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WCAP-13932

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR THE VOGTLE UNIT 1 REACTOR VESSEL

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February 1994

Work Performed Under Shop Order GTXP-108

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for the Georgia Power Company

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PREFACE

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1.0 INTRODUCTION

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a Loss-Of-Coolant-Accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall at relatively high pressurization;

- significant degradation of vessel material toughness caused by radiation embrittlement; and

- the presence of a critical-size defect in the vessel wall.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the reference nil-ductility transition temperature, termed RT_{PTS}[1]. RT_{PTS} screening values were set for beltline axial welds, forgings or plates and for beltline circumferential weld seams for operation to end of plant license. The screening criteria were determined using conservative fracture mechanics analysis techniques. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through end of license. The NRC recently amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991[2]. This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2[3].

The purpose of this report is to determine the RT_{PTS} values for the Vogtle Unit 1 reactor vessel to address the revised PTS Rule. Section 2 discusses the Rule and its requirements. Section 3 provides the methodology for calculating RT_{PTS}. Section 4 provides the reactor vessel beltline region material properties for the Vogtle Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5. The results of the RT_{PTS} calculations are presented in Section 6. The conclusion that all PTS screening criteria are satisfied at 48 EFPY and references for the PTS evaluation follow in Sections 7 and 8, respectively.

2.0 PRESSURIZED THERMAL SHOCK

The PTS Rule requires that the PTS submittal be updated whenever there are changes in core loadings, surveillance measurements or other information that indicates a significant change in projected RT_{PTS} values. The rule outlines regulations to address the potential for PTS events on pressurized water reactor vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (NRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The 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 result in the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The rule establishes the following requirements for all domestic, operating PWRs:

* All plants must submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for time of submittal, the expiration date of the operating license, and the projected expiration date if a change in the operating license or renewal has been requested. This assessment must be submitted within six months after the effective date of this Rule if the value of RT_{PTS} for any material is projected to exceed the screening criteria. Otherwise, it must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance capsule report, or within 5 years from the effective date of June 14, 1991, whichever comes first. These values must be calculated based on the methodology specified in the PTS Rule. The submittal must include the following:

    1) the bases for the projection (including any assumptions regarding core loading patterns), and

    2) copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to the NRC, justification must be provided.)

* The RT_{PTS} (measure of fracture resistance) screening criteria for the reactor vessel beltline region is:

  270 °F for plates, forgings, axial welds; and

  300 °F for circumferential weld materials.

* The equations that must be used to calculate the RT_{PTS} values for each weld, plate or forging in the reactor vessel beltline are specified (see Section 3).

* All values of RT_{PTS} must be verified to be bounding values for the specific reactor vessel. In doing this, each plant should consider plant-specific information that could affect the level of embrittlement.

* Plant-specific PTS safety analyses are required before a plant is within 3 years of reaching the screening criteria, including analyses of alternatives to minimize the PTS concern.

* NRC approval for operation beyond the screening criteria is required.

3.0 METHOD FOR CALCULATION OF RT_{PTS}

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of RT_{PTS} at a given time. For the purpose of comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region as follows:

RT_{PTS} = $I + M + \Delta RT_{PTS}$, where $\Delta RT_{PTS} = (CF) * FF$

I = Initial reference temperature (RT_{NDT}) in °F of the unirradiated material

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures in °F.
M = 66 °F for welds and 48 °F for base metal if generic values of I are used.
M = 56 °F for welds and 34 °F for base metal if measured values of I are used.

FF = fluence factor = $f^{(0.28 - 0.10 \log f)}$, where
f = Neutron fluence (E>1.0 MeV at the clad/base metal interface), divided by 10^{19} n/cm^2

CF = Chemistry factor in °F from the tables[2] for welds and base metals (plates and forgings). If plant-specific surveillance data has been deemed credible per Reg. Guide 1.99, Rev. 2 and is significant, it may be considered in the calculation of the chemistry factor.

4.0 VERIFICATION OF PLANT-SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Vogtle Unit 1 vessel was performed. The beltline region is defined by the PTS Rule[2] to be "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 irradiation 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 Vogtle Unit 1 reactor vessel.

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 surveillance capsule testing program[5]. The average copper and nickel values were calculated for each of the beltline region materials using all of the available material chemistry information as shown in Tables 1 through 3. A summary of the pertinent chemical and mechanical properties of the beltline region forgings and weld materials of the Vogtle Unit 1 reactor vessel are given in Table 4. All of the measured values of initial RT_{NDT} (I) are also presented in Table 4.

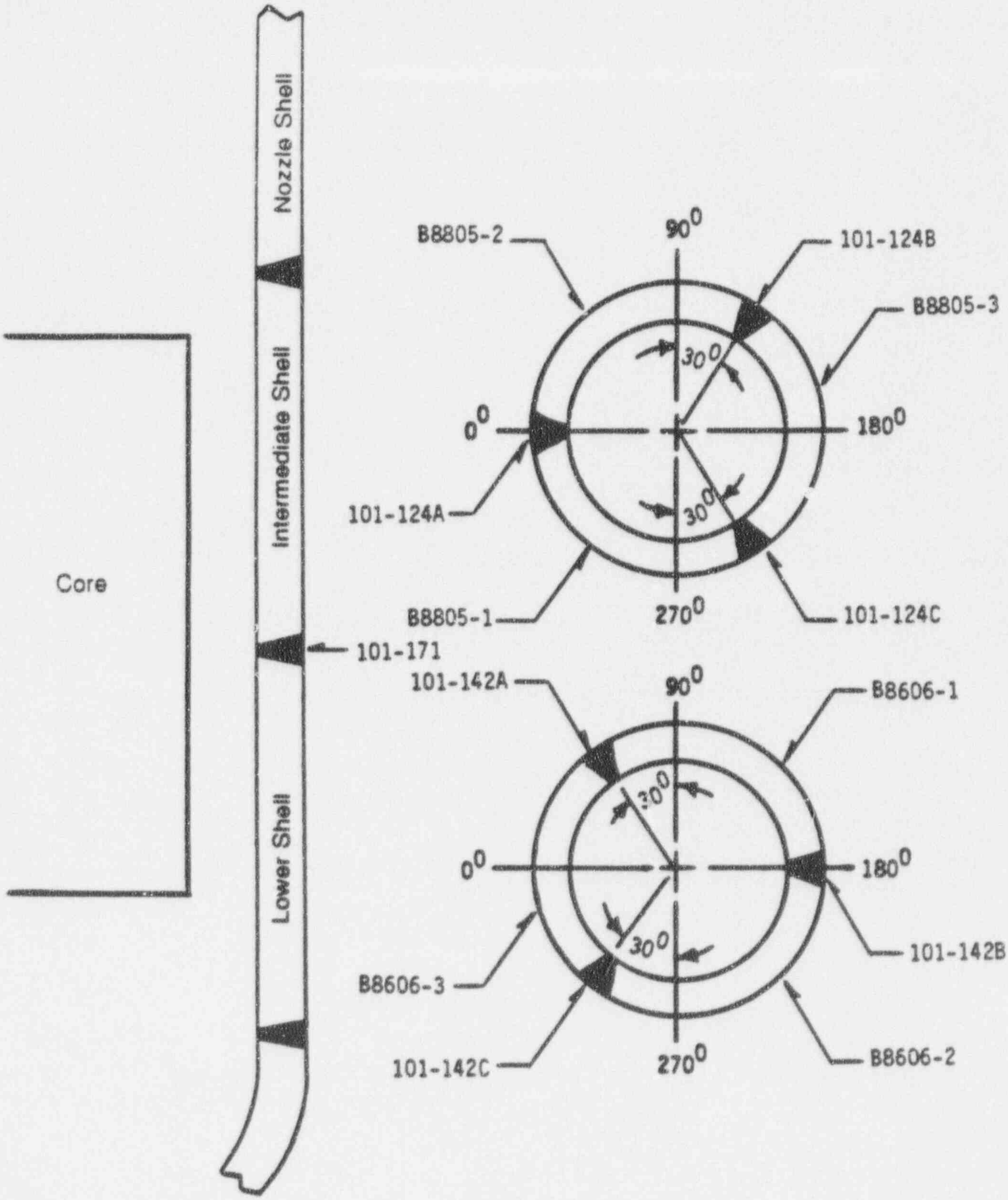

Figure 1. Identification and Location of Beltline Region Materials for the Vogtle Unit 1 Reactor Vessel[7]

Table 1

Calculation of Average Cu and Ni Weight % for Intermediate Shell Plates Using All Previous Vogtle Unit 1 Chemistry Test Results

Reference	Inter. Shell Plate B8805-1		Inter. Shell Plate B8805-2		Inter. Shell Plate B8805-3	
	Cu (wt%)	Ni (wt%)	Cu (wt%)	Ni (wt%)	Cu (wt%)	Ni (wt%)
Material Cert. Report[8]	0.09	0.60				
Material Cert. Report[8]	0.08	0.60				
Chemical Analysis[9]	0.08	0.59				
Surveillance Program[5]	0.08(1)	0.59(1)				
Material Cert. Report[10]			0.09	0.64		
Material Cert. Report[10]			0.08	0.60		
Chemical Analysis[11]			0.08	0.59		
Surveillance Program[5]			0.08(2)	0.59(2)		
Material Cert. Report[12]					0.07	0.60
Material Cert. Report[12]					0.07	0.61
Chemical Analysis[13]					0.06	0.60
Surveillance Program[5]					0.058	0.61
Surveillance Program[5]					0.06(3)	0.60(3)
Capsule U Report[6]					0.053	0.586
Capsule U Report[6]					0.06(3)	0.60(3)
Capsule U Report[6]					0.058(4)	0.61(4)
Capsule Y Report[4]					0.061	0.584
Average	0.083	0.597	0.083	0.61	0.062	0.598

(1) Not used in average calculation since same values as those from Reference 9; reported only for completeness.

(2) Not used in average calculation since same values as those from Reference 11; reported only for completeness.

(3) Not used in average calculation since same values as those from Reference 13; reported only for completeness.

(4) Not used in average calculation since same values as those from Reference 5; reported only for completeness.

Table 2

Calculation of Average Cu and Ni Weight % for Lower Shell Plates Using All Previous Vogtle Unit 1 Chemistry Test Results

Reference	Lower Shell Plate B8606-1		Lower Shell Plate B8606-2		Lower Shell Plate B8606-3	
	Cu (wt%)	Ni (wt%)	Cu (wt%)	Ni (wt%)	Cu (wt%)	Ni (wt%)
Material Cert. Report[14]	0.06	0.61				
Material Cert. Report[14]	0.05	0.58				
Chemical Analysis[15]	0.05	0.59				
Surveillance Program[5]	0.05(1)	0.59(1)				
Material Cert. Report[16]			0.07	0.64		
Material Cert. Report[16]			0.05	0.58		
Chemical Analysis[17]			0.05	0.58		
Surveillance Program[5]			0.05(2)	0.58(2)		
Material Cert. Report[18]					0.07	0.60
Material Cert. Report[18]					0.07	0.63
Chemical Analysis[19]					0.06	0.64
Surveillance Program[5]					0.06(3)	0.64(3)
Average	0.053	0.593	0.057	0.60	0.067	0.623

(1) Not used in average calculation since same values as those from Reference 15; reported only for completeness.

(2) Not used in average calculation since same values as those from Reference 17; reported only for completeness.

(3) Not used in average calculation since same values as those from Reference 19; reported only for completeness.

Table 3

Calculation of Average Cu and Ni Weight % for Weld Metal

Using All Previous Vogtle Unit 1 Chemistry Test Results

Reference	Weld Metal[1]	
	Cu (wt%)	Ni (wt%)
Surveillance Program[5]	0.03	- -
Surveillance Program[5]	0.04	0.10
Surveillance Program[5]	0.037	0.10
Capsule U Report[6]	0.035	0.091
Capsule Y Report[4]	0.048	0.101
Capsule Y Report[4]	0.04	0.117
Capsule Y Report[4]	0.041	0.105
Material Qualification[20]	0.04[2]	0.10[2]
Material Qualification[21]	0.03[2]	- -[2]
Average	0.039	0.102

(1) The core region (beltline) welds are considered to include the intermediate and lower shell plate longitudinal seams and the joining intermediate to lower shell girth seam. All core region (beltline) welds were fabricated using Weld Wire Heat No. 83653, Linde 0091 Flux, Lot No. 3536.

(2) Not used in average calculation since same values as those from Reference 5; reported only for completeness.

Table 4
Vogtle Unit 1 Reactor Vessel Beltline Region Material Properties

Material Description	Cu (%) [a]	Ni (%) [a]	I (°F)[6] [b]
Intermediate Shell Plate B8805-1	0.083	0.597	0
Intermediate Shell Plate B8805-2	0.083	0.610	20
Intermediate Shell Plate B8805-3	0.062	0.598	30
Lower Shell Plate B8806-1	0.053	0.593	20
Lower Shell Plate B8806-2	0.057	0.600	20
Lower Shell Plate B8806-3	0.067	0.623	10
Weld Metal [c]	0.039	0.102	-80

(a) Average values of copper and nickel as indicated in Tables 1 through 3 on the preceding pages.

(b) Initial RT_{NDT} values were estimated per U.S. NRC Standard Review Plan [22]. The initial RT_{NDT} values for the plates and welds are measured values.

(c) The core region (beltline) welds are considered to include the intermediate and lower shell plate longitudinal seams and the joining intermediate to lower shell girth seam. All core region (beltline) welds were fabricated using Weld Wire Heat No. 83653, Linde 0091 Flux, Lot No. 3536.

5.0 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E>1.0 MeV) at the inner surface of the Vogtle Unit 1 reactor vessel is shown in Table 5. These values were projected using the results of the Capsule Y radiation surveillance program[4]. The limiting RT_{PTS} calculations were performed using the peak fluence value, which occurs at the 25° azimuth in the Vogtle Unit 1 reactor vessel.

Table 5

Neutron Exposure Projections[1] at Key Locations on the Vogtle Unit 1

Pressure Vessel Clad/Base Metal Interface

EFPY	0°	15°	25°	30°	35°	45°
4.64	0.1802	0.2678	0.3155	0.2000	0.2622	0.2942
32	1.243	1.847	2.176	1.379	1.809	2.029
48[2]	1.863	2.769	3.262	2.068	2.711	3.041

(1) Fluence in 10^{19} n/cm^2 (E>1.0 MeV)

(2) Fluence values for 48 EFPY were calculated using the following equation:

Fluence = Present fluence + (48 - Present EFPY) * Flux * 3.1536E+7 sec/EFPY

where,

Present fluence = fluence at 4.64 EFPY
Present EFPY = 4.64 EFPY
Flux = Present flux rate (4.64 EFPY)

6.0 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 Vogtle Unit 1 reactor vessel for fluence values at the present time (4.64 EFPY per Capsule Y analysis), end of license (32 EFPY) and 48 EFPY.

The PTS Rule requires that each plant assess the RT_{PTS} values based on plant specific surveillance capsule data whenever:

- Plant specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99, Revision 2, and

- RT_{PTS} values change significantly. (Changes to RT_{PTS} values are considered significant if the value determined with RT_{PTS} equations (1) and (2), or that using capsule data, or both, exceed the screening criteria prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.)

Although the RT_{PTS} value changes are not significant for Vogtle Unit 1, plant specific surveillance capsule data for intermediate shell plate B8805-3 and the weld metal is provided because of the following reasons:

1) There have been two capsules removed from the reactor vessel, and the data is deemed credible per Regulatory Guide 1.99, Revision 2.

2) The surveillance capsule materials are representative of the actual vessel plates and circumferential and longitudinal weld materials.

The chemistry factors for intermediate shell plate B8805-3 and the weld metal were calculated using the surveillance capsule data as shown in Table 6. The chemistry factors were also calculated using Table 2 from 10 CFR 50.61[1].

Tables 7 through 9 provide a summary of the RT_{PTS} values for all beltline region materials for 4.64 EFPY, 32 EFPY, and 48 EFPY, respectively, using the PTS Rule.

Table 6

Calculation of Chemistry Factors Using Vogtle Unit 1 Surveillance Capsule Data[4]

Material	Capsule	Fluence (n/cm^2, E>1.0 MeV)	FF	ΔRT_{NDT} [(b)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Inter. Shell Plate B8805-3 (Longitudinal)	U	3.437×10^{18} [(a)]	0.706	15	10.585	0.498
	Y	1.242×10^{19}	1.060	40	42.4	1.124
Inter. Shell Plate B8805-3 (Transverse)	U	3.437×10^{18} [(a)]	0.706	0	0	0.498
	Y	1.242×10^{19}	1.060	20	21.2	1.124
				Sum:	74.185	3.244
	Chemistry Factor = 74.185 ÷ 3.244 = 22.9 °F					
Weld Metal	U	3.437×10^{18} [(a)]	0.706	15	10.585	0.498
	Y	1.242×10^{19}	1.060	0	0	1.124
				Sum:	10.585	1.622
	Chemistry Factor = 10.585 ÷ 1.622 = 6.5 °F					

(a) Original fluence value has been revised to be based on actual cycle burn up instead of predicted (original value = 3.41×10^{18} from capsule U report, WCAP-12256).

(b) The weld metal ΔRT_{NDT} values contain no adjustment ratio per Regulatory Guide 1.99, Rev. 2 Position 2.1 since the surveillance weld is identical to that used in the core region of the longitudinal seams and the girth seam weld joining the intermediate and lower shells.

Table 7

RT_{PTS} Values for Vogtle Unit 1 for 4.64 EFPY

Material	CF (°F)	Surface Fluence [4] (n/cm², E>1.0 MeV)	FF[1]	ΔRT_{NDT} (°F) (CF x FF)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Plate B8805-1	53.1	3.155×10^{18}	0.6833	36.3	0	34	70.3
Inter. Shell Plate B8805-2	53.1	3.155×10^{18}	0.6833	36.3	20	34	90.3
Inter. Shell Plate B8805-3	38.4	3.155×10^{18}	0.6833	26.2	30	34	90.2
Inter. Shell Plate B8805-3 Using S/C data[2]	22.9	3.155×10^{18}	0.6833	15.6	30	34	79.6
Inter. Shell Plate B8606-1	32.8	3.155×10^{18}	0.6833	22.4	20	34	76.4
Inter. Shell Plate B8606-2	35.2	3.155×10^{18}	0.6833	24.1	20	34	78.1
Inter. Shell Plate B8606-3	41.9	3.155×10^{18}	0.6833	28.6	10	34	72.6
Circ. Weld 101-171	33.2	3.155×10^{18}	0.6833	22.7	-80	56	-1.3
Weld Metal Using S/C data[2]	6.5	3.155×10^{18}	0.6833[3]	4.4	-80	56	-19.6
Long. Weld 101-124A	33.2	1.802×10^{18}	0.5448	18.1	-80	56	-5.9
Long. Weld 101-124B	33.2	2.0×10^{18}	0.5694	18.9	-80	56	-5.1
Long. Weld 101-124C	33.2	2.0×10^{18}	0.5694	18.9	-80	56	-5.1
Long. Weld 101-142A	33.2	2.0×10^{18}	0.5694	18.9	-80	56	-5.1
Long. Weld 101-142B	33.2	1.802×10^{18}	0.5448	18.1	-80	56	-5.9
Long. Weld 101-142C	33.2	2.0×10^{18}	0.5694	18.9	-80	56	-5.1

(1) FF (Fluence Factor) based on inner surface neutron fluence (n/cm², E>1.0 MeV).
(2) Numbers were calculated using a chemistry factor (CF) based on surveillance capsule data.
(3) Peak fluence factor which represents most limiting case for weld metal.

Table 8

RT_{PTS} Values for Vogtle Unit 1 for 32 EFPY

Material	CF (°F)	Surface Fluence [4] (n/cm^2, E>1.0 MeV)	FF[1]	ΔRT_{NDT} (°F) (CF x FF)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Plate B8805-1	53.1	2.176 x 10^{19}	1.2110	64.3	0	34	98.3
Inter. Shell Plate B8805-2	53.1	2.176 x 10^{19}	1.2110	64.3	20	34	118.3
Inter. Shell Plate B8805-3	38.4	2.176 x 10^{19}	1.2110	46.5	30	34	110.5
Inter. Shell Plate B8805-3 Using S/C data[2]	22.9	2.176 x 10^{19}	1.2110	27.7	30	34	91.7
Inter. Shell Plate B8606-1	32.8	2.176 x 10^{19}	1.2110	39.7	20	34	93.7
Inter. Shell Plate B8606-2	35.2	2.176 x 10^{19}	1.2110	42.6	20	34	96.6
Inter. Shell Plate B8606-3	41.9	2.176 x 10^{19}	1.2110	50.7	10	34	94.7
Circ. Weld 101-171	33.2	2.176 x 10^{19}	1.2110	40.2	-80	56	16.2
Weld Metal Using S/C data[2]	6.5	2.176 x 10^{19}	1.2110[3]	7.9	-80	56	-16.1
Long. Weld 101-124A	33.2	1.243 x 10^{19}	1.0606	35.2	-80	56	11.2
Long. Weld 101-124B	33.2	1.379 x 10^{19}	1.0893	36.2	-80	56	12.2
Long. Weld 101-124C	33.2	1.379 x 10^{19}	1.0893	36.2	-80	56	12.2
Long. Weld 101-142A	33.2	1.379 x 10^{19}	1.0893	36.2	-80	56	12.2
Long. Weld 101-142B	33.2	1.243 x 10^{19}	1.0606	35.2	-80	56	11.2
Long. Weld 101-142C	33.2	1.379 x 10^{19}	1.0893	36.2	-80	56	12.2

(1) FF (Fluence Factor) based on inner surface neutron fluence (n/cm^2, E>1.0 MeV).
(2) Numbers were calculated using a chemistry factor (CF) based on surveillance capsule data.
(3) Peak fluence factor which represents most limiting case for weld metal.

Table 9

RT_{PTS} Values for Vogtle Unit 1 for 48 EFPY

Material	CF (°F)	Surface Fluence [(1)] (n/cm², E>1.0 MeV)	FF[(2)]	ΔRT_{NDT} (°F) (CF x FF)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Plate B8805-1	53.1	3.262×10^{19}	1.3104	69.6	0	34	103.6
Inter. Shell Plate B8805-2	53.1	3.262×10^{19}	1.3104	69.6	20	34	123.6
Inter. Shell Plate B8805-3	38.4	3.262×10^{19}	1.3104	50.3	30	34	114.3
Inter. Shell Plate B8805-3 Using S/C data[(2)]	22.9	3.262×10^{19}	1.3104	30.0	30	34	94.0
Inter. Shell Plate B8606-1	32.8	3.262×10^{19}	1.3104	43.0	20	34	97.0
Inter. Shell Plate B8606-2	35.2	3.262×10^{19}	1.3104	46.1	20	34	100.1
Inter. Shell Plate B8606-3	41.9	3.262×10^{19}	1.3104	54.9	10	34	98.9
Circ. Weld 101-171	33.2	3.262×10^{19}	1.3104	43.5	-80	56	19.5
Weld Metal Using S/C data[(3)]	6.5	3.262×10^{19}	1.3104[(4)]	8.5	-80	56	-15.5
Long. Weld 101-124A	33.2	1.863×10^{19}	1.1705	38.9	-80	56	14.9
Long. Weld 101-124B	33.2	2.068×10^{19}	1.1978	39.8	-80	56	15.8
Long. Weld 101-124C	33.2	2.068×10^{19}	1.1978	39.8	-80	56	15.8
Long. Weld 101-142A	33.2	2.068×10^{19}	1.1978	39.8	-80	56	15.8
Long. Weld 101-142B	33.2	1.863×10^{19}	1.1705	38.9	-80	56	14.9
Long. Weld 101-142C	33.2	2.068×10^{19}	1.1978	39.8	-80	56	15.8

(1) Fluence = present fluence + (48 - present EFPY) * present flux * 3.1536×10^7 sec/EFPY.
(2) FF (Fluence Factor) based on inner surface neutron fluence (n/cm², E>1.0 MeV).
(3) Numbers were calculated using a chemistry factor (CF) based on surveillance capsule data.
(4) Peak fluence factor which represents most limiting case for weld metal.

7.0 CONCLUSIONS

As shown in Tables 7 through 9, all RT_{PTS} values remain below the NRC screening values for PTS using fluence values for the present time (4.64 EFPY), projected fluence values for the end of license (32 EFPY), and 48 EFPY. A plot of the RT_{PTS} values versus fluence shown in Figure 2 illustrates the available margin for the most limiting material in the Vogtle Unit 1 reactor vessel beltline region, Intermediate Shell Plate B8805-2.

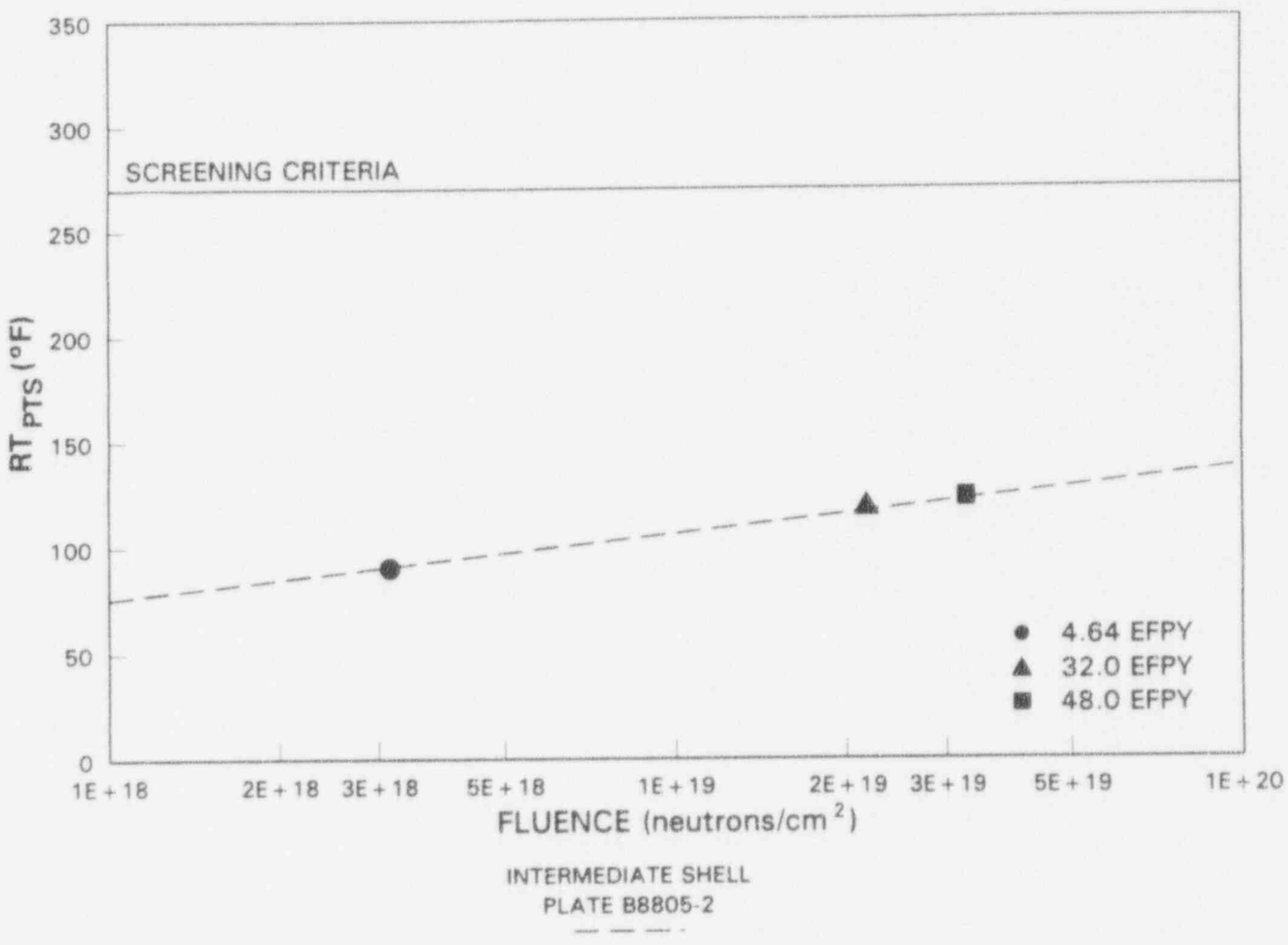

Figure 2. RT_{PTS} Versus Fluence Curves For Vogtle Unit 1 Limiting Material - Intermediate Shell Plate B8805-2

8.0 REFERENCES

[1] 10CFR Part 50, "Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

[2] 10CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)

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

[4] WCAP-13931, "Analysis of Capsule Y From The Georgia Power Company Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program", M. J. Malone, January 1994.

[5] WCAP-11011, "Georgia Power Company Alvin W. Vogtle Unit No. 1 Reactor Vessel Radiation Surveillance Program", L. R. Singer, February 1986.

[6] WCAP-12256, "Analysis of Capsule U From The Georgia Power Company Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko, et al., May 1989.

[7] Combustion Engineering Inc. Drawing No. 8971-161-003, "As Built Location of Weld Seams Vessel and Closure Head Westinghouse Electric Corp. 173" ID PWR", Rev. 01, 8/8/78.

[8] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Contract No. 8971, Job No. 708124-001, Code No. B-8805-1, Heat No. C0613-1, dated October 5, 1972 and Lukens Steel Company Test Certificate, Mill Order No. 10554-1, dated April 25, 1972.

[9] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P18955, Inter. Shell Code B8805-1, dated March 22, 1979.

[10] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Contract No. 8971, Job No. 708124-003, Code No. B-8805-2, Heat No. C0613-2, dated October 5, 1972 and Lukens Steel Company Test Certificate, Mill Order No. 10554-1, dated May 9, 1972.

[11] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P18596, Inter. Shell Code B8805-2, dated March 22, 1979.

[12] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Contract No. 8971, Job No. 708124-005, Code No. B-8805-3, Heat No. C0623-1, dated October 5, 1972 and Lukens Steel Company Test Certificate, Mill Order No. 10554-1, dated May 16, 1972.

[13] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P18957, Inter. Shell Code B8805-3, dated March 22, 1979.

[14] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Revision 1, Contract No. 8971, Job No. 708142-007, Code No. B-8606-1, Heat No. C2146-1, dated March 29, 1974 and Lukens Steel Company Test Certificate, Mill Order No. 12517-2, dated March 23, 1973.

[15] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P15703, Code B8606-1, dated October 30, 1978.

[16] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Revision 1, Contract No. 8971, Job No. 708142-013, Code No. B-8606-2, Heat No. C2146-2, dated March 29, 1974 and Lukens Steel Company Test Certificate, Mill Order No. 12517-2, dated March 23, 1973.

[17] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P13986, Code B8606-2, dated October 30, 1978.

[18] Combustion Engineering Inc., Metallurgical Research and Development Materials Certification Report, Revision 1, Contract No. 8971, Job No. 708142-011, Code No. B-8606-3, Heat No. C2085-2, dated March 29, 1974 and Lukens Steel Company Test Certificate, Mill Order No. 12517-1, dated March 30, 1973.

[19] Combustion Engineering Power Systems Interoffice Correspondence to A.B. Harper from W.A. House, Lab No. P15704, Code B8606-3, dated October 30, 1978.

[20] Combustion Engineering Power Systems Welding Material Qualification To Requirements of ASME Section III, Job No. D32255, Project Number 960009, dated November 6, 1972.

[21] Combustion Engineering Power Systems Interoffice Correspondence from P. C. Kiefer, Qualification Code G1.43, Job No. D32255, dated November 2, 1972.

[22] "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, Rev. 1 July 1981.