeld Seam 101-124A	0°	1.15E+19
Intermediate Shell LongitudinalWeld Seams 101--142B&101-142C	30°	2.18E+19
Intermediate to Lower Shell Circ. Weld Seam 101-171	30°	2.18E+19
Lower Shell Longitudinal Weld Seams 101-142A	0°	1.15E+19
Lower Shell Longitudinal Weld Seams 101-142B&101-142C	30°	2.18E+19

(a) These locations are shown graphically in Figure 1.

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 regions materials of the Wolf Creek reactor vessel for fluence values at EOL (35 EFPY).

Each plant shall assess the RT $_{PTS}$ values based on plant specific surveillance capsule data. For Wolf Creek, surveillance data is only available from the Wolf Creek surveillance program and this data was included in the PTS Evaluation.

As presented in Table 3, chemistry factor values for Wolf Creek 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 data are calculated in Table 4. Table 5 contains the RT $_{PTS}$ calculations for all beltline region materials at EOL (35 EFPY).

Table 3
Calculations of Chemistry Factors using the Wolf Creek Surveillance Capsule Data

Material	Composition		Chemistry Factor, ^{0}F
	Ni, wt%	Cu, wt%	Note (a)
Intermediate Shell Plate R2005 - 1	0.66	0.04	26
Intermediate Shell Plate R2005 - 2	0.64	0.04	26
Intermediate Shell Plate R2005 - 3	0.63	0.05	31
Lower Shell Plate R2508 - 1	0.67	0.09	58
Lower Shell Plate R2508 - 2	0.64	0.06	37
Lower Shell Plate R2508 - 3	0.58	0.09	58
Beltline Region Weld Metal	0.08	0.04	31.6
Surveillance Weld Metal	0.10	0.07	43.5

(a) Chemistry factors were determined per the Chemistry factor tables in 10CFR50.61.

Table 4
Calculation of Chemistry Factors using Surveillance Capsule Data per 10CFR50.61

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Plate R2508-3 (Longitudinal)	U	0.3429	0.705	36.46	25.7	0.5
	Y	1.308	1.075	16.03	17.23	1.16
	V	2.528	1.249	52.03	64.99	1.56
Lower Shell Plate A9154-1 (Transverse)	U	0.3429	0.705	23.79	16.77	0.5
	Y	1.308	1.075	35.39	38.04	1.16
	V	2.528	1.249	54.53	68.11	1.56
				SUM:	230.84	6.44
	$CF_{R2508\text{-}3} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (230.84) \div (6.44) = 35.8°F$					
Surveillance Weld Material	U	0.3429	0.705	19.75	13.92	0.5
	Y	1.308	1.075	32.74	35.2	1.16
	V	2.528	1.249	33.64	42.02	1.56
				SUM:	91.14	3.22
	$CF_{S/P\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (91.14) \div (3.22) = 28.3°F$					

Notes:

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

(b) FF = fluence factor = $f^{(0.28 - .01 * \log f)}$

(c) ΔRT_{NDT} values for lower shell plate R2508-3 are the measured 30 ft-lb shift values given in the capsule V analysis report (WCAP-15078).

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 0.726 (CF_{vw} / CF_{sw} = 31.6 / 43.5 = 0.726)

Table 5

RT_{PTS} Calculations for Wolf Creek Beltline Region Materials @ 35 EFPY

Material	Fluence (n/cm2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Plate R2005-1	2.18E+19	1.21	26.0	31.5	31.5	-20	43
Intermediate Shell Plate R2005-2	2.18E+19	1.21	26.0	31.5	31.5	-20	43
Intermediate Shell Plate R2005-3	2.18E+19	1.21	31.0	37.5	34.0	-20	52
Lower Shell Plate R2508-1	2.18E+19	1.21	58.0	70.2	34.0	0	**104**
Lower Shell Plate R2508-2	2.18E+19	1.21	37.0	44.8	34.0	10	89
Lower Shell Plate R2508-3	2.18E+19	1.21	58.0	70.2	34.0	40	143
Using S/C Data	2.18E+19	1.21	35.8	43.3	17	40	100
Inter. and Lower Shell Long. Weld Seams 101-124A & 101-142A (90° Azimuth)	1.15E+19	1.04	31.6	32.9	32.9	-50	16
Using S/C Data	1.15E+19	1.04	28.3	29.4	28.0	-50	7
Inter. and Lower Shell Long. Weld Seams 101-124B/C & 101-142B/C (210° & 330° Azimuth)	2.18E+19	1.21	31.6	38.2	38.2	-50	26
Using S/C Data	2.18E+19	1.21	28.3	34.2	28.0	-50	12
Intermediate to Lower Shell Circumferential Weld Seam 101-171	2.18E+19	1.21	31.6	38.2	38.2	-50	26
- Using S/C Data	2.18E+19	1.21	28.3	34.2	28.0	-50	12

Notes:

(a) Initial RT NDT values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \text{Margin} + \Delta RT_{PTS}$

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

7 CONCLUSIONS

All of the beltline materials in the Wolf Creek reactor vessel are well below the screening criteria values of 270°F and 300°F at 35 EFPY.

8 REFERENCES

1. 10 CFR Part 50.61, "Fracture Toughness Toughness Requirements for Protection Against Pressurized Thermal Shock Events", Federal register, vol. 60, No.243, December 19, 1995.

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

3. WCAP-10015, Revision 1, "Kansas Gas and Electric Company Wolf Creek Generation Station Unit 1 Reactor Vessel Radiation Surveillance Program", L.R.Singer, June 1982.

4. CE Report NPSD-1039, Revision 2,"Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", ABB Combustion Engineering Nuclear Operations, June 1998

5. WCAP-15078," Analysis of Capsule V from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program", Ed Terek, et.al., August 1998.