WCAP-13874

Westinghouse Non-Proprietary Class 3

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR CATAWBA UNIT 2

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Work Performed Under Shop Order XARP-93209

Prepared by Westinghouse Electric Corporation
for Duke 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 followed by high repressurization;
- 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 nil-ductility reference 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 the end-of-license plant operation. 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 has 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 Catawba Unit 2 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 Catawba Unit 2 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 conclusions 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 (USNRC). 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 this Rule change, whichever comes first. These values must be calculated based on the methodology specified in this 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 following equations must be used to calculate the RT_{PTS} values for each weld, plate or forging in the reactor vessel beltline:

  Equation 1: $RT_{PTS} = I + M + \Delta RT_{PTS}$

  Equation 2: $\Delta RT_{PTS} = CF * f^{(0.28-0.10 \log f)}$

* 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.
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, n/cm^2 (E > 1 MeV at the clad/base metal interface), divided by 10^{19}

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 Regulatory Guide 1.99, Revision 2, 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 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 Catawba 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[4,5]. The average copper and nickel values were calculated for each of the beltline region materials using all of the available material chemistry information.

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the Catawba Unit 2 reactor vessel are given in Table 1. All of the initial RT_{NDT} values ($I\text{-}RT_{NDT}$) are also presented in Table 1.

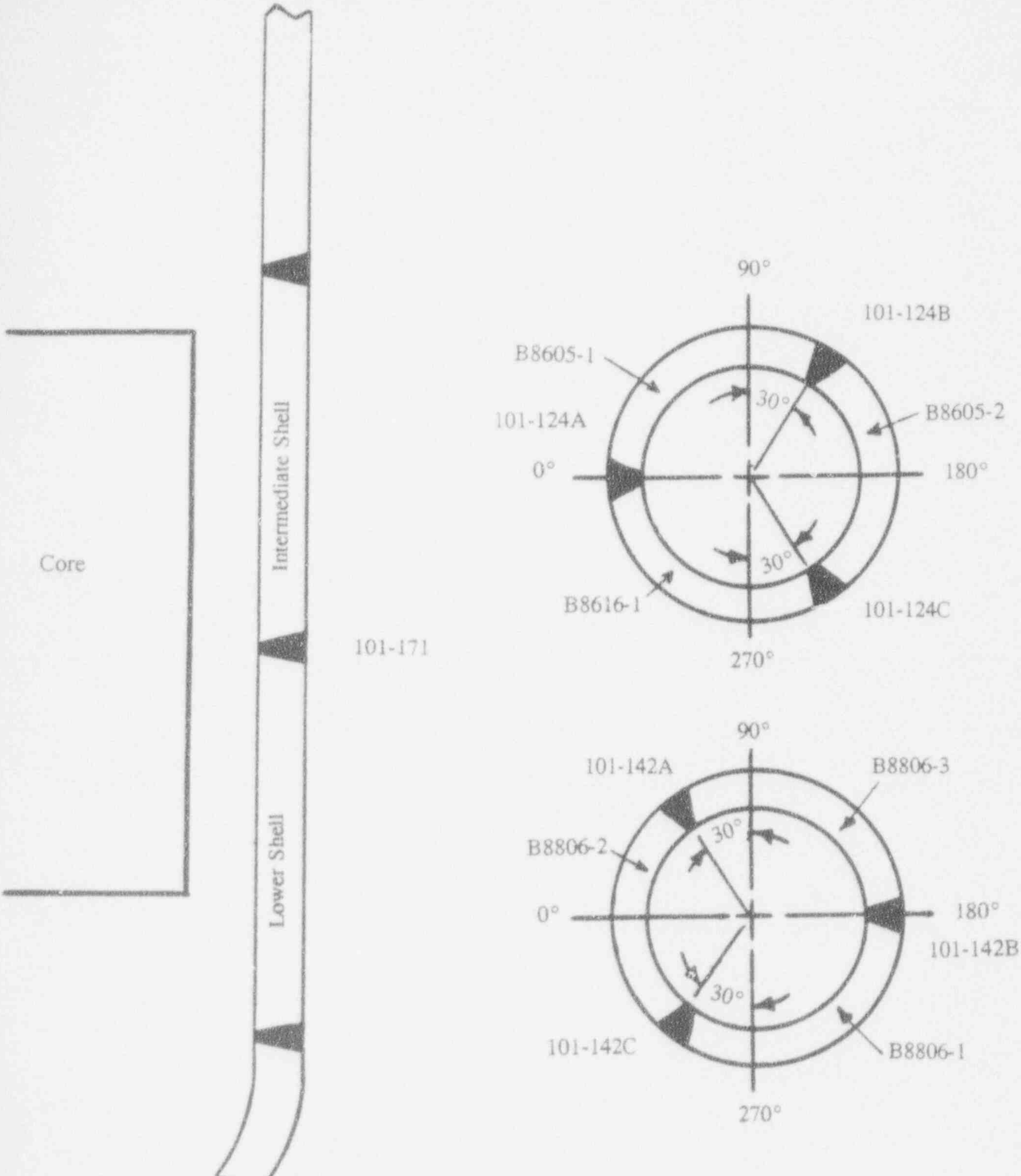

Figure 1. Identification and Location of Beltline Region Materials for the Catawba Unit 2 Reactor Vessel

TABLE 1

CATAWBA UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Material Description	Cu (%) *	Ni (%) *	I-RT_{NDT} (°F)
Intermediate Shell, B8605-1	0.082	0.618	15
using S/C data			15
Intermediate Shell, B8605-2	0.080	0.613	33
Intermediate Shell, B8616-1	0.045	0.595	12
Lower Shell, B8806-1	0.057	0.560	6
Lower Shell, B8806-2	0.057	0.593	-10
Lower Shell, B8806-3	0.057	0.593	8
Longitudinal Welds	0.042	0.153	-80
using S/C data			-80
Circumferential Weld	0.042	0.153	-80
using S/C data			-80

* Average values of copper and nickel as indicated in the following tables

Reference	Inter. Shell, B8605-1		Inter. Shell, B8605-2		Inter. Shell, B8616-1	
	wt % Cu	wt % Ni	wt % Cu	wt % Ni	wt % Cu	wt % Ni
Surveillance Program [4]	0.071	0.590				
Capsule Z Report [5]	0.085	0.640				
Chemical Analysis [6]	0.080	0.610				
Chemical Analysis [6]	0.090	0.630				
Chemical Analysis [7]			0.080	0.610		
Chemical Analysis [7]			0.090	0.620		
Letter from CE [12]			0.070	0.610		
Chemical Analysis [8]					0.040	0.600
Chemical Analysis [8]					0.050	0.590
Average	0.082	0.618	0.080	0.613	0.045	0.595

Reference	Lower Shell, B8806-1		Lower Shell, B8806-2		Lower Shell, B8806-3	
	wt % Cu	wt % Ni	wt % Cu	wt % Ni	wt % Cu	wt % Ni
Chemical Analysis [9]	0.060	0.570				
Chemical Analysis [9]	0.060	0.550				
Chemical Analysis [10]			0.060	0.590		
Chemical Analysis [10]			0.060	0.600		
Chemical Analysis [11]					0.060	0.590
Chemical Analysis [11]					0.060	0.600
Letter from CE [12]	0.050	0.560	0.050	0.590	0.050	0.590
Average	0.057	0.560	0.057	0.593	0.057	0.593

Reference	Surveillance Weld *	
	wt % Cu	wt % Ni
Surveillance Program [4]	0.040	0.140
Surveillance Program [4]	0.036	0.140
Capsule Z Report [5]	0.051	0.180
Letter from CE [13]	0.040	
Average	0.042	0.153

* Per Reference 4, 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. 83648, Linde 0091 Flux, Lot No. 3536.

5.0 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E>1.0 MeV) at the ıner surface of the Catawba Unit 2 reactor vessel is shown in Table 2. These values were projected using the results of the Capsule X radiation surveillance program[14]. The RT_{PTS} calculations were performed using the peak fluence value, which occurs at the 25° azimuth (except for the longitudinal welds which are located at 30° azimuth) in the Catawba Unit 2 reactor vessel.

TABLE 2

NEUTRON EXPOSURE PROJECTIONS* AT KEY LOCATIONS ON THE CATAWBA UNIT 2 PRESSURE VESSEL CLAD/BASE METAL INTERFACE FOR 4.516 AND 32 EFPY [14]

EFPY	0°	15°	25°	30°	35°	45°
4.516	0.234	0.328	0.347	0.209	0.263	0.289
32	1.66	2.32	2.46	1.48	1.86	2.05

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

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 Catawba Unit 2 reactor vessel as a function of present time (4.516 EFPY per Capsule X analysis) and end-of-life (32 EFPY) fluence values. The fluence data was generated based on the most recent surveillance capsule program results [14].

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.)

For Catawba Unit 2, the use of plant specific surveillance capsule data arises for the Intermediate Shell, B8605-1 and Surveillance Welds 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 intermediate shell and surveillance weld materials.

The chemistry factors for the Intermediate Shell, B8605-1 and Surveillance Welds were calculated using the surveillance capsule data as shown in Table 3. The chemistry factors for the lower shells and other intermediate shells were calculated using Table 2 from 10 CFR 50.61[2].

Tables 4 and 5 provide a summary of the RT_{PTS} values for all beltline region materials for 4.516 EFPY and end-of-license (32 EFPY), respectively, using the PTS Rule.

TABLE 3

CALCULATION OF CHEMISTRY FACTORS USING

CATAWBA UNIT 2 SURVEILLANCE CAPSULE DATA [14]

Material	Capsule	Fluence	FF	ΔRT_{NDT}	FF*ΔRT_{NDT}	FF^2
Inter. Shell, B8605-1 (Long.)	Z	3.435×10^{18}	0.706	20	14.12	0.498
	X	1.19×10^{19}	1.055	45	47.48	1.113
Inter. Shell, B8605-1 (Trans.)	Z	3.435×10^{18}	0.706	40	28.24	0.498
	X	1.19×10^{19}	1.055	55	58.03	1.113
				Sum:	147.87	3.222
	Chemistry Factor = 147.87 ÷ 3.222 = 45.89					
Weld Metal	Z	3.435×10^{18}	0.706	0	0	0.498
	X	1.19×10^{19}	1.055	35	36.93	1.113
				Sum:	36.93	1.611
	Chemistry Factor = 36.93 ÷ 1.611 = 22.92					

TABLE 4

RT_{PTS} VALUES FOR CATAWBA UNIT 2 FOR 4.516 EFPY

Material	ΔRT_{NDT} (°F) (CF X FF*)		Initial RT_{NDT} (°F)	Margin (°F)	RT_{PTS} (°F)
Intermediate Shell, B8605-1	52.4	0.7083	15	34	86.1
using S/C data	(45.89)	0.7083	15	34	(81.5)
Intermediate Shell, B8605-2	51.0	0.7083	33	34	103.1
Intermediate Shell, B8616-1	28.5	0.7083	12	34	66.2
Lower Shell, B8806-1	35.2	0.7083	6	34	64.9
Lower Shell, B8806-2	35.2	0.7083	-10	34	48.9
Lower Shell, B8806-3	35.2	0.7083	8	34	66.9
Longitudinal Welds	40.1	0.5800	-80	56	-0.7
using S/C data	(22.92)	0.5800	-80	56	(-10.7)
Circumferential Weld	40.1	0.7083	-80	56	4.4
using S/C data	(22.92)	0.7083	-80	56	(-7.8)

( ) Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 3.47×10^{18} n/cm^2 (2.09×10^{18} n/cm^2 for the longitudinal welds) [14].

TABLE 5

RT_{PTS} VALUES FOR CATAWBA UNIT 2 FOR 32 EFPY

Material	ΔRT_{NDT} (°F) (CF X FF*)		Initial RT_{NDT} (°F)	Margin (°F)	RT_{PTS} (°F)
Intermediate Shell, B8605-1	52.4	1.2422	15	34	114.1
using S/C data	(45.89)	1.2422	15	34	(106.0)
Intermediate Shell, B8605-2	51.0	1.2422	33	34	130.4
Intermediate Shell, B8616-1	28.5	1.2422	12	34	81.4
Lower Shell, B8806-1	35.2	1.2422	6	34	83.7
Lower Shell, B8806-2	35.2	1.2422	-10	34	67.7
Lower Shell, B8806-3	35.2	1.2422	8	34	85.7
Longitudinal Welds	40.1	1.1086	-80	56	20.5
using S/C data	(22.92)	1.1086	-80	56	(1.4)
Circumferential Weld	40.1	1.2422	-80	56	25.8
using S/C data	(22.92)	1.2422	-80	56	(4.5)

( ) Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 2.46×10^{19} n/cm^2 (1.48×10^{19} n/cm^2 for the longitudinal welds) [14].

7.0 CONCLUSIONS

As shown in Tables 4 and 5, all RT_{PTS} values remain below the NRC screening values for PTS using fluence values for the present time (4.516 EFPY) and projected fluence values for the end-of-license (32 EFPY). A plot of the RT_{PTS} values versus fluence is shown in Figure 2 for the most limiting material in the Catawba Unit 2 reactor vessel beltline region, Intermediate Shell, B8605-2.

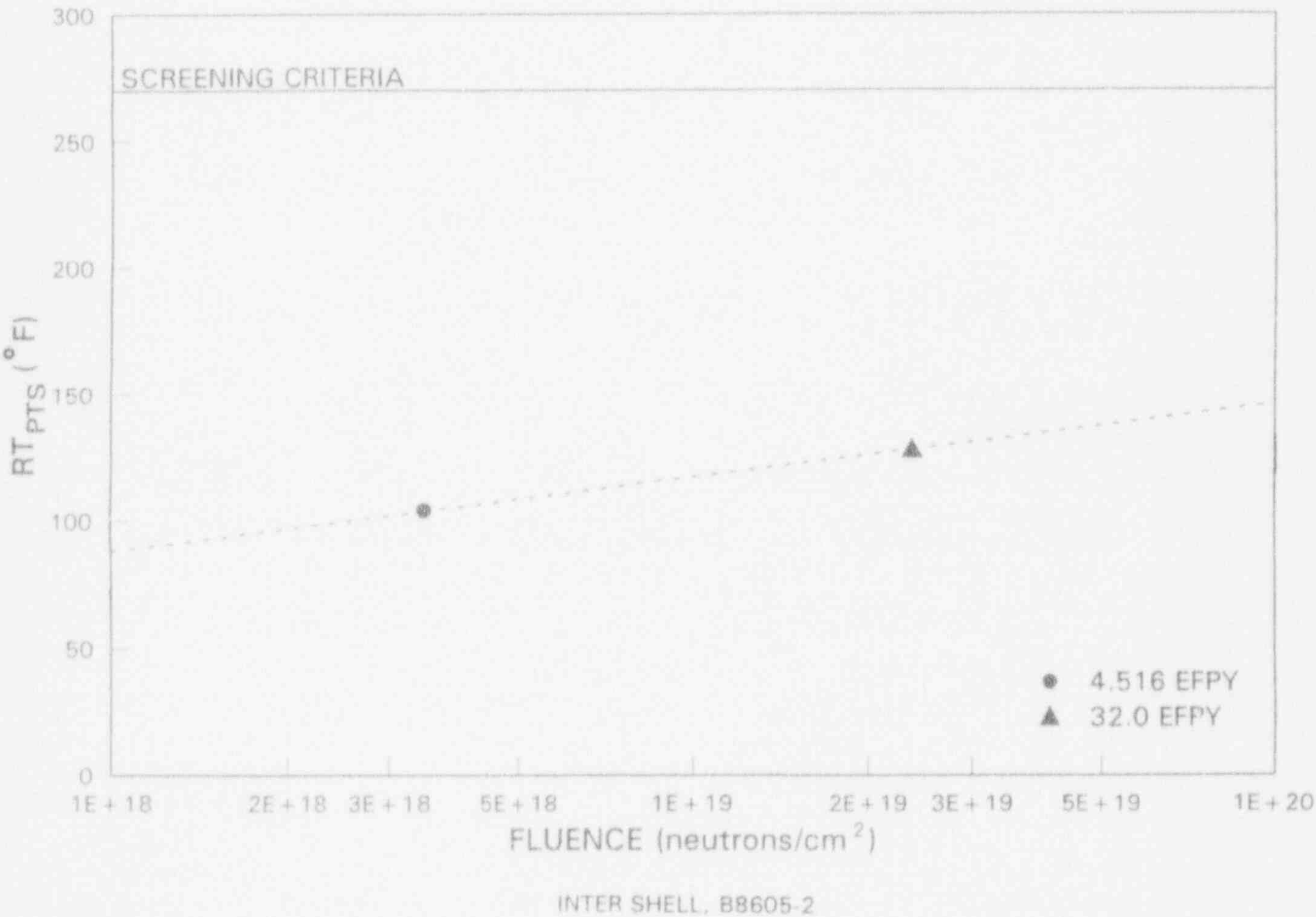

Figure 2. RT_{PTS} versus Fluence Curves for Catawba Unit 2 Limiting Material - Intermediate Shell, B8605-2

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-10868, "Duke Power Company Catawba Unit No. 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, November 1985.

[5] WCAP-11941, "Analysis of Capsule Z from the Duke Power Company Catawba Unit No. 2 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko, September 1988.

[6] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Revision 2, Contract No. 8871, Job No. 707124-001, dated May 22, 1974 and Lukens Steel Company Test Certificate, Mill Order No. 10553-1, dated April 14, 1972.

[7] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Revision 1, Contract No. 8871, Job No. 707124-003, dated May 22, 1974 and Lukens Steel Company Test Certificate, Mill Order No. 10553-1, dated April 14, 1972.

[8] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Contract No. 8871, Job No. 707124-011, dated February 22, 1973 and Lukens Steel Company Test Certificate, Mill Order 27495-1, dated January 17, 1973.

[9] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Contract No. 8871, Job No. Rej. Notice 03664-001, dated October 3, 1973 and Lukens Steel Company Test Certificate, Mill Order No. 26602-1, dated April 18, 1973.

[10] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Contract No. 8871, Job No. 707124-009, dated August 16, 1973 and Lukens Steel Company Test Certificate, Mill Order No. 26602-1, dated April 18, 1973.

[11] Combustion Engineering, Inc. Metallurgical Research and Development Dept. Materials Certification Report, Revision 2, Contract No. 8871, Job No. 707124-011, dated August 16, 1973 and Lukens Steel Company Test Certificate, Mill Order No. 26602-1, dated April 26, 1973.

[12] Letter from Combustion Engineering Metallurgical & Materials Laboratory - Chattanooga to Westinghouse, "Westinghouse Electric Corp. Surveillance Test Material", W. A. House, Contract No. 8871, Project No. 960001, dated April 9, 1976.

[13] Letter from Combustion Engineering, Inc. to Westinghouse, "Core Region Weld Materials Contract 161053/8871 & 161054/8971", M. C. Herder, dated April 8, 1974 and Combustion Engineering Metallurgical Research and Development Department - Chattanooga, Welding Material Qualification to requirements of ASME Section III, Job Number D32255, Project Number 960009, December 12, 1972. (Code G1.45)

[14] WCAP-13875, "Analysis of Capsule X from the Duke Power Company Catawba Unit 2 Reactor Vessel Radiation Surveillance Program", E. Terek, dated February, 1993.