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"EVALUATION OF PRESSURIZED THERMAL SHOCK FOR D. C. COOK UNIT 1
FOR 40 YEARS AND FOR 60 YEARS,"
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Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 1 For 40 Years and 60 Years

Westinghouse

WCAP-15879 Revision 0

Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 1 For 40 Years and 60 Years

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PREFACE

This report has been technically reviewed and verified by:

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EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the D. C. Cook Unit 1 reactor vessel beltline materials based upon the results of the re-analysis of Surveillance Capsule U. The conclusion of this report is that all the beltline materials in the D. C. Cook Unit 1 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY). Specifically, the intermediate to lower shell circumferential Weld, Heat 1P3571, was the most limiting material with 32 and 48 EFPY PTS values of 242°F and 267°F respectively.

1 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 a flaw or cause the propagation of a flaw 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 D. C. Cook Unit 1 reactor vessel using the results of the surveillance Capsule U re-analysis. 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 D. C. Cook Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6.0 of WCAP-12483, Revision 1[1]. The results of the RT_{PTS} calculations are presented in Section 6.0 herein. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) 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 latest revision of the PTS Rule, 10 CFR Part 50.61[2], 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 also described in Regulatory Guide 1.99, Revision 2[3].

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 license (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 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 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 at the clad/base metal interface 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.

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

Where,

$RT_{NDT(U)}$ = 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

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

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 RT_{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 determined from Tables 1 and 2 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, when using credible surveillance data, is determined using Equation 5.

The EOL Fluence (f) is the calculated 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 RT_{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 for surveillance materials is determined from Equation 5.

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

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 measured values of RT_{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.

Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}. For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{PTS,\ measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{PTS} = \Delta RT_{PTS\ measured} + 1.0*(T_{capsule} - T_{plant})$$

Note that the temperature adjust methodology has been reinforced by the NRC at the NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the D. C. Cook Unit 1 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". Figure 1 identifies and indicates the location of all beltline region materials for the D. C. Cook Unit 1 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from DIT-B-02230-00[5]. The best estimate copper and nickel content is also documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for the D. C. Cook Unit 1 reactor vessel beltline material properties are also shown in Table 1.

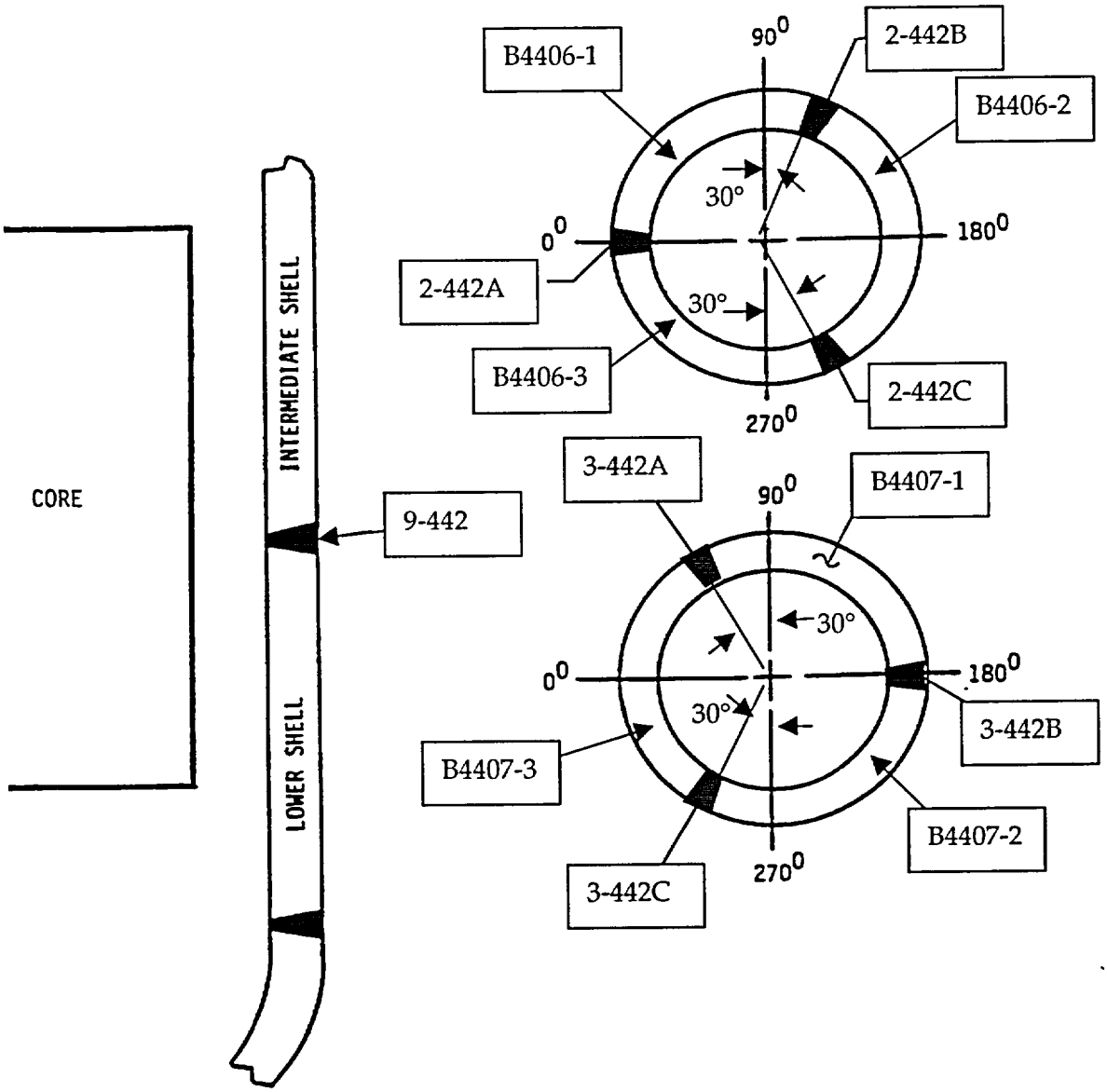

Figure 1: Identification and Location of Beltline Region Materials for the D. C. Cook Unit 1 Reactor Vessel

Table 1

D. C. Cook Unit 1 Reactor Vessel Beltline Unirradiated Material Properties

Material Description		Cu (%)[a]	Ni (%)[a]	Initial RT_{NDT}[b]
Intermediate Shell Plate	B4406-1	0.12	0.52	5
Intermediate Shell Plate	B4406-2	0.15	0.50	33
Intermediate Shell Plate	B4406-3	0.15	0.49	40
Lower Shell Plate	B4407-1	0.14	0.55	28
Lower Shell Plate	B4407-2	0.12	0.59	-12
Lower Shell Plate	B4407-3	0.14	0.50	38
Intermediate Shell Axial Welds (Heat 13253/12008)	2-442 A,B,C	0.21	0.873	-56
Lower Shell Axial Welds (Heat 13253/12008)	3-442 A,B,C	0.21	0.873	-56
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	9-442	0.287	0.756	-56
D. C. Cook Unit 1 Surveillance Weld Metal (Heat 13253)	---	0.27	0.74	---
Kewaunee / Maine Yankee Surv. Weld (1P3571)	---	0.285	0.748	---

Notes:

a) Copper and Nickel content values from Ref. 5, unless otherwise noted.

b) The Initial RT_{NDT} values for the plates are based on measured data from Reference 1 Table A-1, while the welds are generic values.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the clad base metal interface of the D. C. Cook Unit 1 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the results of the Capsule U re-analysis. See Section 6.0 of the Capsule U analysis report, WCAP-12483, Revision 1[1].

TABLE 2
Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for D. C. Cook Unit 1 at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Intermediate and Lower Shell Plates B4406-1, 2 & 3 and B4407-1, 2 & 3.	45°	1.802×10^{19} n/cm^2	2.831×10^{19} n/cm^2
Intermediate Shell Longitudinal Weld Seams 2-442 A (Heat 13253/12008)	0°	0.607×10^{19} n/cm^2	0.927×10^{19} n/cm^2
Intermediate Shell Longitudinal Weld Seams 2-442 B & C (Heat 13253/12008)	30°	1.204×10^{19} n/cm^2	1.883×10^{19} n/cm^2
Lower Shell Longitudinal Weld Seams 3-442 A & C (Heat 13253/12008)	30°	1.204×10^{19} n/cm^2	1.883×10^{19} n/cm^2
Lower Shell Longitudinal Weld Seams 3-442 B (Heat 13253/12008)	0°	0.607×10^{19} n/cm^2	0.927×10^{19} n/cm^2
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	45°	1.802×10^{19} n/cm^2	2.831×10^{19} n/cm^2

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 D. C. Cook Unit 1 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Per 10 CFR Part 50.61, Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. The D. C. Cook Unit 1 surveillance program data has been evaluated and shown to be credible in WCAP-15878[4]. The Kewaunee/Maine Yankee Surveillance weld data was shown to be credible per DIT-B-02230-00[5]. The related surveillance program results have been included in this PTS evaluation.

As presented in Table 3, chemistry factor values for D. C. Cook Unit 1 based on average copper and nickel weight percent values were calculated using Tables 1 and 2 from 10 CFR 50.61[2]. Additionally, the intermediate shell plate B4406-3 chemistry factor, based on credible surveillance capsule data from D. C. Cook Unit 1[4], was calculated in Table 4. The chemistry factor for weld heat 1P3571 for use at D. C. Cook Unit 1 was provided in DIT-B-02230-00 (Temperature and chemistry effects were considered in the calculation). Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material Description		Cu (%)	Ni (%)	Chemistry Factor, °F
Intermediate Shell Plate	B4406-1	0.12	0.52	81.4
Intermediate Shell Plate	B4406-2	0.15	0.50	104.5
→ Using surveillance data	---	---	---	102.3[a]
Intermediate Shell Plate	B4406-3	0.15	0.49	104
→ Using surveillance data	---	---	---	102.3[a]
Lower Shell Plate	B4407-1	0.14	0.55	97.8
Lower Shell Plate	B4407-2	0.12	0.59	82.8
Lower Shell Plate	B4407-3	0.14	0.50	95.5
Intermediate Shell Axial Welds (Heat 13253/12008)	2-442 A,B,C	0.21	0.873	208.7
Lower Shell Axial Welds (Heat 13253/12008)	3-442 A,B,C	0.21	0.873	208.7
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	9-442	0.287	0.756	214
→ Using surveillance data	---	---	---	218.6[b]

Notes:

(a) See Table 4

(b) Determined using Kewaunee and Maine Yankee Surveillance data. The calculation included a chemistry ratio and temperature adjustment (Ref. 5)

TABLE 4

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
Inter. Shell Plate B4406-3 (Longitudinal)	T	0.267	0.641	60	38.460	0.411
	X	0.831	0.948	90	85.320	0.899
	Y	1.195	1.049	105	110.145	1.100
	U	1.837	1.167	115	134.205	1.362
Inter. Shell Plate B4406-3 (Transverse)	T	0.267	0.641	70	44.870	0.411
	X	0.831	0.948	110	104.280	0.899
	Y	1.195	1.049	115	120.635	1.100
	U	1.837	1.167	115	134.205	1.362
				SUM:	772.120	7.544
	$CF_{B4406-3} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (772.120) \div (7.544) = 102.3$[d]					

Notes:

a) f = fluence from the DC Cook Unit 1 Capsule U Re-Analysis (x 10^{19} n/cm^2)[1].

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

c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (Ref 1)

d) Rounded to 1 decimal point.

TABLE 5
RT_{PTS} Calculation for D. C. Cook Unit 1 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence ($x\ 10^{19}$ n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Plate B4406-1	1.802	1.162	81.4	94.6	34	5	134
Intermediate Shell Plate B4406-2	1.802	1.162	104.5	121.4	34	33	188
→ Using surveillance data	1.802	1.162	102.3	118.9	17	33	169
Intermediate Shell Plate B4406-3	1.802	1.162	104	120.8	34	40	195
→ Using surveillance data	1.802	1.162	102.3	118.9	17	40	176
Lower Shell Plate B4407-1	1.802	1.162	97.8	113.6	34	28	176
Lower Shell Plate B4407-2	1.802	1.162	82.8	96.2	34	-12	118
Lower Shell Plate B4407-3	1.802	1.162	95.5	110.9	34	38	183
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C (Heat 13253/12008)	1.204[d]	1.052	208.7	219.6	65.5	-56	229
Lower Shell Longitudinal Weld Seams 3-442 A, B & C (Heat 13253/12008)	1.204[d]	1.052	208.7	219.6	65.5	-56	229
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	1.802	1.162	214	248.7	65.5	-56	258
→ Using surveillance data	1.802	1.162	218.6	254.0	44	-56	242

Notes:
(a) Initial RT_{NDT} values are measured values for plates and generic for welds.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(c) ΔRT_{PTS} = CF * FF
(d) Bounding Fluence Values, See Table 2.

TABLE 6

RT_{PTS} Calculation for D. C. Cook Unit 1 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence (x 10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Plate B4406-1	2.831	1.277	81.4	103.9	34	5	143
Intermediate Shell Plate B4406-2	2.831	1.277	104.5	133.4	34	33	200
→ Using surveillance data	2.831	1.277	102.3	130.6	17	33	181
Intermediate Shell Plate B4406-3	2.831	1.277	104	132.8	34	40	207
→ Using surveillance data	2.831	1.277	102.3	130.6	17	40	188
Lower Shell Plate B4407-1	2.831	1.277	97.8	124.9	34	28	187
Lower Shell Plate B4407-2	2.831	1.277	82.8	105.7	34	-12	128
Lower Shell Plate B4407-3	2.831	1.277	95.5	121.9	34	38	194
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C (Heat 13253/12008)	1.883[d]	1.173	208.7	244.8	65.5	-56	254
Lower Shell Longitudinal Weld Seams 3-442 A, B & C (Heat 13253/12008)	1.883[d]	1.173	208.7	244.8	65.5	-56	254
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	2.831	1.277	214	273.3	65.5	-56	283
→ Using surveillance data	2.831	1.277	218.6	279.2	44	-56	267

Notes:

(a) Initial RT_{NDT} values are measured values for plates and generic for welds.

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

(d) Bounding Fluence Values, See Table 2.

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the D. C. Cook Unit 1 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the intermediate to lower shell circumferential Weld, Heat 1P3571, was the most limiting material with 32 and 48 EFPY PTS values of 242°F and 267°F respectively.

8 REFERENCES

1 WCAP-12483, Revision 1, Analysis of Capsule U from the American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program", J. H. Ledger, E. T. Hayes, December 2002.

2 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.

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

4 WCAP-15878, "D. C. Cook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", J. H. Ledger, November 2002.

5 AEP Design Information Transmittal (DIT) Number DIT-B-02230-00 "Material Chemistry of the Reactor Vessel Belt-line Materials for Cook Nuclear Plant Units 1 & 2" T.Satyan-Sharma, 10/23/01.

APPENDIX A

PROJECTED UPPER SHELF ENERGY

TABLE A-1

Projected End-of-License (32 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE[b] (ft-lb)	Projected USE Decrease[a] (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B4406-1	0.12	1.082	83	22	65
Intermediate Shell Plate B4406-2	0.15	1.082	96	25	72
Intermediate Shell Plate B4406-3	0.15	1.082	98	25	74
Lower Shell Plate B4407-1	0.14	1.082	103	23	79
Lower Shell Plate B4407-2	0.12	1.082	126	22	98
Lower Shell Plate B4407-3	0.14	1.082	108	23	83
Intermediate and Lower Shell Weld Longitudinal Weld Seams (Heat 13253/12008)	0.21	1.082	108	36	69
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	0.287	1.082	105	44	59

Notes

(a) Values are deduced from Figure 6.3-1: Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence.

(b) From RPVData (Reactor Vessel Material Database Version 2.0, PWR-MRP-20, CM-114285, June 2000)

TABLE A-2

Projected End-of-License (48 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE[b] (ft-lb)	Projected USE Decrease[a] (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B4406-1	0.12	1.70	83	23	64
Intermediate Shell Plate B4406-2	0.15	1.70	96	28	69
Intermediate Shell Plate B4406-3	0.15	1.70	98	28	71
Lower Shell Plate B4407-1	0.14	1.70	103	27	75
Lower Shell Plate B4407-2	0.12	1.70	126	23	97
Lower Shell Plate B4407-3	0.14	1.70	108	28	78
Intermediate and Lower Shell Weld Longitudinal Weld Seams (Heat 13253/12008)	0.21	1.70	108	40	65
Intermediate to Lower Shell Circ. weld Seams (Heat 1P3571)	0.287	1.70	105	46	57

Notes:

(c) Values are deduced from Figure 6.3-1: Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence.

(d) From RPVData (Reactor Vessel Material Database Version 2.0, PWR-MRP-20, CM-114285, June 2000)

REQUEST FOR EXEMPTION FROM REQUIREMENTS IN 10 CFR 50.60(a) AND 10 CFR 50, APPENDIX G

Background

Regulation 10 CFR 50.60(a) requires that operational nuclear reactors meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in 10 CFR 50, Appendix G. Regulation 10 CFR 50, Appendix G, Section IV.2.a states that appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions. Further, 10 CFR 50, Appendix G, Section IV.2.b, and the associated table require that the limits be at least as conservative as limits obtained by following the methods of analysis and the safety margins of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Requested Exemption

Indiana Michigan Power Company (I&M) requests an exemption from the requirements of 10 CFR 50.60(a), 10 CFR 50, Appendix G, Section IV.2.b, and the associated table, for Donald C. Cook Nuclear Plant (CNP), Unit 1. I&M requests approval to use ASME Code Case N-641, "Alternative Pressure/Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division I, approved January 17, 2000," in lieu of these requirements. ASME Code Case N-641 presents alternative procedures for calculating pressure-temperature relationships and for calculating low temperature over pressure protection (LTOP) system enable temperatures and allowable pressures.

Application of Code Case N-641

The revised reactor coolant system (RCS) pressure-temperature curves proposed for inclusion in the CNP Unit 1 Technical Specifications, and the LTOP system enable temperatures calculated by Westinghouse Electric Company LLC for future use have been developed in accordance with ASME Code Case N-641. ASME Code Case N-641 allows use of the lower bound K_{IC} fracture toughness curve in lieu of the lower bound K_{IA} fracture toughness curve.

10 CFR 50.60 and 10 CFR 50.12 Requirements

Regulation 10 CFR 50.60(b) states that proposed alternatives to the requirements in 10 CFR 50, Appendix G, or portions thereof, may be used when an exemption is granted by the Nuclear Regulatory Commission (NRC) under 10 CFR 50.12. Regulation 10 CFR 50.12 states that the NRC may grant an exemption from requirements contained in 10 CFR 50 if certain criteria are met. These criteria are addressed below.

- The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G when an exemption is granted by the Commission under 10 CFR 50.12. As described below the other criteria of 10 CFR 50.12 have been met.

- The requested exemption does not present an undue risk to the public health and safety: Use of the K_{IC} curve as the basis fracture toughness curve for the development of RCS pressure-temperature limits, LTOP system pressure setpoints, and LTOP system enable temperatures is more accurate technically than use of the K_{IA} curve. The K_{IC} curve appropriately implements a relationship based on static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor pressure vessel, whereas the K_{IA} curve was developed from more conservative crack arrest and dynamic fracture toughness test data.

  The application of the K_{IA} fracture toughness curve was initially codified in Appendix G to Section XI of the ASME Code in 1974, to provide a conservative representation of RPV material fracture toughness. As documented in an NRC letter approving a similar exemption for Arkansas Nuclear One, Unit No. 2 (see "Precedent Licensing Actions" below), this initial conservatism was necessary due to the limited knowledge of RPV material behavior at that time. The letter also documents that, since 1974, the level of knowledge about the fracture mechanics behavior of RCS materials has been greatly expanded, especially regarding the effects of radiation embrittlement and the understanding of fracture toughness properties under static and dynamic loading conditions. As stated in the letter, this additional knowledge has demonstrated that the lower bound on fracture toughness provided by the K_{IA} fracture toughness curve is beyond the margin of safety required to protect the public health and safety from potential RPV failure.

  Additionally, use of pressure-temperature curves based on the K_{IC} fracture toughness curve may enhance overall unit safety by enlarging the RCS pressure-temperature operating window, with the greatest safety benefit in the region of low temperature operations. The RCS heatup and cooldown operating window is defined by the maximum allowable pressure as determined by brittle fracture considerations, and the minimum required pressure for the reactor coolant pump seals adjusted for instrument uncertainties. A small operating window may have an adverse safety impact by increasing the possibility of inadvertent overpressure protection system actuation due to pressure surges associated with normal unit evolutions such as RCS pump starts and swapping operating charging pumps with the RCS in a water-solid condition. By allowing an increased upper pressure limit that still provides adequate brittle fracture protection, application of ASME Code Case N-641 can result in a benefit to safety by precluding unnecessary overpressure protection system actuation.

- The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

- Special circumstances are present which necessitate the request for an exemption: Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. The regulation lists the conditions that constitute special circumstances. This requested exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G, meets the special circumstances described in paragraph 10 CFR 50.12(a)(2)(ii) which states: "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

  The underlying purpose of the regulations in 10 CFR 50 Appendix G is to specify fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary in order to provide adequate margins of safety under normal operating conditions, system hydrostatic tests, and during transient conditions to which the system may be subjected over its service lifetime. As described above, application of ASME Code Case N-641 to determine pressure-temperature limits and LTOP system enable temperatures provides appropriate procedures to determine limiting maximum postulated defects and consider those defects in establishing the limits and enable temperature. This application of the code case maintains an adequate margin of safety in the fracture toughness requirements for the reactor coolant pressure boundary as was originally contemplated in the regulations. Accordingly, use of ASME Code Case N-641, as described above, achieves the underlying purpose of the associated NRC regulations regarding brittle fracture concerns.

  Therefore, I&M considers that special circumstances are present as defined in 10 CFR 50.12(a)(2)(ii), in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

Precedent Licensing Actions

This exemption request is similar to exemption requests approved for Arkansas Nuclear One, Unit 2, North Anna Power Station, Units 1 and 2, and Turkey Point Units 3 and 4, as documented in the following letters:

- Letter from S. P. Sekerak, NRC, to C. G. Anderson, Entergy Operations, Inc., "Arkansas Nuclear One, Unit No. 2 - Issuance of Amendment re: Reactor Vessel Pressure-Temperature Limits And Exemption From the Requirements of 10 CFR Part 50, Section 50.60(a) (TAC Nos. MB3301 AND MB3302)," dated April 15, 2002

- Letter from S. R. Monarque, NRC, to D. A. Christian, Virginia Electric and Power Company, "North Anna Power Station, Units 1 and 2 - Issuance of Amendments and Exemption From the Requirements of 10 CFR Part 50, Section 50.60(a) re: Amended Pressure-Temperature Limits (TAC Nos. MA9343, MA9344, MA9347, and MA9348)," dated May 2, 2001

- Letter from K. N. Jabbour, NRC, to T. F. Plunkett, Florida Power and Light Company, "Turkey Point Units 3 and 4 - Exemption from the Requirements of 10 CFR Part 50, Section 50.60 and Appendix G (TAC Nos. MA9504 and MA9505)," dated October 24, 2000

The acceptability of Code Case N-641 is also recognized in the proposed Revision 13 of R.G. 1.147 (Draft Regulatory Guide DG-1091, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated December 2001).

RESPONSES, APPLICABLE TO UNIT 1, TO
NRC REQUESTS FOR ADDITIONAL INFORMATION REGARDING
SIMILAR UNIT 2 AMENDMENT REQUEST

The references for this attachment are identified in Section 8 of Enclosure 2 to this letter.

This attachment provides Indiana Michigan Power Company's (I&M's) responses, applicable to the proposed amendment for Donald C. Cook Nuclear Plant (CNP) Unit 1, to Nuclear Regulatory Commission (NRC) requests for additional information regarding a similar proposed license amendment for CNP Unit 2. The proposed Unit 2 amendment was transmitted by a letter from J. E. Pollock, I&M, to the NRC Document Control Desk, AEP:NRC:2349-01, dated July 23, 2002. I&M provided the requested information for Unit 2 by a letter from J. E. Pollock, I&M, to NRC Document Control Desk, AEP:NRC:2349-02, dated November 15, 2002.

NRC Questions 1 through 5 below were transmitted by a letter dated September 27, 2002, from J. F. Stang, NRC, to A. C. Bakken III, I&M. NRC Question 6 was provided to I&M on October 7, 2002, via telecopy.

NRC Question 1

What is the physical basis for the calculated peak inside surface E > 1.0 MeV at 32 effective full power years to be lower than the original FERRET Code adjusted value? This plant has a thermal shield and the transport cross sections were changed to ENDF/B-VI which should have increased the original value.

Response to NRC Question 1

This question is not applicable to Unit 1, since the 32 EFPY end-of-life (EOL) peak fluence value that was calculated for the pressure-temperature curves in Reference 2 is less than the EOL peak fluence calculated under the FLUENCE program conditions. Specifically, the peak fluence at 32 EFPY identified in Table 6-14 of Attachment 3 is 1.802×10^{19} n/cm^2, which is higher than the value of 1.587×10^{19} n/cm^2 in Reference 2.

NRC Question 2

Is the old and the new FERRET Code the same? (We noted that the calculated value was used and not the FERRET Code adjusted value).

Response to NRC Question 2

The FERRET code is a linear least squares adjustment code. The FERRET code itself has not been changed between the capsule U application documented in the original WCAP-12483, (submitted in support of the previous amendment, Reference 4, that updated the Unit 1 RCS pressure-temperature curves prior to Reference 2) and the current application documented in Attachment 3 to this letter. However, due to the evolution from ENDF/B-IV to ENDF/B-VI cross-sections, some of the inputs to the adjustment procedure have changed.

There are three fundamental inputs to the FERRET adjustment procedure. These are:

- The calculated neutron energy spectrum and associated uncertainties.
- The dosimetry reaction cross-sections and associated uncertainties.
- The measured dosimeter reaction rates and associated uncertainties.

In current evaluations, the calculated neutron energy spectra are based on discrete ordinates transport calculations using ENDF/B-VI cross-sections and the uncertainties are based on the latest benchmarking comparisons and sensitivity studies. Prior evaluations used calculations based on ENDF/B-IV transport cross-sections. The ENDF/B-VI analyses result in an increase in the magnitude of the calculated spectra and a reduction in the uncertainty associated with the calculations.

The dosimetry reaction cross-sections used in the current least squares analyses are, likewise, based on the latest ENDF/B-VI data and include extensive uncertainty evaluations. The dosimetry cross-section data set used by Westinghouse Electric Company LLC (Westinghouse) is recommended by the American Society for Testing and Materials (ASTM) for light water reactor applications (ASTM E1018-01, "Standard Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E 706 (IIB)," 2002). Prior least squares evaluations used dosimetry cross-sections obtained from the ENDF/B-IV and ENDF/B-V data files.

Measured reaction rates have not changed in the CNP Unit 1 analyses.

NRC Question 3

The former plates have been added to the revised analysis. Were any of the dosimeters in the shadow of the former plates? Is the peak location in the shadow of the former plates?

Response to NRC Question 3

As described in the following discussion, four dosimetry sets (top-middle, middle, bottom-middle, and bottom) are in the shadow of the former plates. However, the peak location; i.e., maximum pressure vessel fluence, is not in the shadow of the former plates.

Considering the midplane of the active fuel as Z = 0.0, the elevation of the former plates relative to the core midplane is summarized as follows:

Component	Height [cm]
Core Top	182.88
Former 7	140.90 to 144.39
Former 6	89.69 to 93.19
Former 5	38.49 to 41.98
Core Midplane	0.00
Former 4	-16.38 to –12.88
Former 3	-67.58 to –64.09
Former 2	-118.79 to –115.30
Former 1	-173.65 to –170.16
Core Bottom	-182.88

Relative to the maximum pressure vessel fluences listed in Table 6-14 of Attachment 3 to this letter, the fluence values given for 25 EFPY of operation occur at an elevation of approximately +15 centimeters (cm); i.e., 15 cm above core midplane. Due to the average axial shape used in the future fluence projections for 32, 36, 48, and 54 EFPY, the location of the maximum pressure vessel fluence shifts to an elevation of approximately -88 cm; i.e., 88 cm below core midplane. Both of these axial elevations are located midway between former plates. Therefore, the peak location is not in the shadow of the former plates.

The surveillance capsules incorporated into the CNP Unit 1 reactor are centered on the core midplane (Z = 0.0) and have a specimen stack height of 99.56 cm. Thus, the capsules span an axial range extending from –49.78 to +49.78 cm relative to the core midplane. From the table above, it can be seen that only formers 4 and 5 impact this axial span. Former 4 is located below the midplane of the specimen stack and former 5 is positioned near the top of the stack.

Dosimeters are located within the specimen stack at five axial elevations designated top, top-middle, middle, bottom-middle, and bottom. The following tabulation indicates the axial center of the specimens containing dosimeter wires.

Dosimeter Designation	Center of Dosimetry Set [cm]
Top	44.38
Top-Middle	18.85
Middle	-1.97
Bottom-Middle	-25.49
Bottom	-47.08

From this tabulation, it is noted that the neutron sensors positioned at the top-middle, middle, bottom-middle, and bottom axial elevations are located well away from the formers. The sensors located at the top elevation are positioned near the axial location of former 5.

In each capsule, the positioning of former 5 has the potential to impact the measurements obtained with one iron wire, one bare cobalt-aluminum wire, and one cadmium-covered cobalt-aluminum wire. Of these, only the iron wire has an impact on fast neutron evaluations. In the CNP Unit 1 application, iron wires are placed at all five axial elevations. An examination of Table 6-8 of Attachment 3 shows that for all four capsules removed from Unit 1 there is no statistically significant difference between iron measurements at the top location and measurements obtained at the other four axial locations. It can be concluded, therefore, that the presence of former 5 has a minimal impact on the measurements obtained near the top of the capsule.

NRC Question 4

The γ-fission, U-235 impurity, and Pu-239 built-in corrections (Page 6-8) seem to be new in the revision. How were these corrections derived?

Response to NRC Question 4

The γ fission corrections to the U-238 and Np-237 fission dosimeters are now standard practice for dosimetry evaluations, in accordance with Regulatory Guide 1.190 (Reference 3), Regulatory Position 2.1.2. The corrections were determined for each capsule location from the results of the ENDF/B-VI transport calculations using the BUGLE-96 library. The transport calculations were completed for the entire 67 group structure (47 neutron, 20 gamma ray) included in the BUGLE-96 library. From these calculations, the ratio of gamma ray-induced fission to neutron-induced fission was obtained for both of the fission sensors. Based on these calculated ratios, the correction factors associated with the Unit 1 capsules were determined as follows.

Capsule ID and Location	Ratio [U-238(γ,f)]/[U-238(n,f)]	(γ,f) Correction $[1+Ratio]^{-1}$
T (40 Degrees)	0.0439	0.958
X (40 Degrees)	0.0439	0.958
Y (40 Degrees)	0.0439	0.958
U (40 Degrees)	0.0439	0.958

Capsule ID and Location	Ratio [Np-237(γ,f)]/[Np-237(n,f)]	(γ,f) Correction $[1+Ratio]^{-1}$
T (40 Degrees)	0.0156	0.985
X (40 Degrees)	0.0156	0.985
Y (40 Degrees)	0.0156	0.985
U (40 Degrees)	0.0156	0.985

The data in the above tables indicates that the gamma ray-induced fission corrections for the Unit 1 fission sensors are approximately 4 percent and 1.5 percent for U-238 and Np-237, respectively.

Additional corrections for trace impurities of U-235 and for the build-in of plutonium isotopes in U-238 fission sensors have always been a part of dosimetry evaluations performed by Westinghouse. Due to the conversion of U-238 to Pu-239 over time, these corrections are a function of the total fluence accrued by the individual sensors. That is, the longer the irradiation the greater the impact of plutonium fissioning. The corrections used in the CNP Unit 1 dosimetry evaluations were obtained using the ORIGEN code to develop a correlation defining the U-238(n,f) contribution to the total integrated fissions in the dosimeter as a function of the neutron fluence experienced by the sensor. The specific corrections used in the evaluation of the Unit 1 U-238 sensors are summarized as follows:

Capsule ID and Location	Calculated Fluence (E > 1.0 MeV) [n/cm^2]	Fractional U-238 Contribution
T (40 Degrees)	2.667e+18	0.874
X (40 Degrees)	8.313e+18	0.853
Y (40 Degrees)	1.195e+19	0.838
U (40 Degrees)	1.837e+19	0.815

NRC Question 5

It is stated that a 10 percent positive bias was applied to the neutron sources for Cycles 13 and on. Was there also an assumption of low leakage loadings made for the same cycles?

Response to NRC Question 5

WCAP-12483 Revision 1 contains the results of re-analysis to establish that future fluence projections are based on a core power distribution that is representative of the average of fuel cycles 15 through 17. All of these cycles were based on the low leakage fuel management concept. Since I&M intends to treat the average core power distribution used in the fluence projections as a guide for future core designs, a 10% positive bias was applied in the fluence evaluation to establish margin for future core designs.

NRC Question 6

The staff requires the following information to complete its review of the license amendment request for the D.C. Cook 2 32 EFPY pressure-temperature (P-T) limit curves that were proposed based on the methods of Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1." The latest editions of ASME Section XI, Appendix G, endorsed by reference in 10 CFR 50.55a, allow for use of plant specific K_{IT} values and temperature gradient values as acceptable inputs for the calculation of P-T limits for operating reactors. For each 32 EFPY P-T limit data point given in Tables 9-1 and 9-2 of Topical Report WCAP-15047, Revision 2, provide the corresponding K_{IT} value and temperature gradient value (i.e., the ΔT values between the temperatures for RCS coolant and those at the 1/4T and 3/4T thickness locations of the RV) that were used for calculation of the data point.

Response to NRC Question 6

The WCAP referenced in the NRC question, WCAP-15047, Revision 2, applies to CNP Unit 2. The corresponding WCAP for Unit 1 is WCAP-15878, Revision 0, which is provided as Attachment 4 to this letter. The requested data for Tables 9-1 and 9-2 Tables of Attachment 4 is provided in Attachment 8 to this letter.

Attachment 9 to AEP:NRC:2349-03

WESTINGHOUSE LETTER CAW-02-1572, "APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE," DATED NOVEMBER 12, 2002

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-5282
Direct fax: (412) 374-4011
e-mail: Sepp1ha@westinghouse.com

Attention: Mr. Samuel J. Collins

Our ref: CAW-02-1572

November 12, 2002

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-EMT-02-316, "Thermal Stress Intensity Factors for D. C. Cook Unit 1 PT Curves (Proprietary Version)", November 2002.

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1572 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by American Electric Power Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1572 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: G. Shukla/NRR

CAW-02-1572

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

MARGARET L. GONANO
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 14th day
of November, 2002

Margaret L. Gonano
Notary Public

Notarial Seal
Margaret L. Gonano, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 3, 2006
Member, Pennsylvania Association Of Notaries

/cm/0253S.doc

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in letter LTR-EMT-02-316 (Proprietary), November 2002 for D. C. Cook Unit 1 being transmitted by the American Electric Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by American Electric Company for D. C. Cook Unit 1 is expected to be applicable in other licensee submittals in response to certain NRC requests for information to support the Pressure-Temperature curve calculations for D. C. Cook Unit 1.

This information is part of that which will enable Westinghouse to:

(a) Justify the use of plant-specific thermal stress intensity factors for the Pressure-Temperature curve calculations.

(b) Assist the customer to respond to NRC requests for information.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and justification for the use of plant-specific thermal stress intensity factors for the Pressure-Temperature curve calculations.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

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Attachment 10 to AEP:NRC:2349-03

WESTINGHOUSE LETTER LTR-EMT-02-319, "THERMAL STRESS INTENSITY FACTORS FOR D. C. COOK UNIT 1 PT CURVES (NON-PROPRIETARY VERSION)," DATED NOVEMBER 12, 2002

Westinghouse

To: Dave Sklarsky
cc: James Gresham
Thomas Laubham
Justin Ledger

Date 11/12/2002

From Engineering & Materials Technology
Ext· WIN 284-3898
Fax. WIN 284-6647

Your ref·
Our ref· LTR-EMT-02-319

Subject **Thermal Stress Intensity Factors for D.C. Cook Unit 1 PT Curves (Non-Proprietary Version)**

In response to a request for additional information (RAI) from the NRC on pressure-temperature (PT) limit curves, AEP requested Westinghouse supply them with the thermal stress intensity factors associated with the 32 EFPY PT limit curves from WCAP-15878, Revision 0. Attached for AEP's use are the thermal stress intensity factors in question. Table 1 contains the 1/4T and 3/4T thermal stress intensity factors for the 60°F/hr heatup curve, while Table 2 contains the 1/4T thermal stress intensity factors for all the cooldown curves (20, 40, 60 and 100°F/hr). Note that the Cooldown is only limited at the 1/4T location, thus the 3/4T values are not supplied. The heatup curves are limited at both the 1/4T and 3/4T locations, depending on the temperature. In addition, the steady state curve is also limiting over a portion of the heatup curve.

If you have any questions or need additional information, please contact the undersigned.

Author:

J. H. Ledger[1]
Engineering and Materials Technology

Approved by:

T. J. Laubham[1]
Engineering and Materials Technology

Attachments

[1]**Official record electronically approved in EDMS 2000**

Table 1
Kit Values for 60°F/hr Heatup Curve (32 EFPY)

a, b, c

Water Temp (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
Values up to 140°F are limited by the 3/4T location		
60		
65		
70		
75		
80		
85		
90		
95		
100		
105		
110		
115		
120		
125		
130		
135		
140		
Values above 140°F up to 225°F are limited by Steady State		
145		
150		
155		
160		
165		
170		
175		
180		
185		
190		
195		
200		
205		
210		
215		
220		
225		

Table 1 (*Continued*)
Kit Values for 60°F/hr Heatup Curve (32 EFPY)

a, b, c

Water Temp (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
Values above 225°F are limited by the 1/4T location		
230		
235		
240		
245		
250		
255		
260		
265		
270		
275		
280		
285		

- **Note that the Vessel Radius to the ¼T and ¾T Locations are as follows:**

  1/4T Radius = 88.844" &

  3/4T Radius = 93.094"

Table 2
Kit Values for all Cooldown Curves (32 EFPY)

a, b, c

Water Temp. (°F)	20°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	40°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	60°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
215				
210				
205				
200				
195				
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80				
75				
70				
65				
60				

(*) Values above 215°F are limited by Steady State.
(**) Values above 210°F are limited by lower rate or Steady State.
(***) Values above 205°F are limited by lower rate or Steady State.
(****) Values above 195°F are limited by lower rate or Steady State