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WCAP-14345

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR THE COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 2

P. A. Peter

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Prepared by Westinghouse Electric Corporation
for the Texas Utilities Electric Company

Approved by: [signature]
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:

E. Terek ______________________

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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. 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 Texas Utilities Electric Company (TU Electric) Comanche Peak Steam Electric Station (CPSES) Unit 2 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 CPSES 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 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 five 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} 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 three 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.

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

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 $0.5 * \Delta RT_{PTS}$

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

f = Neutron fluence (10^{19} n/cm^2, E>1.0 MeV) at the clad/base metal interface

CF = Chemistry Factor in °F from the tables[2] for welds and base metals (plates or forgings). If plant-specific surveillance data from two or more surveillance capsules has been deemed credible per Regulatory Guide 1.99, Revision 2, it should 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 CPSES Unit 2 vessel beltline region was performed. The beltline is defined by ASTM E185-82[4] to be "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of the surveillance material". Figure 1 identifies and indicates the location of all beltline region material for the CPSES Unit 2 reactor vessel.

Material property values along with 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 plates and weld materials of the CPSES Unit 2 reactor vessel are given in Table 1.

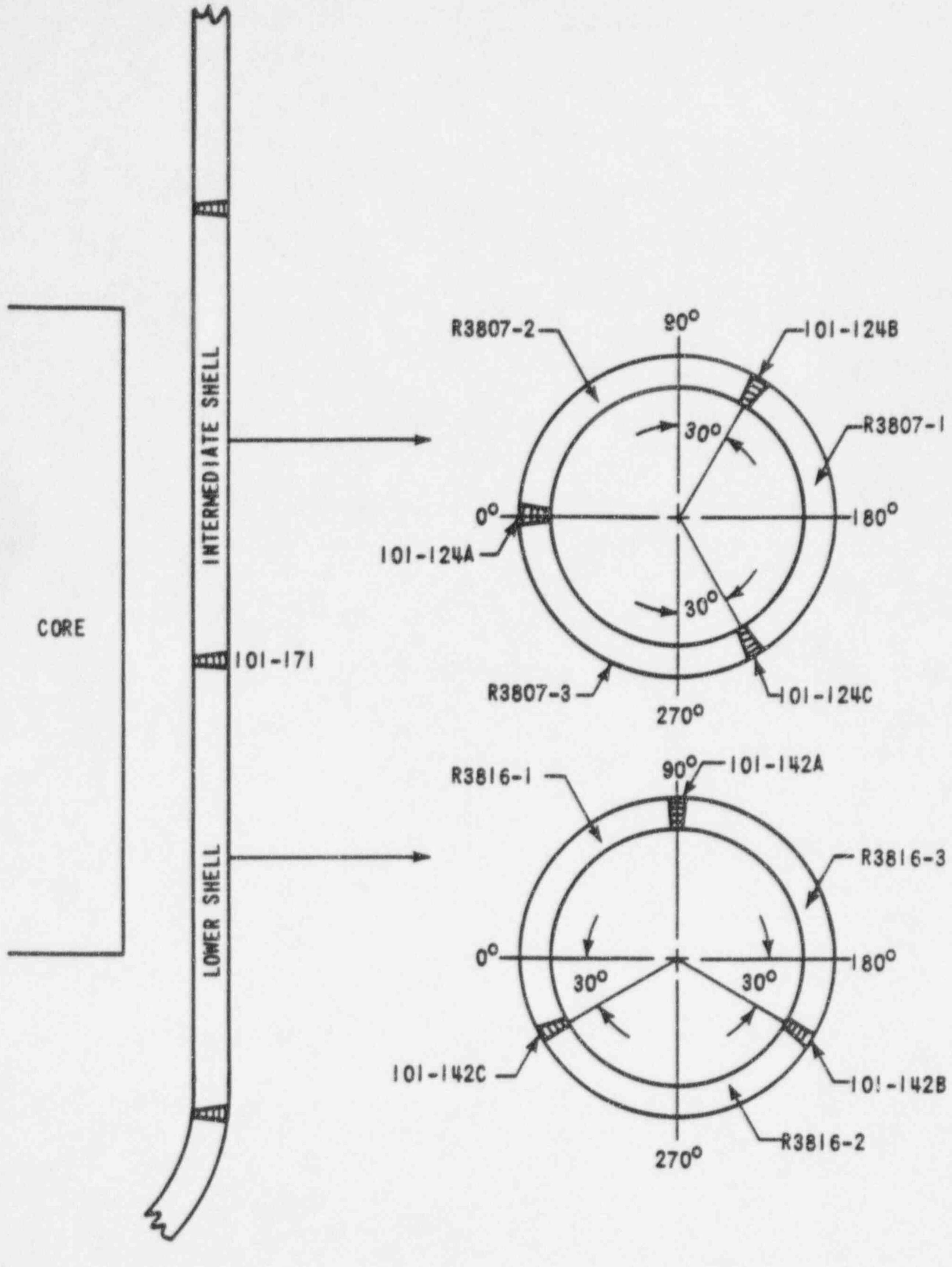

Figure 1 Identification and Location of Beltline Region Material for the Comanche Peak Unit 2 Reactor Vessel[6]

TABLE 1

CPSES Unit 2 Reactor Vessel Beltline Region Material Properties Used in Calculations

Material	Cu%[a]	Ni%[a]	Chemistry Factor[b]	Initial RT_{NDT}[c]
Intermediate Shell Plate R3807-1	0.06	0.64	37.0	-20
Intermediate Shell Plate R3807-2	0.0625	0.63	38.8	10
Intermediate Shell Plate R3807-3	0.05	0.60	31.0	-20
Lower Shell Plate R3816-1	0.05	0.59	31.0	-30
Lower Shell Plate R3816-2	0.03	0.65	20.0	0
Lower Shell Plate R3816-3	0.04	0.63	26.0	-40
Circumferential Weld 101-171	0.04	0.051	28.8	-60
Inter. & Lower Shell Longitudinal Welds	0.06	0.065	36.5	-50

NOTES:

(a) Average Cu and Ni weight percent values from WCAP-10684, Tables A-1 through A-3. The average Cu and Ni values are reported as integers in order to determine the most accurate Chemistry Factor values possible.

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

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

SECTION 5.0
NEUTRON FLUENCE VALUES

The calculated peak fast neutron fluence (E>1.0 MeV) values at the inner surface of the CPSES Unit 2 reactor vessel are shown in Table 2. These values were projected using the results of the surveillance Capsule U radiation analysis[7]. The RT_{PTS} calculations for the base metal and circumferential weld were performed using the peak fluence values, which occur at the 45° azimuthal angle. The fluence values at the 0° and 30° azimuthal angles were used for the RT_{PTS} calculations for the longitudinal welds of the CPSES Unit 2 reactor vessel. (Additional details on the basis for the fluence projections can be found in Section 6.0 of Reference 7.)

TABLE 2

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

Time	0° Long. Welds 101-124A & 101-142A	30° Long Welds 101-124B/C & 101-142B/C	45° Base Metal Plates & Circ. Weld 101-171
0.904 EFPY	3.919×10^{17}	7.195×10^{17}	8.011×10^{17}
32 EFPY	1.387×10^{19}	2.548×10^{19}	2.836×10^{19}
48 EFPY	2.081×10^{19}	3.821×10^{19}	4.255×10^{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 CPSES Unit 2 reactor vessel for fluence values at the present time (0.904 EFPY per Capsule U 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.)

Plant specific surveillance capsule data is not provided since Capsule U is the first capsule to be removed from the CPSES Unit 2 reactor vessel. Per Regulatory Guide 1.99, Revision 2, Position 2.1, when *two* or more credible sets of surveillance data become available from the reactor in question, they may be used in the calculation of the chemistry factors. (Furthermore, if the surveillance capsule data gives a higher value of adjusted reference temperature (or RT_{PTS}) than that given by using the procedure of Position 1.1, the surveillance data should be used. If this procedure gives a lower value, either may be used.)

Therefore, the chemistry factors for CPSES Unit 2 were calculated using only Tables 1 and 2 from 10 CFR 50.61[2]. Tables 3 through 5 provide a summary of the RT_{PTS} values for all beltline region material for 0.904, 32, and 48 EFPY, respectively.

TABLE 3

PTS Calculations for CPSES Unit 2 at 0.904 EFPY

Material	CF	f	FF	I	M	ΔRT_{PTS}	RT_{PTS}	Screening Criteria
0.904 EFPY								
Inter. Shell Plate R3807-1	37.0	0.08011	0.3740	-20	13.8	13.8	8	270
Inter. Shell Plate R3807-2	38.8	0.08011	0.3740	10	14.5	14.5	39	270
Inter. Shell Plate R3807-3	31.0	0.08011	0.3740	-20	11.6	11.6	3	270
Lower Shell Plate R3816-1	31.0	0.08011	0.3740	-30	11.6	11.6	-7	270
Lower Shell Plate R3816-2	20.0	0.08011	0.3740	0	7.5	7.5	15	270
Lower Shell Plate R3816-3	26.0	0.08011	0.3740	-40	9.7	9.7	-21	270
Circ. Weld 101-171	28.8	0.08011	0.3740	-60	10.8	10.8	-38	300
Inter Long Weld 101-124A	36.5	0.03919	0.2560	-50	9.3	9.3	-31	270
Inter Long Weld 101-124B/C	36.5	0.07195	0.3543	-50	12.9	12.9	-24	270
Lower Long Weld 101-142A	36.5	0.03919	0.2560	-50	9.3	9.3	-31	270
Lower Long Weld 101-142B/C	36.5	0.07195	0.3543	-50	12.9	12.9	-24	270

NOTE: Initial RT_{NDT} values are measured values.

TABLE 4

PTS Calculations for CPSES Unit 2 at 32 EFPY

Material	CF	f	FF	I	M	ΔRT_{PTS}	RT_{PTS}	Screening Criteria
32 EFPY								
Inter. Shell Plate R3807-1	37.0	2.836	1.277	-20	34	47.3	61	270
Inter. Shell Plate R3807-2	38.8	2.836	1.277	10	34	49.6	94	270
Inter. Shell Plate R3807-3	31.0	2.836	1.277	-20	34	39.6	54	270
Lower Shell Plate R3816-1	31.0	2.836	1.277	-30	34	39.6	44	270
Lower Shell Plate R3816-2	20.0	2.836	1.277	0	25.5	25.5	51	270
Lower Shell Plate R3816-3	26.0	2.836	1.277	-40	33.2	33.2	26	270
Circ. Weld 101-171	28.8	2.836	1.277	-60	36.8	36.8	14	300
Inter Long Weld 101-124A	36.5	1.387	1.091	-50	39.8	39.8	30	270
Inter Long Weld 101-124B/C	36.5	2.548	1.251	-50	45.7	45.7	41	270
Lower Long Weld 101-142A	36.5	1.387	1.091	-50	39.8	39.8	30	270
Lower Long Weld 101-142B/C	36.5	2.548	1.251	-50	45.7	45.7	41	270

NOTE: Initial RT_{NDT} values are measured values.

TABLE 5

PTS Calculations for CPSES Unit 2 at 48 EFPY

Material	CF	f	FF	I	M	ΔRT_{PTS}	RT_{PTS}	Screening Criteria
48 EFPY								
Inter. Shell Plate R3807-1	37.0	4.255	1.369	-20	34	50.7	65	270
Inter. Shell Plate R3807-2	38.8	4.255	1.369	10	34	53.1	97	270
Inter. Shell Plate R3807-3	31.0	4.255	1.369	-20	34	42.5	57	270
Lower Shell Plate R3816-1	31.0	4.255	1.369	-30	34	42.5	47	270
Lower Shell Plate R3816-2	20.0	4.255	1.369	0	27.4	27.4	55	270
Lower Shell Plate R3816-3	26.0	4.255	1.369	-40	34	35.6	30	270
Circ. Weld 101-171	28.8	4.255	1.369	-60	39.4	39.4	19	300
Inter Long Weld 101-124A	36.5	2.081	1.199	-50	43.8	43.8	38	270
Inter Long Weld 101-124B/C	36.5	3.821	1.346	-50	49.1	49.1	48	270
Lower Long Weld 101-142A	36.5	2.081	1.199	-50	43.8	43.8	38	270
Lower Long Weld 101-142B/C	36.5	3.821	1.346	-50	49.1	49.1	48	270

NOTE: Initial RT_{NDT} values are measured values.

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

SECTION 7.0

CONCLUSIONS

As shown in Tables 3, 4, and 5, all RT_{PTS} values remain below the NRC PTS screening criteria values using fluence values for the present time (0.904 EFPY), end-of-license (32 EFPY), and 48 EFPY. A plot of the RT_{PTS} versus EFPY, shown in Figure 2, illustrates the available margin for the most limiting material in the CPSES Unit 2 reactor vessel beltline region, Intermediate Shell Plate R3807-2.

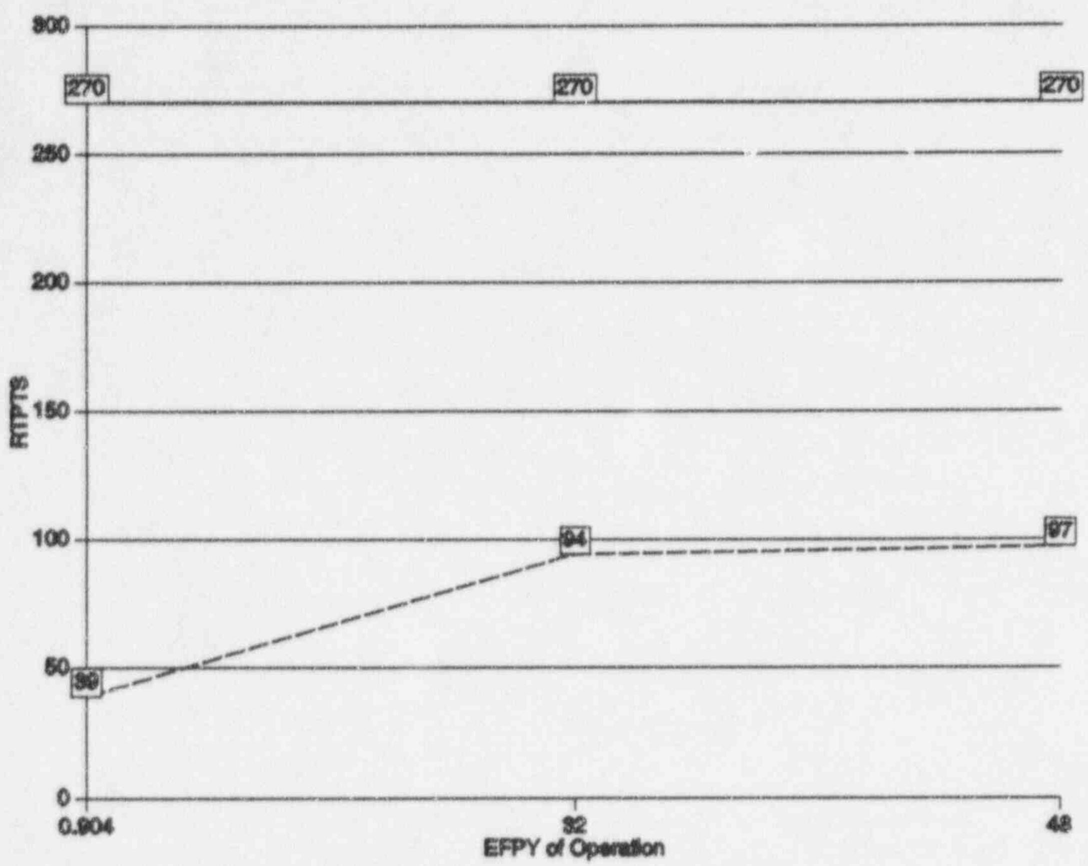

Figure 2 RT_{PTS} versus EFPY of Operation for CPSES Unit 2 Limiting Material - Intermediate Shell Plate R3807-2

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.

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

[4] ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", E706 (1F).

[5] WCAP-10684, "Texas Utilities Generating Company Comanche Peak Unit No. 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, October 1984.

[6] Comanche Peak S.E.S. Final Safety Analysis Report Units 1 and 2, Figure 5.3-1B, "Unit No. 2 Reactor Vessel Beltline Region Material Identification", Amendment 8, November 30, 1979.

[7] WCAP-14315, "Analysis of Capsule U from the Texas Utilities Electric Company Comanche Peak Unit No. 2 Reactor Vessel Radiation Surveillance Program", R. Auerswald, et al., July 1995.