Attachment 3

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

WCAP-14638 Revision 3

Evaluation of Pressurized Thermal Shock for Prairie Island Unit 2

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14638, Revision 3

Evaluation of Pressurized Thermal Shock for Prairie Island Unit 2

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PREFACE

Revision 3:

This revision was completed to change the initial RT_{NDT} of the Intermediate Shell Forging and Lower Shell Forging to 14°F and -4°F respectively. This change is a result of a customer request. Only Pages 8 and 13 are affected by this updated information.

Revision 2:

- Added Executive Summary.
- Revised Table 1 and added new Table 2 to add upper shell Cu and Ni content data.
- Renumbered Tables 2, 3, 4, 5 and 6 to Tables 3, 4, 5, 6 and 7.
- Revised Table 3 to add upper shell initial RT_{NDT} data.
- Revised Table 4 to add "top of core" fluence data.
- Revised paragraph (1) in Section 6.0 to indicate that the surveillance capsule data is deemed not credible.
- Revised Table 5 by adding upper shell and revising chemistry factors per RG 1.99, Rev. 2, Position 1.1
- Revised Table 6 to reflect lower shell and RG 1.99, Rev. 2 Position 2.1 including ratio procedure in weld metal calculation.
- Revised Table 7 to add upper shell and update chemistry factors and RT_{PTS} results.

Revision 1:

- Revised format to fit updated WCAP standards.
- Revised Tables 3 and 6 per updated fluences given in reference 5.

Verified By: Ed Terek
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EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Prairie Island Unit No. 2 reactor vessel beltline based upon the results of the Surveillance Capsule P evaluation. However, the surveillance capsule data is deemed to be not credible per Regulatory Guide 1.99, Revision 2 criteria. Since the upper shell to intermediate shell weld seam is located below the top of the active core, the upper shell forging material and the weld material were considered in the evaluation as beltline materials. Since the surveillance capsule data is not credible, the RT_{PTS} values are calculated in accordance with Regulatory Guide 1.99, Revision 2 using the procedures and margins which give the most conservative results. Based upon these conservative results, the limiting forging material in the Prairie Island Unit No. 2 beltline is the lower shell forging with a projected EOL RT_{PTS} value of 110°F using the Prairie Island Unit No. 2 surveillance capsule data. This value is well below the screening criteria of 270°F for plates and forgings in Regulatory Guide 1.99, Revision 2. The limiting weld material in the Prairie Island Unit No. 2 reactor vessel beltline is found in upper to intermediate shell weld seam W2 with an EOL RT_{PTS} value of 143°F using the Prairie Island Unit No. 1 surveillance capsule data (which has surveillance weld material fabricated from the same heat of weld wire as Unit No. 2 weld seam W2). This RT_{PTS} value is well below the screening criteria value of 300°F for circumferential welds at EOL (35 EFPY).

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1.0 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A 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 purpose of this report is to determine the RT_{PTS} values for the Prairie Island Unit 2 reactor vessel using the results of the surveillance Capsule P evaluation. Section 2.0 discusses the PTS 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 Prairie Island 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 conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2.0 PRESSURIZED THERMAL SHOCK

The Nuclear Regulatory Commission (NRC) recently amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The revised PTS Rule[1], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is currently described in Regulatory Guide 1.99, Revision 2[2].

2. The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of life (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each vessel beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:
  270°F for plates, forgings, and axial weld materials, and
  300°F for circumferential weld materials.

3.0 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \quad (1)$$

$RT_{NDT(U)}$ = reference temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2.

$$M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2} \quad (2)$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value

σ_Δ is the standard deviation for ΔRT_{NDT}.

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 one-half of ΔRT_{NDT}.

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \quad (3)$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is given in Table 1 for welds and Table 2 for base metal (plates or forgings) of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the best estimate neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining ΔRT_{PTS}.

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF is determined from Equation 5.

$$CF = \frac{\Sigma[A_i * f_i^{(0.28-0.10\log f_i)}]}{\Sigma[f_i^{(0.56-0.20\log f_i)}]} \tag{5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measure values of ΔRT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

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 Prairie Island Unit 2 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "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 radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". For Prairie Island Unit 2, the upper shell forging and the upper shell to intermediate shell weld seam are included in the beltline region since the weld seam is 6.833 inches below the top of the active fuel stack.

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 Prairie Island Unit 2 surveillance capsule testing program[3]. The average copper and nickel values were calculated for each beltline region material using all of the available material chemistry information as shown in Tables 1 and 2. A summary of the pertinent chemical and mechanical properties of the beltline region forgings and weld material of the Prairie Island Unit 2 reactor vessel is given in Table 3.

Table 1 Calculation of Average Cu and Ni Weight Percent Values for Beltline Region Base Materials

Ref.	Intermediate Shell Forging 22829		Lower Shell Forging 22642[a]		Upper Shell Forging 22231/39088[b]		A533 Gr. B, CL1 Correlation Monitor Material (HSST Plate 02)	
	Cu %	Ni %	Cu %	Ni %	Cu %	Ni %	Cu %	Ni %
3			0.085	0.70			0.14	0.68
4			0.068	0.694				
5			0.068	0.575				
6	0.075	0.75						
6	0.070	0.75						
7			0.085	0.700				
7			0.085	0.700				
8					0.065	0.73		
8					0.065	0.73		
Avg.	0.0725	0.75	0.0782	0.6738	0.065	0.73	0.14	0.68

NOTES:

(a) Surveillance program base metal material.

(b) Forging B is included since it extends 6.833 inches below the top of the active fuel stack.

Table 2 Calculation of the Average Cu and Ni Weight Percent Values for the Prairie Island Unit 2 Weld Materials

Ref.	Surveillance Program Weld Metal[a]		Inter. to Lower Shell Weld Seam W3[a]		Upper to Inter. Shell Weld Seam W2[b]	
	Cu %	Ni %	Cu %	Ni %	Cu %	Ni %
3	0.082	0.072				
4	0.076	0.071				
5	0.094	0.103				
5	0.081	0.087				
5	0.078	0.081				
9			0.090	0.130		
			0.0822[c]	0.0828[c]		
10					0.14	0.14
11					0.14	0.17
12					0.105	0.11
13					0.14[d]	0.11[d]
Average	0.0822	0.0828	0.0861	0.1064	0.13125	0.1325

NOTES:

(a) The Surveillance Program weld metal was fabricated with Weld Wire Type UM40, Heat No. 2721, Flux Type UM89, Lot No. 1263 and is identical to the intermediate to lower shell girth weld seam W3.

(b) Weld seam W2 is being included here, since it is 6.833 inches below the top of the active fuel stack. Weld seam W2 was fabricated with Weld Wire Type UM40, Heat No. 1752, FluxType UM89, Lot No. 1263.

(c) These values are the average of all data points from the Prairie Island Unit 2 surveillance weld which was fabricated with the same heat of weld wire (Ref. 5).

(d) These values are the average of all data points from the Prairie Island Unit 1 surveillance weld which was fabricated with the same heat of weld wire (Ref. 14).

Table 3 Prairie Island Unit 2 Reactor Vessel Beltline Region Material Properties

Material Description	Cu (%) [a]	Ni (%) [a]	$RT_{NDT(U)}$ (°F)
Intermediate Shell Forging 22829	0.07	0.75	14[b]
Lower Shell Forging 22642	0.08	0.67	-4[b]
Upper Shell Forging 22231/39088	0.07	0.73	-13[c]
Inter./Lower Shell Girth Weld	0.09	0.11	-31[b]
Upper/Inter. Shell Girth Weld	0.13	0.13	-13[d]

NOTES:

(a) Average values of copper and nickel are from Tables 1 and 2 and rounded per the procedure given in ASTM E 29.

(b) The $RT_{NDT(U)}$ values for the intermediate and lower shell forgings and the inter./lower shell girth weld are measured values and were obtained from Prairie Island Unit 2 CMTR.

(c) The $RT_{NDT(U)}$ value for the upper shell forging is a measured value obtained from Reference 8.

(d) The $RT_{NDT(U)}$ value for the upper/inter. shell girth weld is a measured value obtained from Reference 14.

5.0 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Prairie Island Unit 2 reactor vessel are shown in Table 4. The core mid-plane and top of core values were projected using the results of the Capsule P radiation analysis. See Section 6.0 of WCAP-14613[5].

Table 4 **Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for Prairie Island Unit 2**

EFPY	0°
Core Mid-Plane	
17.24	2.44
24	3.11
32	3.89
35	4.18
Top of Core	
17.24	1.39
24	1.76
32	2.21
35	2.38

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 Prairie Island Unit 2 reactor vessel for fluence values at the EOL (35 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For Prairie Island Unit 2, the related surveillance program results have been included in this PTS evaluation. Specifically, the Prairie Island Unit 2 plant-specific surveillance capsule data for the lower shell forging D and weld metal is provided and applied as follows:

1) There have been four capsules removed from the reactor vessel.

2) The data for the surveillance program forging material is deemed **not** credible. However, the data was used with a σ_Δ margin of 17°F.

3) The data for the Unit 2 surveillance program weld material is deemed credible.

4) The data for the Unit 1 surveillance program weld material used to evaluate RT_{PTS} for the Unit 2 upper to intermediate shell weld metal is deemed **not** credible. However, the data was used with a σ_Δ margin of 28°F.

5) The surveillance capsule materials are representative of the actual vessel forgings and circumferential weld metal.

6) The resulting RT_{PTS} values are well below the PTS Rule screening criteria.

As presented in Table 5, chemistry factor values for Prairie Island Unit 2 based on average copper and nickel weight percent were calculated using Tables 1 and 2 from Regulatory Guide 1.99, Revision 2 (Position 1.1)[1]. Additionally, chemistry factor values based on the surveillance capsule data are calculated in Table 6. Table 7 contains the RT_{PTS} calculations for all beltline region materials at 35 EFPY.

Table 5	Prairie Island Unit 2 Chemistry Factors per Table 1 and 2 of Regulatory Guide 1.99, Revision 2, Position 1.1	
Material	**Ni, wt %**	**Chemistry Factor, °F**
Upper Shell Forging 22231/39088 Given Cu wt% = 0.07	0.73	44
Intermediate Shell Forging 22829 Given Cu wt% = 0.07	0.75	44
Lower Shell Forging 22642 Given Cu wt% = 0.08	0.67	51
Upper to Intermediate Shell Girth Weld Given Cu wt% = 0.13	0.13	69.7
Intermediate to Lower Shell Girth Weld Given Cu wt% = 0.09	0.11	51.6
Surveillance Weld (Unit 2) Given Cu wt % = 0.08	0.08	44.8
Surveillance Weld (Unit 1) Given Cu wt % = 0.14	0.11	70.9

Table 6 Calculation of Chemistry Factors Using Surveillance Capsule Data Per Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging 22642 (Axial Orientation)	V	0.6206	0.866	35.28	30.55	0.75
	T	1.199	1.051	29.93	31.46	1.10
	R	4.376	1.375	84.73	116.50	1.89
	P	4.165	1.365	103.87	141.78	1.86
Lower Shell Forging 22642 (Tangential Orientation)	V	0.6206	0.866	32.89	28.48	0.75
	T	1.199	1.051	55.69	58.53	1.10
	R	4.376	1.375	90.02	123.78	1.89
	P	4.165	1.365	99.91	136.38	1.86
				SUM	667.46	11.2
	$CF_{Lower\ Shell\ Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 667.46 ÷ 11.2 = **59.6 °F**					
Weld Metal[d]	V	0.6206	0.866	80.58[d]	69.78	0.75
	T	1.199	1.051	66.39[d]	69.78	1.10
	R	4.376	1.375	115.36[d]	158.62	1.89
	P	4.165	1.365	110.68[d]	151.08	1.86
				SUM	449.26	5.6
	$CF_{Weld\ Metal} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 449.26 ÷ 5.6 = **80.2°F**					

NOTES:

(a) f = fluence (10^{19} n/cm^2, E > 1.0 MeV). All updated fluence values were taken from the Capsule P analysis. WCAP-14613[5].

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$

(c) ΔRT_{NDT} values are measured values.

(d) These measured ΔRT_{NDT} values obtained from the Capsule P analysis[5] were multiplied by a ratio factor of 1.15. ($CF_{Vessel} \div CF_{S/C\ Weld}$ = 51.6 ÷ 44.8 = 1.15)

Table 7 RT_{PTS} Calculations for Prairie Island Unit 2 Beltline Region Materials at EOL (35 EFPY)

Material	CF	f[a]	FF[b]	$RT_{NDT(U)}$[c]	M	ΔRT_{PTS}	RT_{PTS}
Upper Shell Forging 22231/39088	44°F	2.38	1.234	-13°F	34°F	54.3°F	75°F
Upper to Inter. Shell Weld Seam W2	70°F	2.38	1.234	-13°F	56°F	86.4°F	129°F
Using Unit 1 S/C Data[d]	81°F	2.38	1.234	-13°F	56°F[d]	100.0°F	**143°F**
Intermediate Shell Forging 22829	44°F	4.18	1.366	14°F	34°F	60.1°F	108°F
Lower Shell Forging 22642	51°F	4.18	1.366	-4°F	34°F	69.7°F	100°F
Using Surveillance Capsule Data	60°F	4.18	1.366	-4°F	34°F[e]	82.0°F	**112°F**
Inter. to Lower Shell Weld Seam W3	52°F	4.18	1.366	-31°F	56°F	71.0°F	96°F
Using S/C Data	80°F	4.18	1.366	-31°F	28°F[f]	109.3°F	106°F

NOTES:

(a) f = peak clad/base metal interface fluence (10^{19} n/cm^2, E > 1.0 MeV) at 35 EFPY. Best-Estimate values are used since the Best-Estimate values are greater than the calculated values (See Table 6-15 in Ref. 5).

(b) $FF = f^{(0.28 - 0.10 \log f)}$

(c) $RT_{NDT(U)}$ values are measured values.

(d) This calculation uses the chemistry factor based on the surveillance capsule weld data from the Prairie Island Unit 1 surveillance program. The margin used is the full σ_Δ value of 28°F since the surveillance weld data is not credible (See App. D in Ref. 16).

(e) Margin used is the full σ_Δ value of 17°F since the forging surveillance data is not credible (See App. D in Ref. 5).

(f) See Appendix D in Ref. 5.

7.0 CONCLUSIONS

As shown in Table 7, all of the beltline region materials in the Prairie Island Unit 2 reactor vessel have EOL RT_{PTS} values well below the screening criteria values of 270°F for plates or forgings and longitudinal welds and 300°F for circumferential welds at EOL (35 EFPY).

8.0 REFERENCES

1. 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

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

3. WCAP-8193, "Northern States Power Co. Prairie Island Unit No. 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko et. al., September 1973.

4. WCAP-11343, "Analysis of Capsule R from the Northern States Power Company Prairie Island Unit 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko et. al., December 1986.

5. WCAP-14613 Rev. 2, "Analysis of Capsule P from the Northern States Power Company Prairie Island Unit 2 Reactor Vessel Radiation Surveillance Program", S.L. Abbott, et. al., February 1998.

6. Societe Des Forges Et Ateliers Du Creusot Usines Schneider, Report No. 3.5.7, Rev. 1, Item C, Heat No. 22829, Order No. 700 814/54, Dated September 8, 1970.

7. Societe Des Forges Et Ateliers Du Creusot Usines Schneider, Report No. 4.5.7, Rev. 1, Item D, Heat No. 22642, Order No. 700 814/54, Dated June 24, 1970.

8. Societe Des Forges et Ateliers Du Creusot Usines Schneider, Report No. 2.5.7, Rev. 1, Item B, Heat No. 22231/39088, Order No. 700 814/54, Dated February 27, 1970.

9. Proces Verbal De Recette De Produits De Soudure, SFAC, Code No. 119 957 TN 54 - 119 937 TN 54, Designation UM 40 (fill) Lot No. 2721, UM 89 (flux), Lot No. 1263, Specification PS 308/R, Dated 7/31/70.

10. Proces Verbal De Recette De Produits De Soudure, S.A.F., Code No. 118 716 A 54 - 119 937 TN 54, Designation UM 40 (fill) Lot No. 1752-69, UM 89 (flux), Lot No. 1263, Specification PS 308/R, Dated 3/26/70.

11. Proces Verbal De Recette De Produits De Soudure, S.A.F., Code No. 118 716 A 54, Designation UM 40 (fill) Lot No. 1752-69, UM 89 (flux), Lot No. 1230, Specification PS 308/R, Dated 3/31/70.

12. Proces Verbal De Recette De Produits De Soudure, S.A.F., Code No. 118 716 A 54, Designation UM 40 (fill) Lot No. 1752-69, UM 89 (flux), Lot No. 1180, Specification PS 308/R, Dated 10/13/68.

13. Calc No. 1, "Prairie Island Unit No. 2 Reactor Vessel Toughness Properties," S. E. Yanichko, 5/4/77.

14. WCAP-14781, Revision 3, "Evaluation of Pressurized Thermal Shock for Prairie Island Unit 1," S.L. Abbott, February 1998.

15. SAE-REA-98-190, "Calculated and Best Estimate Reactor Vessel Fast Neutron Exposures for Prairie Island Units 1 and 2," Letter from REA to MSI dated January 6, 1998.

16. WCAP-14779 Rev. 2, "Analysis of Capsule S from the Northern States Power Company Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program", S.L. Abbott, et. al., February 1998.