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EVALUATION OF PRESSURIZED THERMAL SHOCK FOR BRAIDWOOD UNIT 1

P. A. Peter

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Approved by: R D Rishel
R. D. Rishel, Manager
Metallurgical & NDE Analysis

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Technology Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

PREFACE

This report has been technically reviewed and verified by:

P. L. Strauch E. Trak FOR P.L.S.

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SECTION 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 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 Braidwood Unit 1 reactor vessel to address the revised PTS Rule. Section 2.0 discusses the 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 Braidwood Unit 1 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 conclusions and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

SECTION 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 material 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 material.

* 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.

SECTION 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, per Regulatory Guide 1.99, Revision 2, in °F.

$M = \text{margin} = 2\sqrt{\sigma_\Delta^2 + \sigma_i^2}$, °F

σ_i = 0°F when I is a measured value

σ_i = 17°F when I is a generic value

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 $0.5*\Delta RT_{NDT}$

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 from the surveillance program[5] has been deemed credible per Regulatory Guide 1.99, Revision 2, it may be considered in the calculation of the chemistry factor.

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 Braidwood 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 material for the Braidwood 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[4]. The copper and nickel values used in the calculations were obtained from Reference 5. A summary of the pertinent chemical and mechanical properties of the beltline region forgings and weld material of the Braidwood Unit 1 reactor vessel are given in Table 1.

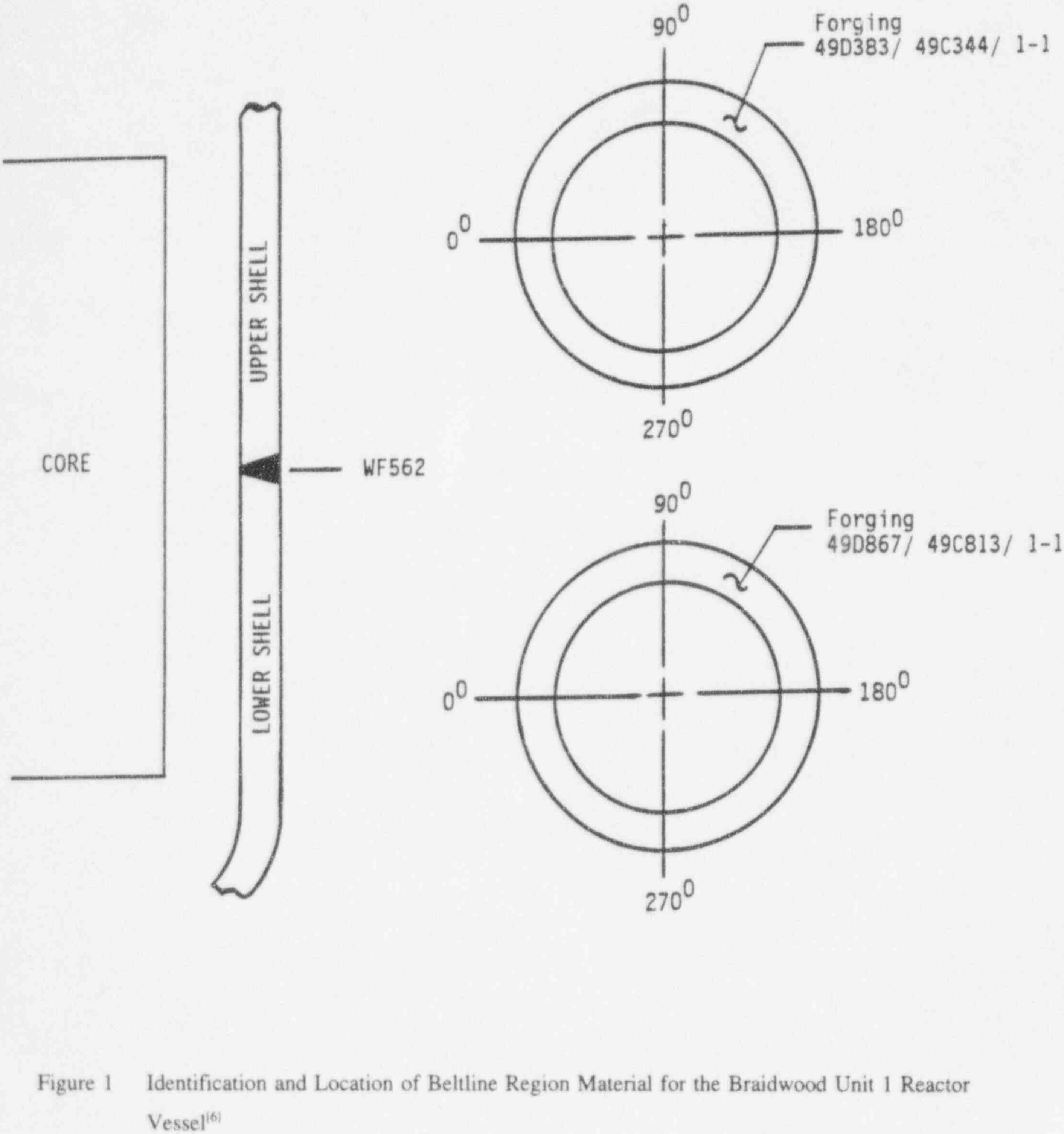

Figure 1 Identification and Location of Beltline Region Material for the Braidwood Unit 1 Reactor Vessel[6]

TABLE 1

Braidwood Unit 1 Reactor Vessel Beltline Region Material Properties Used in Calculations

Material	Cu%	Ni%	Chemistry Factor	Initial RT_{NDT} [(a)]
Intermediate Shell Forging	0.05	0.73	31.0 [(b)]	-30
Lower Shell Forging	0.04	0.74	26.0 [(b)]	-20
using S/C data			18.8 [(c)]	-20
Weld Metal	0.03	0.67	41.0 [(b)]	40
using S/C data			20.6 [(c)]	40

NOTE:

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

(b) Chemistry Factor calculated per Regulatory Guide 1.99, Revision 2, Position 1.1.

(c) Chemistry Factor calculated per Regulatory Guide 1.99, Revision 2, Position 2.1.

SECTION 5.0
NEUTRON FLUENCE VALUES

The calculated peak fast neutron fluence (E>1.0 MeV) values at the inner surface of the Braidwood Unit 1 reactor vessel are shown in Table 2. These values were projected using the results of the Capsule X radiation surveillance program[5]. The RT_{PTS} calculations were performed using peak fluence values, which occur at the 25° azimuthal angle of the Braidwood Unit 1 reactor vessel.

TABLE 2

Neutron Exposure (n/cm^2, E>1.0 MeV) Projections at Peak Location on the Braidwood Unit 1 Pressure Vessel Clad/Base Metal Interface

EFPY	25° Base & Weld Metal
4.234	2.963×10^{18}
32	2.239×10^{19}
48	3.358×10^{19}

SECTION 6.0

DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIAL

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region material of the Braidwood Unit 1 reactor vessel for fluence values at the present time (4.234 EFPY per Capsule X 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 Braidwood Unit 1, plant specific surveillance capsule data for the lower shell forging and 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 forging and weld material.

The chemistry factors for the lower shell forging and weld metal were calculated using the surveillance capsule data as shown in Table 3. The chemistry factors were also calculated using Tables 1 and 2 from 10 CFR 50.61[2]. Table 4 provides a summary of the RT_{PTS} values for all beltline region material for 4.234, 32, and 48 EFPY.

TABLE 3

Calculation of Chemistry Factors Using Braidwood Unit 1 Surveillance Capsule Data[5]

Material	Capsule	f	FF	ΔRT_{NDT}	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging (Tangential)	U	0.3814	0.733	5	3.666	0.538
	X	1.144	1.038	30	31.127	1.077
Lower Shell Forging (Axial)	U	0.3814	0.733	0	0.000	0.538
	X	1.144	1.038	25	25.939	1.077
				Sum:	60.733	3.228
	$CF = \sum(FF * RT_{NDT}) \div \sum(FF^2) = 18.8$					
Weld Metal	U	0.3814	0.733	10	7.333	0.538
	X	1.144	1.038	25	25.939	1.077
				Sum:	33.272	1.614
	$CF = \sum(FF * RT_{NDT}) \div \sum(FF^2) = 20.6$					

NOTES:
f = fluence ÷ 10^{19} n/cm^2; All values taken from Section 6.0 of Reference 5.
FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$

∴ Chemistry Factor for the Lower Shell Forging based on surveillance capsule data = 18.8°F

∴ Chemistry Factor for the Weld Metal based on surveillance capsule data = 20.6°F

TABLE 4

PTS Calculations for Braidwood Unit 1

Material	Cu%	Ni%	CF	f	FF	I	M	ΔRT_{PTS}	RT_{PTS}
4.234 EFPY									
Inter. Shell Forging	0.05	0.73	31.0	0.2963	0.6671	-30	20.68	20.68	11.4
Lower Shell Forging	0.04	0.74	26.0	0.2963	0.6671	-20	17.34	17.34	14.7
using S/C data			18.8	0.2963	0.6671	-20	12.54	12.54	5.1
Weld Metal	0.03	0.67	41.0	0.2963	0.6671	40	27.35	27.35	94.7
using S/C data			20.6	0.2963	0.6671	40	13.74	13.74	67.5
32 EFPY									
Inter. Shell Forging	0.05	0.73	31.0	2.239	1.218	-30	34	37.77	41.8
Lower Shell Forging	0.04	0.74	26.0	2.239	1.218	-20	31.68	31.68	43.4
using S/C data			18.8	2.239	1.218	-20	17	22.90	19.8
Weld Metal	0.03	0.67	41.0	2.239	1.218	40	49.95	49.95	139.9
using S/C data			20.6	2.239	1.218	40	25.10	25.10	90.2
48 EFPY									
Inter. Shell Forging	0.05	0.73	31.0	3.358	1.317	-30	34	40.83	44.8
Lower Shell Forging	0.04	0.74	26.0	3.358	1.317	-20	34	34.25	48.3
using S/C data			18.8	3.358	1.317	-20	17	24.76	21.8
Weld Metal	0.03	0.67	41.0	3.358	1.317	40	54.00	54.00	148.0
using S/C data			20.6	3.358	1.317	40	27.13	27.13	94.3

NOTE:

Initial RT_{NDT} values are measured values.

All of the beltline material in the Braidwood Unit 1 reactor vessel is below the screening criteria at 32 and 48 EFPY. The following plot of RT_{PTS} versus fluence (Figure 2) also illustrates the available margin, where $RT_{PTS} = I + M + \Delta RT_{PTS}$.

SECTION 7.0
CONCLUSIONS

As shown in Table 4, all RT_{PTS} values remain below the NRC PTS screening criteria values using fluence values for the present time (4.234 EFPY), 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 Braidwood Unit 1 reactor vessel beltline region, the weld metal WF-562.

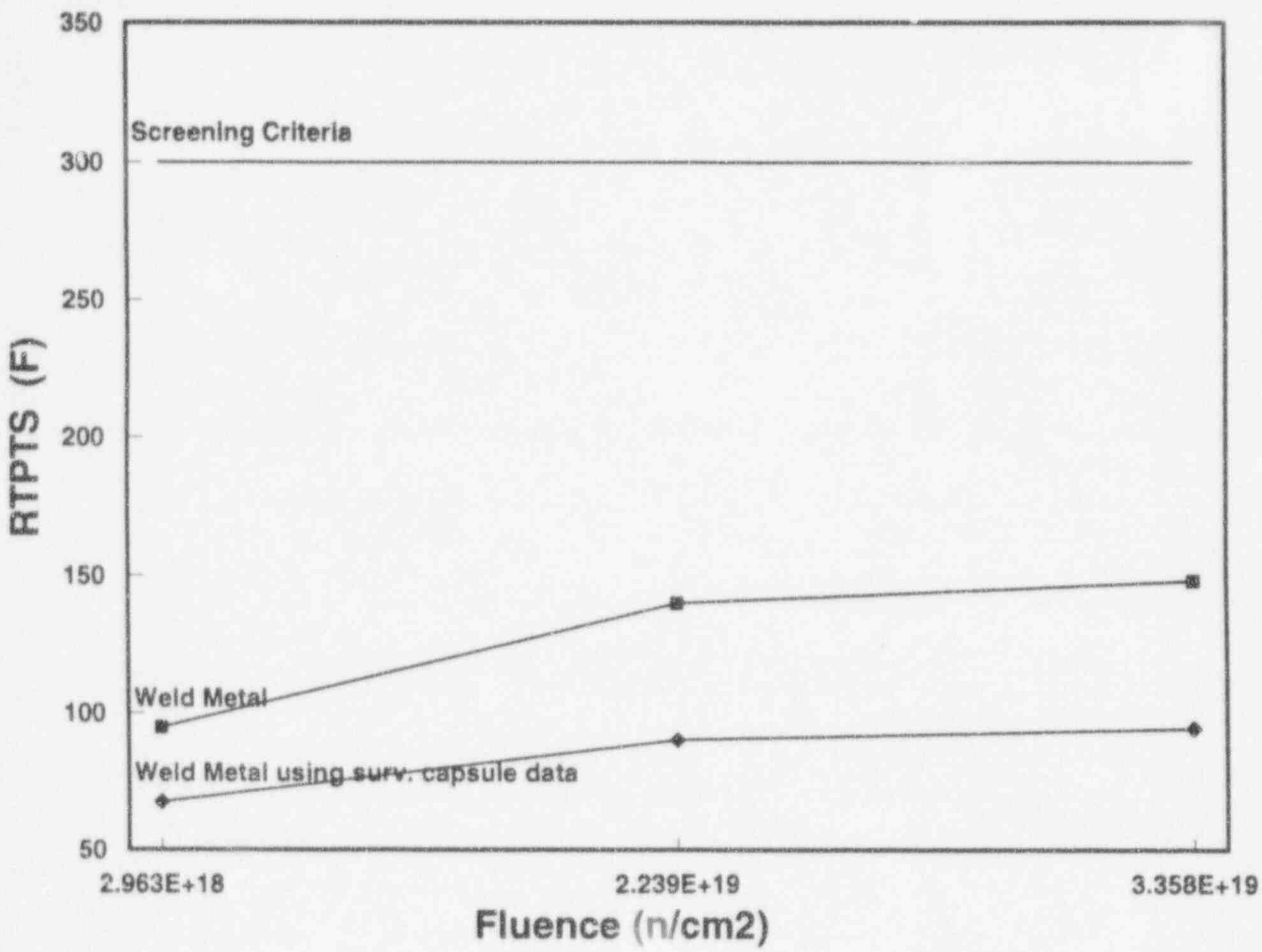

Figure 2 RT_{PTS} versus Fluence Curves for Braidwood Unit 1 Limiting Material - Weld Metal

SECTION 8.0

REFERENCES

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

[2] 10 CFR 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-9807, "Commonwealth Edison Company Braidwood Station Unit No. 1 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko, et al., February 1981.

[5] WCAP-14241, "Analysis of Capsule X from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program", P. A. Peter, et al., December 1994.

[6] WCAP-13070, "Evaluation of Pressurized Thermal Shock for Braidwood Units 1 & 2", M. A. Ramirez, et al., September 1991.