WCAP-14534

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR VOGTLE ELECTRIC GENERATING PLANT (VEGP) UNIT 2

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PREFACE

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SECTION 1.0
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 the propagation of flaws 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 Vogtle Unit 2 reactor vessel using the results of the surveillance Capsule Y 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 Vogtle Unit 2 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 that all PTS screening criteria are satisfied at end-of-license (EOL) and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

SECTION 2.0
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 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.

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 currently described in Regulatory Guide 1.99, Revision 2[2].
2. The section is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of life 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 vessel 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 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.

SECTION 3.0

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.

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

$RT_{NDT(U)}$ = 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 ΔRT_{NDT}.

For plates and forgings:

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

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

For welds:

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

σ_Δ = 14°F when credible 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 RT_{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 given in Table 1 for welds and Table 2 for base metal (plates or forgings) of 10 CFR Part 50.61. If surveillance data is deemed credible, it must be used to determine the material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the best estimate 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 ΔRT_{PTS} and Equation 2 for determining M.

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

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 levels 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.

When credible surveillance data is available, a material-specific value of CF is determined from Equation 5.

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

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 he vessel weld and the surveillance weld, the measure 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.

SECTION 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 2 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat-affected zones and plates 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." Figure 1 identifies and indicates the location of all beltline region materials for the Vogtle Unit 2 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[3]. 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 and 2. A summary of the pertinent chemical and mechanical properties of the beltline region plates and weld materials of the Vogtle Unit 2 reactor vessel are given in Table 3.

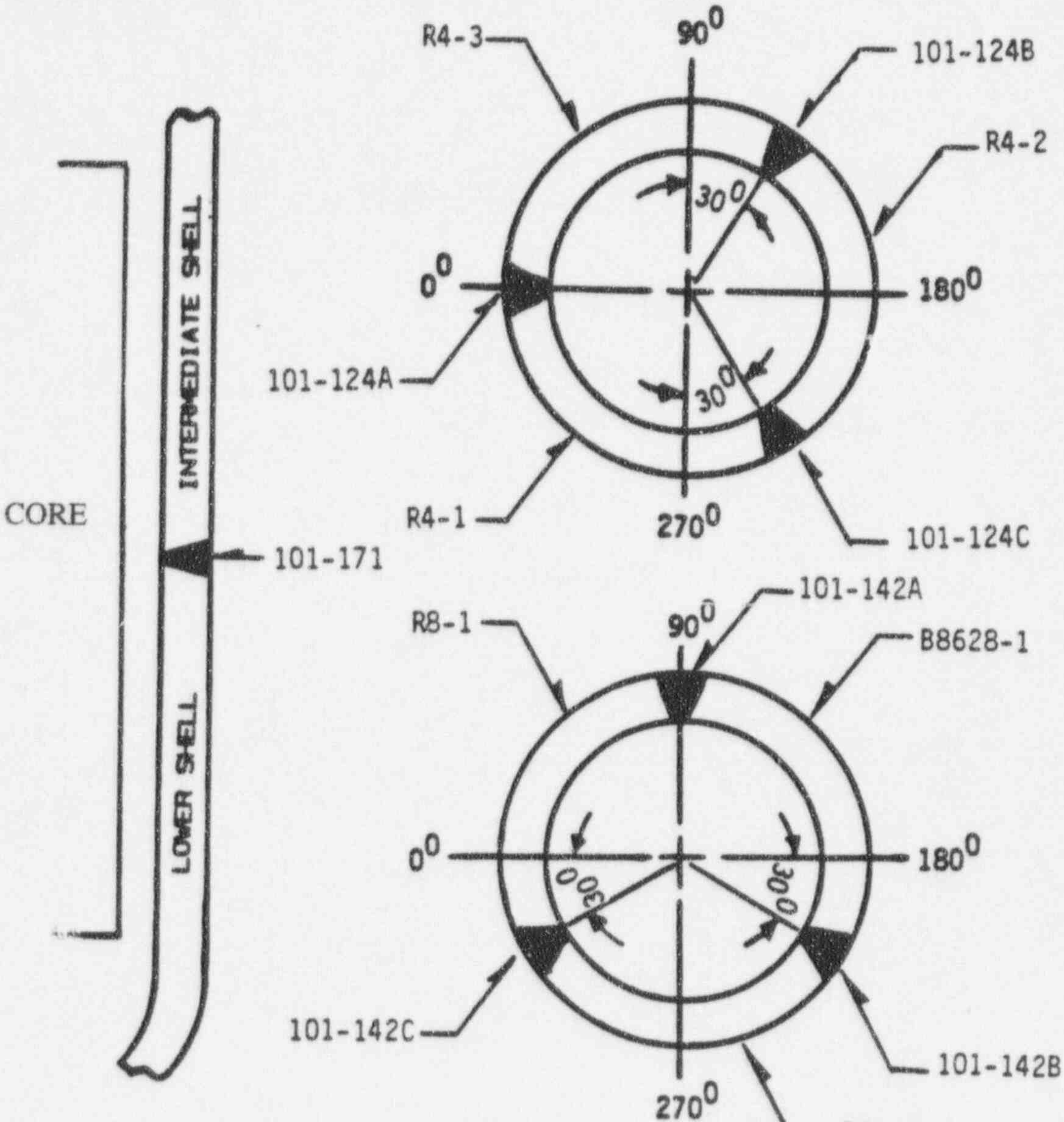

FIGURE 1 Identification and Location of Beltline Region Materials for the Vogtle Unit 2 Reactor Vessel

TABLE 1

Calculation of Average Cu and Ni Weight Percent Values for Base Materials

	Inter. Shell Plate R4-1		Inter. Shell Plate R4-2		Inter. Shell Plate R4-3		Lower Shell Plate R8-1		Lower Shell Plate B8628-1*		Lower Shell Plate B8825-1	
Ref.	Cu %	Ni %	Cu%	Ni %	Cu%	Ni %	Cu%	Ni %	Cu %	Ni %	Cu %	Ni %
4	0.07	0.61	0.07	0.59	0.05	0.60						
5							0.07	0.64	0.07	0.63	0.07	0.64
3	0.06	0.64	0.05	0.62	0.05	0.59	0.06	0.62	0.05	0.59	0.05	0.59
3									0.05	0.59		
6									0.053	0.598		
7									0.049	0.549		
Avg.	0.07	0.63	0.06	0.61	0.05	0.60	0.07	0.63	0.05	0.59	0.06	0.62

NOTE:

* Surveillance program base metal material.

TABLE 2

Calculation of Average Cu and Ni Weight Percent Values for Weld Materials

Reference	Weld Material		
	Cu %	Ni %	Weighting Factor
3: WCAP-11381	0.07	0.13	1
3: WCAP-11381	0.06	0.12	1
3: WCAP-11381 Tandem Arc*	0.04	0.17	2
6: WCAP-13007 Tandem Arc*	0.045	0.091	2
7: Cap. Y Chem. Analysis Tandem Arc*	0.039	0.127	2
7: Tandem Arc*	0.037	0.118	2
7: Tandem Arc*	0.040	0.137	2
8: GL 92-01, Supp. 1 R3493	0.03	--	1
8: Supplier Analysis	0.04	--	1
8: D18140	0.04	0.15	1
8: D18142	0.04	0.16	1
8: D18138	0.04	0.17	1
8: D18139	0.04	0.17	1
8: D21867	0.05	0.15	1
8: D18143	0.05	0.19	1
8: D21865	0.05	0.20	1
8: D21866	0.05	0.21	1
8: D18141	0.06	0.27	1
Average	0.04	0.15	--

NOTE:

* Per the request of the Southern Nuclear Operating Company (SNC), the Cu and Ni weight percent values will be determined per the methods used by SNC in the Vogtle Unit 2 Generic Letter 92-01, Revision 1, Supplement 1 submittal[8]. Surveillance program weld metal specimens are of the tandem arc weld type, and therefore, will use a weighting factor of 2 in the calculation of the average Cu and Ni values for the weld metal.

TABLE 3
Vogtle Unit 2 Reactor Vessel Beltline Region Material Properties

Material Description	Cu (%) [a]	Ni (%) [a]	$RT_{NDT(U)}$ (°F) [b]
Intermediate Shell Plate R4-1	0.07	0.63	10
Intermediate Shell Plate R4-2	0.06	0.61	10
Intermediate Shell Plate R4-3	0.05	0.60	30
Lower Shell Plate R8-1	0.07	0.63	40
Lower Shell Plate B8628-1	0.05	0.59	50
Lower Shell Plate B8825-1	0.06	0.62	40
Inter. and Lower Shell Long. Welds	0.04	0.15	-10
Circumferential Weld	0.04	0.15	-30

NOTES:
(a) Average values of copper and nickel as indicated in Tables 1 and 2 on preceding pages.
(b) The $RT_{NDT(U)}$ values for the plates and welds are measured values per U.S. NRC Standard Review Plan [9].

SECTION 5.0

NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Vogtle Unit 2 reactor vessel are shown in Table 4. These values were projected using the results of the Capsule Y radiation analysis[7].

TABLE 4

Fluence (10^{19} n/cm^2, E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for Vogtle Unit 2

EFPY	0°	15°	30° (a)	30° (b)	30° (c)	45°
4.83	0.175	0.254	0.306	0.200	0.166	0.304
16	0.581	0.842	1.02	0.663	0.551	1.01
32	1.16	1.69	2.03	1.33	1.10	2.01
48	1.74	2.53	3.05	1.99	1.65	3.02
54	1.96	2.84	3.43	2.24	1.86	3.40

NOTES:

(a) Indicates location in octants with a 12.5° neutron pad span (no surveillance capsules).
(b) Indicates location in octants with a 20.0° neutron pad span (single capsule holder).
(c) Indicates location in octants with a 22.5° neutron pad span (dual capsule holder).

SECTION 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 2 reactor vessel for fluence values at the EOL (32 and 54 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. Vogtle Unit 2 plant-specific surveillance capsule data for lower shell plate B8628-1 and the weld metal is provided for the following reasons:

1) There have been two capsules removed from the reactor vessel and the data is deemed credible per 10 CFR Part 50.61. (See Appendix B of WCAP-14532, Reference 7.)

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

Per the Southern Nuclear Operating Company, the beltline region materials of the Vogtle Unit 2 reactor vessel are not contained in any other commercial plant reactor vessel surveillance program.

As presented in Table 5, chemistry factor values for Vogtle Unit 2 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 6. Tables 7 and 8 contain the RT_{PTS} calculations for all beltline region materials for 32 and 54 EFPY, respectively.

TABLE 5

Interpolation of Chemistry Factors from Regulatory Guide 1.99, Revision 2, Position 1.1

Material	Ni, wt %	Chemistry Factor, °F
Intermediate Shell Plate R4-1 Given Cu wt% = 0.07	0.63	44
Intermediate Shell Plate R4-2 Given Cu wt % = 0.06	0.61	37
Intermediate Shell Plate R4-3 Given Cu wt % = 0.05	0.60	31
Lower Shell Plate R8-1 Given Cu wt % = 0.07	0.63	44
Lower Shell Plate B8628-1 Given Cu wt % = 0.05	0.59	31
Lower Shell Plate B8825-1 Given Cu wt % = 0.06	0.62	37
Weld Metal Given Cu wt % = 0.04	0.15	38.3

NOTE:
The weld metal CF was determined using a Cu weight percent value of 0.04% and interpolating between Ni weight percent values of 0% and 0.20% from 10 CFR Part 50.61, Table 1. Specifically, the CF for Ni = 0% is 24°F and Ni = 0.20% is 43°F, respectively. Therefore, for a Cu weight percent value of 0.04% and a Ni weight percent value of 0.15%, the weld metal CF value is interpolated to be 38.3°F.

TABLE 6

Calculation of Chemistry Factors Using Credible Surveillance Capsule Data

Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f	FF	ΔRT_{NDT}	FF*ΔRT_{NDT}	FF^2
Lower Shell Plate B8628-1 (Longitudinal)	U	0.422	0.760	2.12	1.61	0.578
	Y	1.13	1.03	5.76	5.96	1.07
Lower Shell Plate B8628-1 (Transverse)	U	0.422	0.760	0.00	0.00	0.578
	Y	1.13	1.03	1.93	2.00	1.07
				Sum:	9.56	3.30
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) =$					2.9
Weld Metal	U	0.422	0.760	0.00	0.00	0.578
	Y	1.13	1.03	18.59	19.15	1.07
				Sum:	19.15	1.65
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) =$					11.6

NOTES:

f = fluence (10^{19} n/cm^2); All updated fluence values taken from Section 6.0 of the Capsule Y analysis, WCAP-14532[7].

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

ΔRT_{NDT} values obtained from CVGRAPH Version 4.0. (See WCAP-14532, Reference 7.) These values differ from those previously reported in WCAP-13007 since those were hand-fit using engineering judgement.

TABLE 7

RT_{PTS} Calculations for Vogtle Unit 2 Beltline Region Materials at 32 EFPY

Material	CF	f	FF	$RT_{NDT(U)}$	M	ΔRT_{PTS}	RT_{PTS}
EOL - 32 EFPY							
Inter. Shell Plate R4-1	44.0	2.03	1.19	10	34	52.5	96
Inter. Shell Plate R4-2	37.0	2.03	1.19	10	34	44.1	88
Inter. Shell Plate R4-3	31.0	2.03	1.19	30	34	37.0	101
Lower Shell Plate R8-1	44.0	2.03	1.19	40	34	52.5	**126**
Lower Shell Plate B8628-1	31.0	2.03	1.19	50	34	37.0	121
using surv. capsule data	2.9	2.03	1.19	50	3.5	3.5	57
Lower Shell Plate B8825-1	37.0	2.03	1.19	40	34	44.1	118
Inter. Shell Long. Weld 101-124A	38.3	1.16	1.04	-10	39.9	39.9	70
Inter. Shell Long. Weld 101-124B	38.3	1.33	1.08	-10	41.3	41.3	73
Inter. Shell Long. Weld 101-124C	38.3	1.10	1.03	-10	39.3	39.3	69
Lower Shell Long. Weld 101-142A	38.3	1.16	1.04	-10	39.9	39.9	70
Lower Shell Long. Weld 101-142B	38.3	2.03	1.19	-10	45.7	45.7	81
Lower Shell Long. Weld 101-142C	38.3	2.03	1.19	-10	45.7	45.7	81
Circumferential Weld 101-171	38.3	2.03	1.19	-30	45.7	45.7	61
using surv. capsule data	11.6	2.03	1.19	-30	13.8	13.8	-2

NOTES:
FF = fluence factor = $f^{(0.28 - 0.10 \cdot \log f)}$
$RT_{NDT(U)}$ values are measured values.

TABLE 8

RT_{PTS} Calculations for Vogtle Unit 2 Beltline Region Materials at 54 EFPY

Material	CF	f	FF	$RT_{NDT(U)}$	M	ΔRT_{PTS}	RT_{PTS}
54 EFPY							
Inter. Shell Plate R4-1	44.0	3.42	1.32	10	34	58.1	102
Inter. Shell Plate R4-2	37.0	3.42	1.32	10	34	48.9	93
Inter. Shell Plate R4-3	31.0	3.42	1.32	30	34	41.0	105
Lower Shell Plate R8-1	44.0	3.42	1.32	40	34	58.1	**132**
Lower Shell Plate B8628-1	31.0	3.42	1.32	50	34	41.0	125
using surv. capsule data	2.9	3.42	1.32	50	3.8	3.8	58
Lower Shell Plate B8825-1	37.0	3.42	1.32	40	34	48.9	123
Inter. Shell Long. Weld 101-124A	38.3	1.96	1.18	-10	45.3	45.3	81
Inter. Shell Long. Weld 101-124B	38.3	2.25	1.22	-10	46.7	46.7	83
Inter. Shell Long. Weld 101-124C	38.3	1.85	1.17	-10	44.8	44.8	80
Lower Shell Long. Weld 101-142A	38.3	1.96	1.18	-10	45.3	45.3	81
Lower Shell Long. Weld 101-142B	38.3	3.42	1.32	-10	50.6	50.6	91
Lower Shell Long. Weld 101-142C	38.3	3.42	1.32	-10	50.6	50.6	91
Circumferential Weld 101-171	38.3	3.42	1.32	-30	50.6	50.6	71
using surv. capsule data	11.6	3.42	1.32	-30	15.3	15.3	1

NOTES:
FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$
$RT_{NDT(U)}$ values are measured values.

SECTION 7.0
CONCLUSIONS

As shown in Tables 7 and 8, all of the beltline region materials in the Vogtle Unit 2 reactor vessel have EOL RT_{PTS} values well below the screening criteria values of 270°F for plates and longitudinal welds and 300°F for circumferential welds at EOL (32 and 54 EFPY).

SECTION 8.0
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, effective January 18, 1996.

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

3. WCAP-11381, "Georgia Power Company Alvin W. Vogtle Unit No. 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, April 1986.

4. Metallurgical Research and Development Dept., Materials Certification Reports, Vendor - Lukens Steel Company, Contract No. 7372, Code Nos. R-4-1, R-4-2, and R-4-3, J. M. Arnold, April 5, 1974.

5. Metallurgical Research and Development Dept., Materials Certification Reports, Vendor - Lukens Steel Company, Contract No. 7372, Code Nos. B-8628-1, B-8825-1, and R-8-1, C. E. Bingham, March 12, 1975.

6. WCAP-13007, "Analysis of Capsule U from the Georgia Power Company Vogtle Electric Generating Plant Unit 2 Reactor Vessel Radiation Surveillance Program", E. Terek, et al., August 1991.

7 WCAP-14532, "Analysis of Capsule Y from the Georgia Power Company Vogtle Unit 2 Reactor Vessel Radiation Surveillance Program", P. A. Grendys, et al. February 1996.

8. LCV-0648B, "Vogtle Electric Generating Plant Response to Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity", C. K. McCoy, dated 11/15/95.

9. "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.