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EVALUATION OF PRESSURIZED THERMAL SHOCK FOR MCGUIRE UNIT 2

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PREFACE

This report nas been technically reviewed and verified.

Reviewer: M. A. Ramirez [signature]

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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; 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 McGuire 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 McGuire 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. 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. 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.

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

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.

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

CF = Chemistry factor in °F 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." Figure 1 identifies and indicates the location of all beltline region materials for the McGuire 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 the available material chemistry information.

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

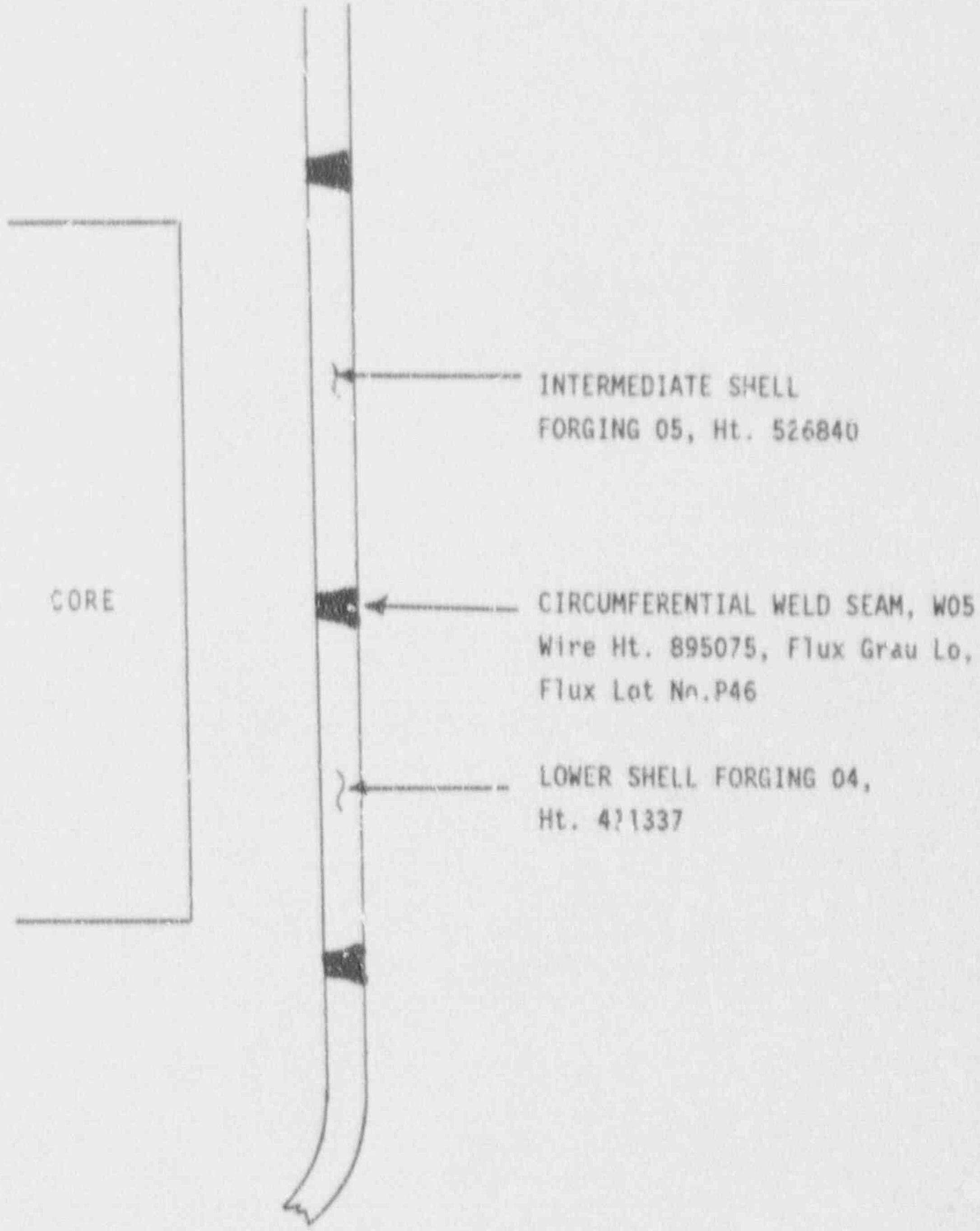

Figure 1. Identification and Location of Beltline Region Materials for the McGuire Unit 2 Reactor Vessel

TABLE 1

MCGUIRE UNIT 2 REACTOR VESSEL

BELTLINE REGION MATERIAL PROPERTIES

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Intermediate Shell Forging 05*	0.153	0.793	- 4
Lower Shell Forging 04	0.15	0.88	-30
Circumferential Weld*	0.039	0.73	-68

* Mean values of copper and nickel as indicated below

Material	Data Source	Copper (wt. %)	Nickel (wt. %)
Forging 05	Original Mill Test Report	0.16	0.85
	Surveillance Program [6]	0.16	0.79
	Capsule V Report [7]	0.14	0.71
	Capsule U Report [4]	0.151	0.82
	Mean Average Value	0.153	0.793
Forging 04	Original Mill Test Report	0.15	0.88
Weld	Original Mill Test Report	0.05	0.70
	Surveillance Program [6]	0.031	0.73
	Capsule V Report [7]	0.03	0.66
	Capsule U Report [4]	0.039	0.765
	Capsule U Report [4]	0.036	0.747
	Capsule U Report [4]	0.045	0.776
	Mean Average Value	0.039	0.730

5. NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E>1 MeV) at the inner surface of the McGuire Unit 2 reactor vessel is shown in Table 2. These values were projected using the results of the Capsule U radiation surveillance program[4]. The RT_{PTS} calculations were performed using the peak fluence value, which occurs at the 45° azimuth in the McGuire Unit 2 reactor vessel.

TABLE 2

NEUTRON EXPOSURE PROJECTIONS* AT KEY LOCATIONS ON THE MCGUIRE UNIT 2 PRESSURE VESSEL CLAD/BASE METAL INTERFACE FOR 6.05 AND 32 EFPY[4]

EFPY	0°	15°	30°	45°
6.05	0.252	0.376	0.279	0.385
32	1.33	1.99	1.48	2.04

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

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 of the McGuire Unit 2 reactor vessel as a function of present time (6.05 EFPY per Capsule U analysis) and end-of-life (32 EFPY) fluence values. The fluence data were generated based on the most recent surveillance capsule program results[4].

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 McGuire Unit 2, the use of plant specific surveillance capsule data arises for the intermediate shell forging 05 and circumferential weld because of the following reasons:

1) There have been three 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 forging and weld materials.

The chemistry factors for the intermediate shell forging 05 and circumferential weld were calculated using the surveillance capsule data as shown in Table 3. The chemistry factor value for the lower shell forging 04 was calculated using the Table 2 from 10 CFR 50.61[2].

TABLE 3

CALCULATION OF CHEMISTRY FACTORS USING MCGUIRE UNIT 2 SURVEILLANCE CAPSULE DATA[4]

Component	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	(FF)^2
FORGING 05 (Long) TANG	V	0.337	0.701	65	45.534	0.491
	X	1.45	1.103	100	110.301	1.217
	U	2.02	1.192	90	107.255	1.420
FORGING 05 (Trans) AXIAL	V	0.337	0.701	70	49.037	0.491
	X	1.45	1.103	105	115.816	1.217
	U	2.02	1.192	85	101.296	1.420
					529.239	6.255
	Chemistry Factor = 529.239 /6.255 =				84.61	
Weld Metal	V	0.337	0.701	45	31.524	0.491
	X	1.45	1.103	35	38.605	1.217
	U	2.02	1.192	20	23.834	1.420
					93.963	3.128
	Chemistry Factor = 93.963 /3.128 =				30.04	

Tables 4 and 5 provide a summary of the RT_{PTS} values for all beltline region materials for 6.05 EFPY and end-of-license (32 EFPY), respectively, using the PTS Rule.

TABLE 4

RT_{PTS} VALUES FOR MCGUIRE UNIT 2 FOR 6.05 EFPY

Material	ΔRT_{NDT}(°F) (CF	x FF*)	+ Initial RT_{NDT} (°F)	+ Margin (°F)	= RT_{PTS} (°F)
Intermediate Shell	117.2	0.7358	- 4	34	116
Forging 05	(84.6)	0.7[illegible]	- 4	34	(92)
Lower Shell Forging 04	115.8	0.7358	-30	34	89
Circumferential	52.7	0.7358	-68	56	27
Weld Seam	(30)	0.7358	-68	56	(10)

() Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 3.85×10^{18} n/cm^2[4].

TABLE 5

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

Material	ΔRT_{NDT}(°F) (CF	x FF*)	+ Initial RT_{NDT} (°F)	+ Margin (°F)	= RT_{PTS} (°F)
Intermediate Shell	117.2	1.1942	- 4	34	170
Forging 05	(84.6)	1.1942	- 4	34	(131)
Lower Shell	115.8	1.1942	-30	34	142
Forging 04					
Circumferential	52.7	1.1942	-68	56	51
Weld Seam	(30)	1.1942	-68	56	(24)

() Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 2.04×10^{19} n/cm^2[4].

7. CONCLUSIONS

As shown in Tables 4 and 5, all the RT_{PTS} values remain below the NRC screening values for PTS using the fluence values for the present time (6.05 EFPY) and the projected fluence values for the end-of-license (32 EFPY). A plot of the RT_{PTS} values versus the fluence is shown in Figure 2 for the most limiting material, the intermediate shell forging 05, in the McGuire Unit 2 reactor vessel beltline region.

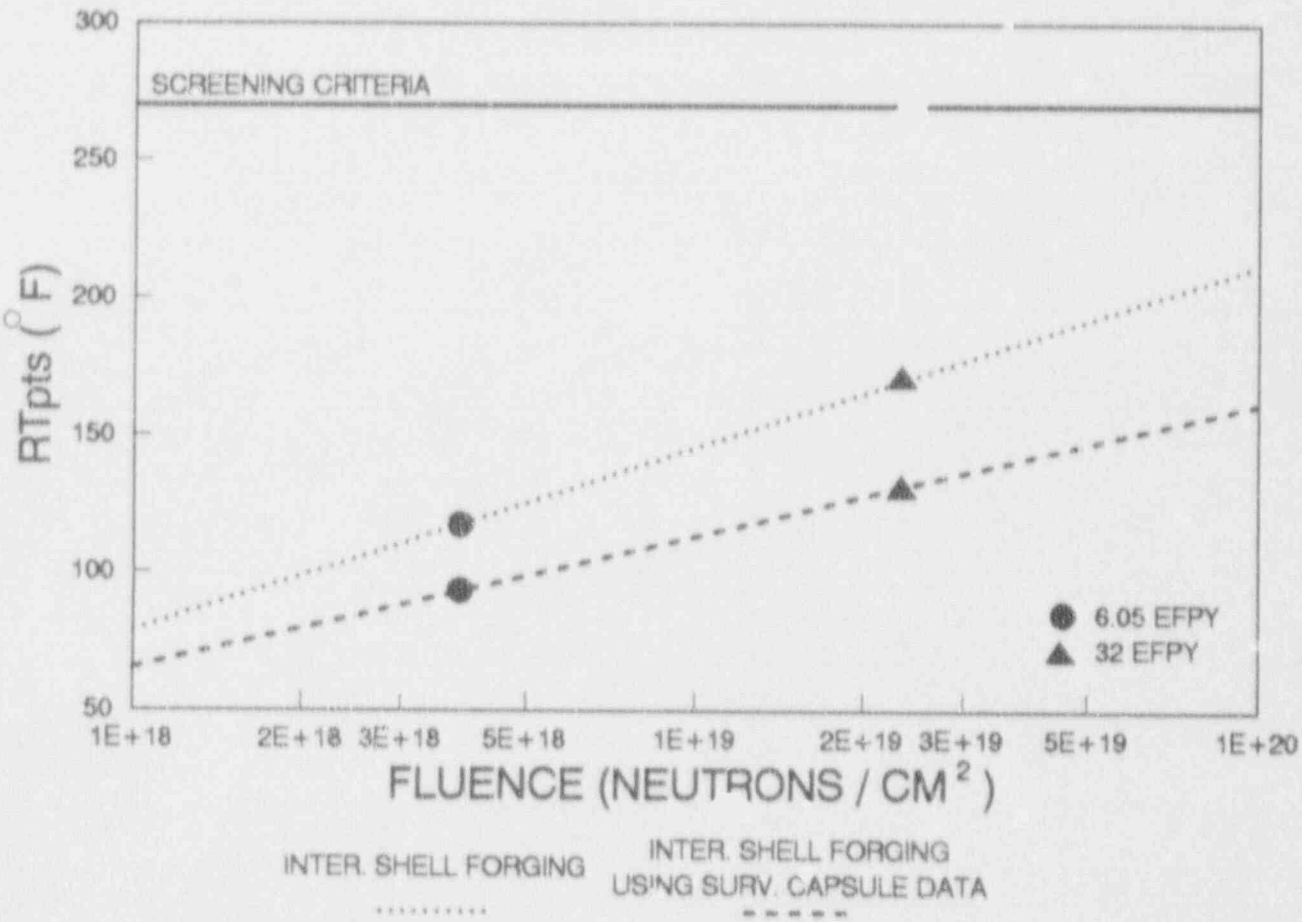

Figure 2. RT_{PTS} versus Fluence Curves for McGuire Unit 2 Limiting Material - Intermediate Shell Forging 05

8. 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-13516, "Analysis of Capsule U from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," J. M. Chicots, et al., October 1992. (Westinghouse Proprietary Class 3)

[5] WCAP-12556, "Analysis of Capsule X from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," E. Terek, et al., April 1990. (Westinghouse Proprietary Class 3)

[6] WCAP-9489, "Duke Power Company William B. McGuire Unit No. 2 Reactor Vessel Radiation Surveillance Program," K. Koyama and J. A. Davidson,May 1979.

[7] WCAP-11029, "Analysis of Capsule V from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program", S. ". Yanichko, et al., January 1986.