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1400 Opus Place
Downers Grove, Illinois 60515

June 13, 1994

Mr. William Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Supplemental Information Relating to the Request to Amend Technical Specification Sections 3.4.9.1 and 3.4.9.3

Braidwood Station Units 1 and 2
NPF-72/77; NRC Docket Nos. 50-456/457

Reference: D. Saccomando letter to W. Russell dated March 30, 1994, transmitting request to amend Technical Specification Sections 3.4.9.1 and 3.4.9.3

Dear Mr. Russell:

The reference letter transmitted Commonwealth Edison Company Station's request to amend Sections 3.4.9.1 and 3.4.9.3 of the Technical Specifications for Braidwood Unit 1 and 2.

Attached you will find additional information that may be useful when reviewing this amendment request. In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on pressurized thermal shock (PTS). All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through end-of-life. The NRC 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 1991. As a result of the revised rule, Braidwood completed their evaluation entitled, "Evaluation of Pressurized Thermal Shock For Braidwood Units 1&2." Please find that evaluation attached.

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If you have any questions concerning this correspondence please contact this office.

Sincerely,

Denise M. Saccomando
Nuclear Licensing Administrator

Attachment

cc: R. P. Assa, Braidwood Project Manager - NRR
S. G. Dupont, Senior Resident Inspector - Braidwood
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

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WCAP-13070

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR BRAIDWOOD UNITS 1 & 2

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1. 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;
- the presence of a critical-size defect in the vessel wall.

Fracture mechanics analysis can be used to evaluate reactor vessel integrity under severe transient conditions.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on pressurized thermal shock. It established screening criterion 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 and plates and for beltline circumferential weld seams for end-of-life 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-life. The Nuclear Regulatory Commission 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 reference temperatures for pressurized thermal shock (RT_{PTS}) values for the Braidwood Units 1 & 2 reactor vessels to address the Pressurized Thermal Shock (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 both Braidwood Units 1 & 2. 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. 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 values.

The Rule outlines regulations to address the potential for PTS events on pressurized water reactor (PWR) 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 produce 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 by 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 should be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance report, or 5 years from the effective date of this Rule, 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),
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 Criterions for the reactor vessel beltline region is

  270°F for plates, forgings, axial welds
  300°F for circumferential weld materials

* The following equations should 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 Criterion, including analyses of alternatives to minimize the PTS concern.

* NRC approval for operation beyond the Screening Criterion is required.

3. 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 Criterion, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region as given below.

$RT_{PTS} = I + M + \Delta RT_{PTS}$, where $\Delta RT_{PTS} = (CF) f^{(0.28-0.10 \log f)}$

I = Initial reference temperature (RT_{NDT}) 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.

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

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

4. 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." Figures 1 & 2 identify and indicate the location of all beltline region materials for the Braidwood Units 1 & 2 reactor vessels, respectively.

Material property values were derived from vessel fabrication test certificate results. Fast neutron irradiation-induced changes in the tension, fracture and impact properties of reactor vessel materials are

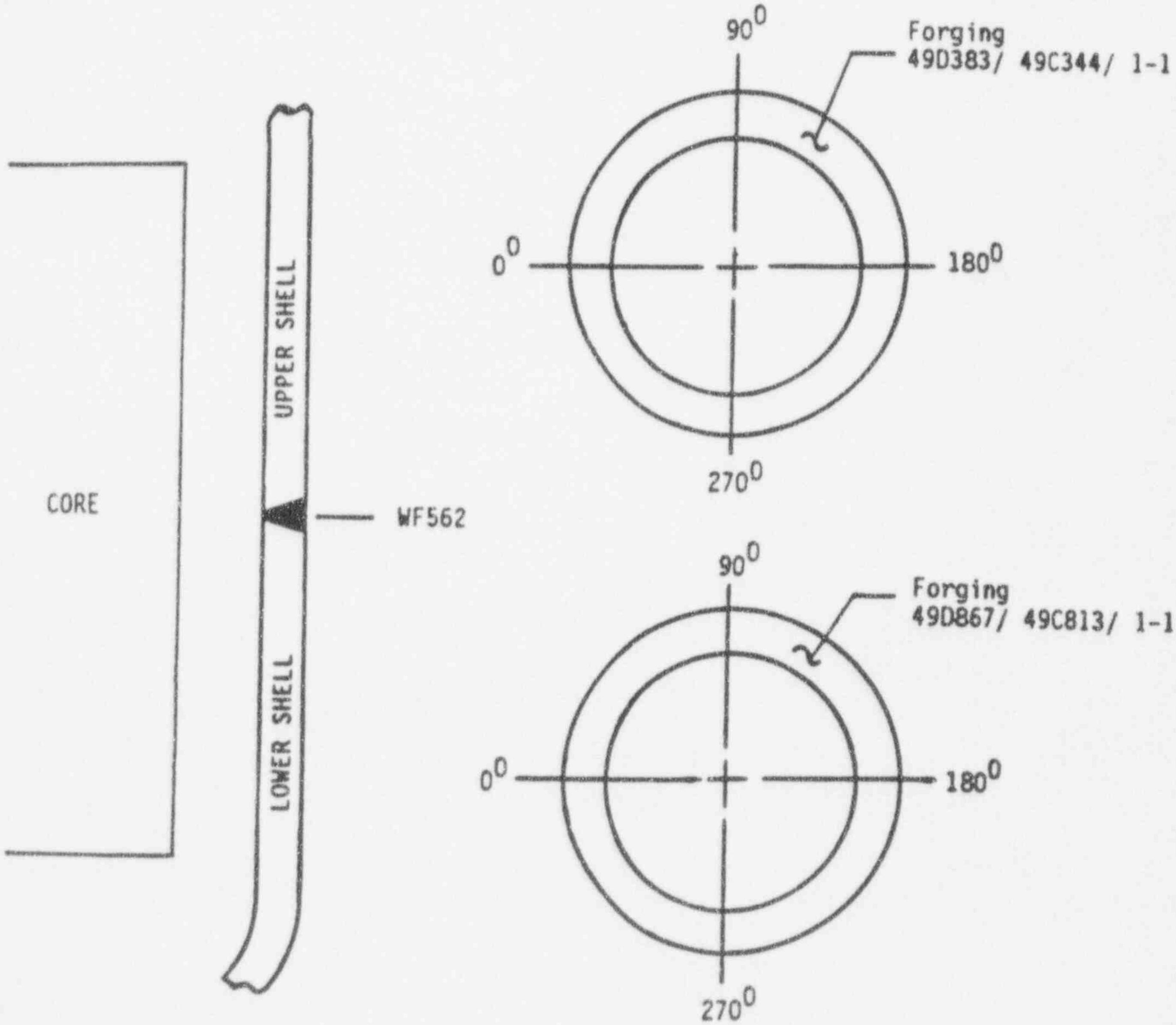

Figure 1. Identification and Location of Beltline Region Materials for the Braidwood Unit 1 Reactor Vessel

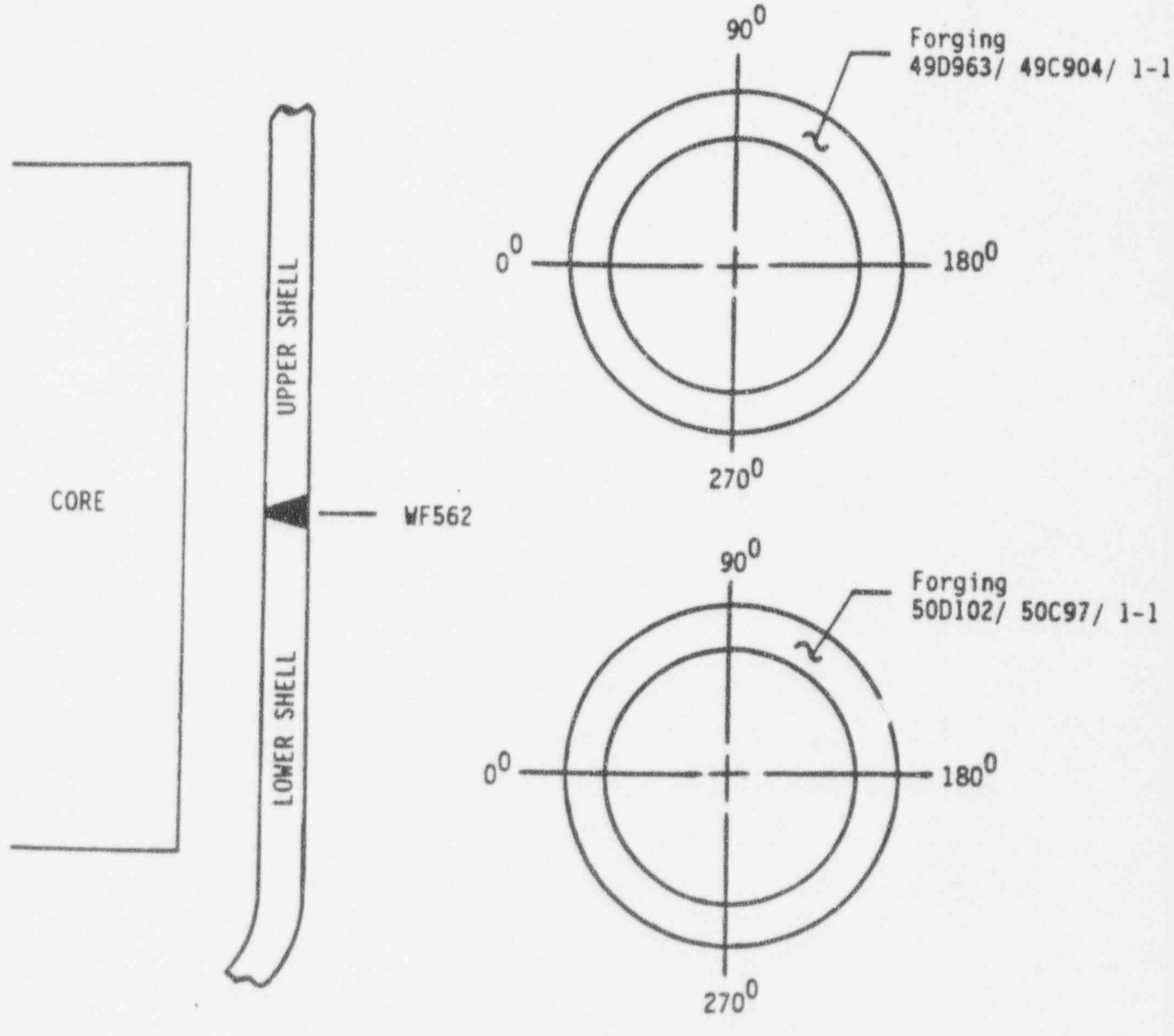

Figure 2. Identification and Location of Beltline Region Materials for the Braidwood Unit 2 Reactor Vessel

largely dependent on chemical composition, particularly in the copper concentration. The variability in irradiation-induced property changes, which exists in general, is compounded by the variability of copper concentration with the weldments.

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the Braidwood Units 1 & 2 reactor vessels are given in Tables 1 & 2, respectively. All of the initial RT_{NDT} values (I-RTNDT) are also presented in these Tables.

TABLE 1

BRAIDWOOD UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Upper Shell, forging 49D383/ 49C344/ 1-1	0.05	0.73	-30
Lower Shell, forging 49D867/ 49C813/ 1-1	0.03	0.73	-20
Circumferential Weld WF562	0.04	0.67	40

TABLE 2

BRAIDWOOD UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Upper Shell, forging 49D963/ 49C904/ 1-1	0.03	0.71	-30
Lower Shell, forging 50D102/ 50C97/ 1-1	0.06	0.75	-30
Circumferential Weld WF562	0.04	0.67	40

5. NEUTRON FLUENCE VALUES

The projected fluence values used for the evaluation of RT_{PTS} are provided in Tables 3 & 4 for both Braidwood Units 1 & 2, respectively. These values are extracted from the latest surveillance capsule U, withdrawn from the reactor vessel [4],[5]. For the evaluation of RT_{PTS}, peak fluence values are used. The peak fluence values are at 25° azimuthal angle for both reactor vessels.

TABLE 3
BRAIDWOOD UNIT 1 FLUENCE VALUES USED FOR RT_{PTS} EVALUATION

Material Description	Fluence 32 EFPY (10^{19}n/cm^2)	Fluence 48 EFPY (10^{19}n/cm^2)
Upper Shell, forging 49D383/ 49C344/ 1-1	3.03	3.615
Lower Shell, forging 49D867/ 49C813/ 1-1	3.03	3.615
Circumferential Weld WF562	3.03	3.615

TABLE 4
BRAIDWOOD UNIT 2 FLUENCE VALUES USED FOR RT_{PTS} EVALUATION

Material Description	Fluence 32 EFPY (10^{19}n/cm^2)	Fluence 48 EFPY (10^{19}n/cm^2)
Upper Shell, forging 49D963/ 49C904/ 1-1	3.03	3.614
Lower Shell, forging 50D102/ 50C97/ 1-1	3.03	3.614
Circumferential Weld WF562	3.03	3.614

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 region materials in the Braidwood Units 1 & 2 reactor vessels, up to the end-of-life (32 EFPY) and an extended life (48 EFPY). The applicable fluence values of both Braidwood Units are shown in Tables 3 & 4.

Tables 5 & 6 provide a summary of the RT_{PTS} values for all beltline region materials of Braidwood Units 1 & 2 respectively, for the end-of-life (32 EFPY) and the extended life (48 EFPY) using the PTS Rule.

The PTS Rule requires that each plant assess the RT_{PTS} values based on plant specific surveillance capsule data under certain conditions. These conditions are:

- 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 Braidwood Units 1 & 2, plant specific surveillance capsule data was not used because no credible surveillance data is available. Only one capsule has been withdrawn from each Braidwood Unit reactor vessel, hence it can not be used in the calculation of the chemistry factors.

TABLE 5

RT_{PTS} VALUES FOR BRAIDWOOD UNIT 1

Material Description	Screening Criteria (°F)	32 EFPY (°F)	48 EFPY (°F)
Upper shell, forging 49D383/ 49C344/ 1-1	270	44	45
Lower Shell, forging 49D867/ 49C813/ 1-1	270	40	41
Circumferential Weld WF 562	300	166	168

TABLE 6

RT_{PTS} VALUES FOR BRAIDWOOD UNIT 2

Material Description	Screening Criteria (°F)	32 EFPY (°F)	48 EFPY (°F)
Upper shell, forging 49D963/ 49C904/ 1-1	270	30	31
Lower Shell, forging 50D102/ 50C978/ 1-1	270	52	53
Circumferential Weld WF 562	300	166	168

7. CONCLUSIONS

As shown in Tables 5 & 6, all the RT_{PTS} values remain below the NRC screening values for PTS using the projected fluence values for both the end-of-life (32 EFPY) and the extended life up to 48 EFPY. The highest RT_{PTS} values for 32 EFPY and 48 EFPY for Braidwood Unit 1 are 166°F and 168°F, respectively, in the circumferential weld WF562. Similarly, the highest RT_{PTS} values for 32 EFPY and 48 EFPY for Braidwood Unit 2 are also 156°F and 168°F, respectively, in the circumferential weld WF562. The plots of the RT_{PTS} values versus the fluence are shown in Figures 3 & 4 for the beltline region materials in the Braidwood Units 1 & 2 reactor vessels, respectively. These plots indicate the three materials in the beltline region in Braidwood Unit 1 & 2 reactor vessels are not expected to exceed the screening criteria or to create any PTS concern, provided the fluence projections and the material chemistry (based on Reference 2 Table and Surveillance Capsule data) of the beltline regions in each reactor vessel does not change significantly.

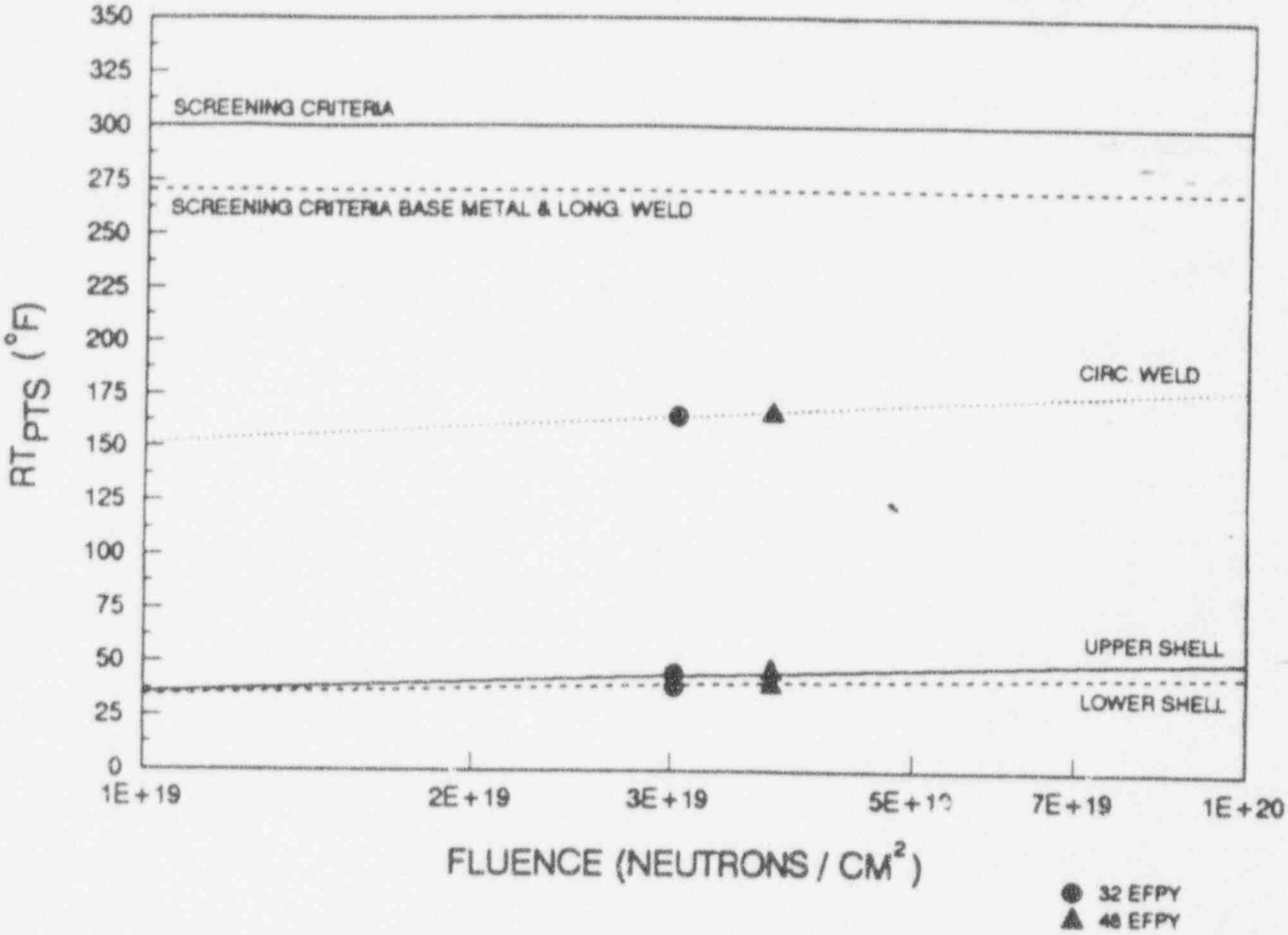

Figure 3. RT$_{PTS}$ versus Fluence Curves for Braidwood Unit 1 Beltline Region Materials.

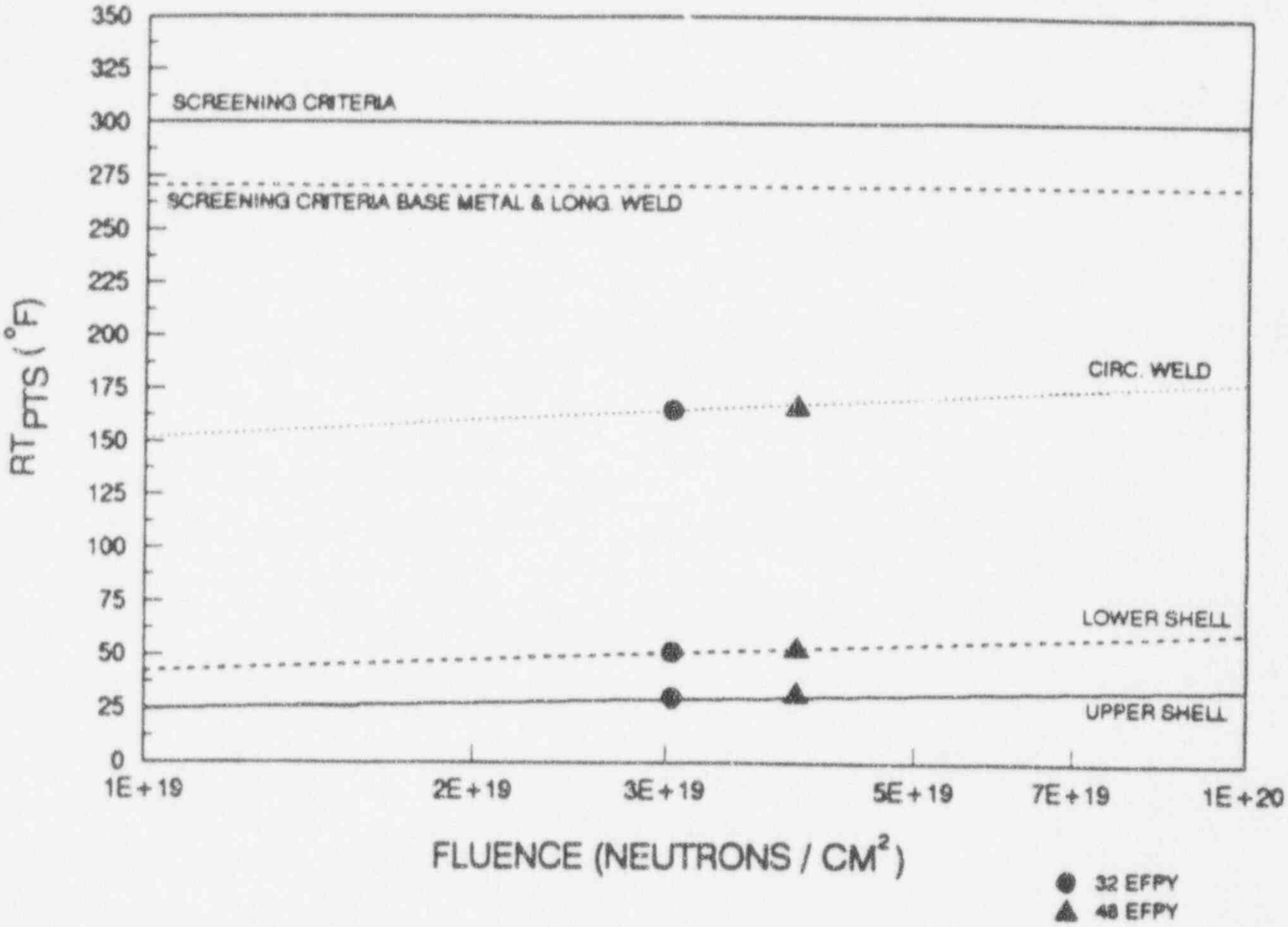

Figure 4. RT_{PTS} versus Fluence Curves for Braidwood Unit 2 Beltline Region Materials.

8. REFERENCES

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

[2] 10CFR Part 50, "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-12685, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," E. Terek, et al., August 1990.

[5] WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," E. Terek, et al., March 1991.