Westinghouse Non-Proprietary Class 3

WCAP – 13517
Revision 1

Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 2

Westinghouse Electric Company LLC

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-13517, Revision 1

Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 2

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Prepared by the Westinghouse Electric Company LLC
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PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham T.J. Laubham 5/30/02

Record of Revision

Revision 1: The PTS evaluation in Revision 0 was based on "best-estimate" fluences. WCAP-13515 was revised to update the fluence methodology and to include the "calculated" fluences. Thus, this report was revised to incorporate the "calculated" fluences into the D. C. Cook Unit 2 PTS calculations.

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the D. C. Cook Unit 2 reactor vessel beltline materials based upon the results of the Surveillance Capsule U evaluation. The conclusion of this report is that all the beltline materials in the D. C. Cook Unit 2 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, and 300°F for circumferential welds at EOL (32 EFPY) and EOLE (48 EFPY).

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 2 reactor vessel using the results of the surveillance Capsule U 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 D. C. Cook Unit 2 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-13515, Rev. 1[1]. 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 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.

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

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

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{2}$$

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.

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

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.

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

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 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". Figure 1 identifies and indicates the location of all beltline region materials for the D. C. Cook Unit 2 reactor vessel.

The calculated copper and nickel contents of the beltline materials were obtained from ATI report DIT-B-02230-00[4]. The calculated 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 D. C. Cook Unit 2 reactor vessel beltline material properties are also shown in Table 1.

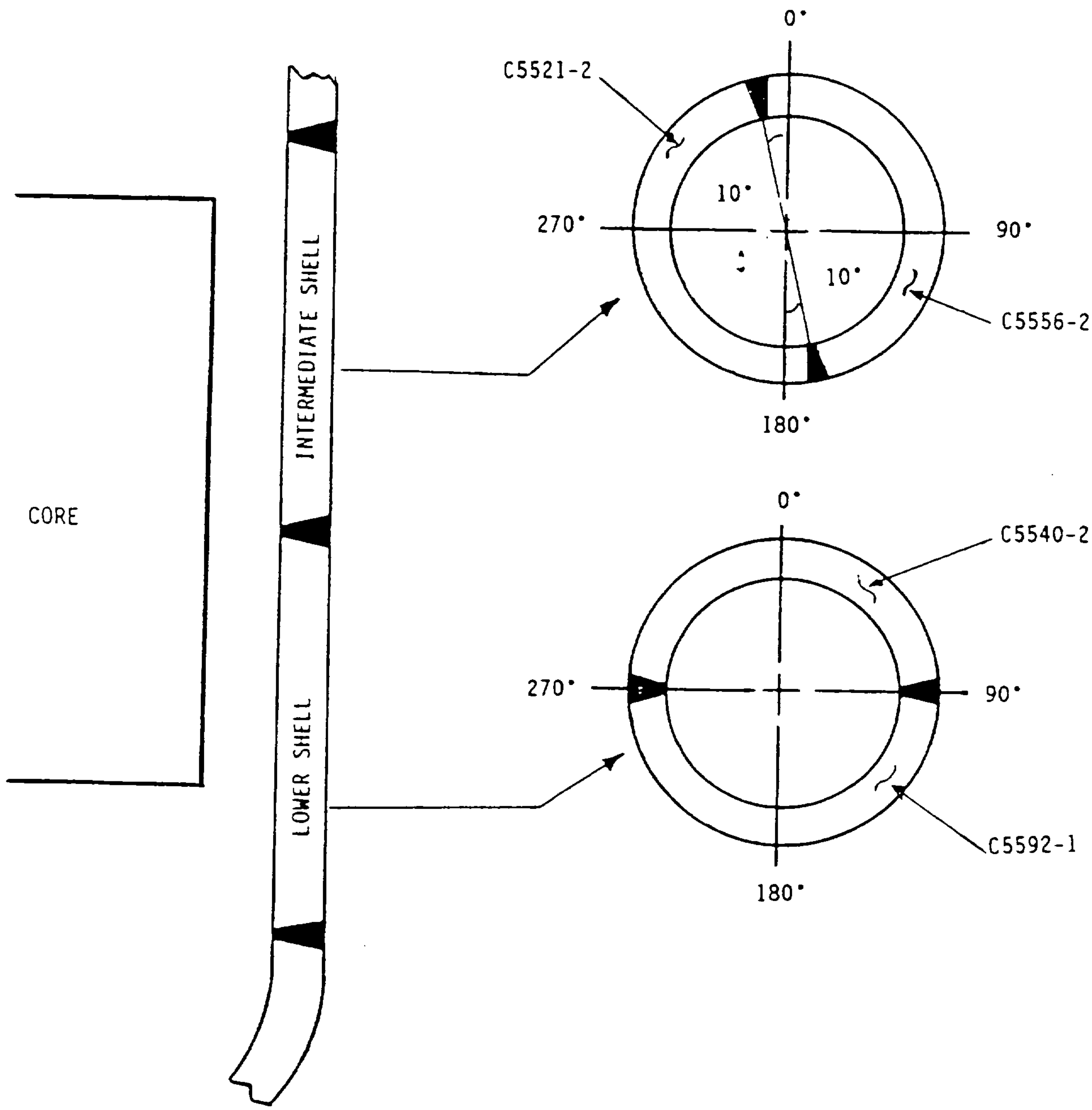

Figure 1: Identification and Location of Beltline Region Materials for the D. C. Cook Unit 2 Reactor Vessel

Table 1

D. C. Cook Unit 2 Reactor Vessel Beltline Unirradiated Material Properties

Material Description		Cu (%)[a]	Ni (%)[a]	Initial RT_{NDT}[b]
Intermediate Shell Plate 10-1	C5556-2	0.150	0.570	58
Intermediate Shell Plate 10-2	C5521-2	0.130[d]	0.580	38
Lower Shell Plate 9-1	C5540-2	0.110	0.640	-20
Lower Shell Plate 9-2	C5592-1	0.140	0.590	-20
Intermediate to Lower Shell Weld	S3986	0.056[c]	0.956[c]	-35
Intermediate Longitudinal Weld	S3986	0.056[c]	0.956[c]	-35
Lower Longitudinal Weld	S3986	0.056[c]	0.956[c]	-35
Surveillance Weld	S3986	0.055[c]	0.970	- -

Notes:
(a) Copper and nickel content values from Ref. 1, Table 4-1 unless otherwise noted.
(b) The Initial RT_{NDT} values from reference 1, Table 5-1.
(c) Per ATI-01-024-T004 (Ref. 5).
(d) Actual value is 0.125 and was conservatively rounded to 0.130. It should also be noted that Inter. Shell Plate 10-2 has credible surveillance data, overriding the weight percent Cu & Ni.

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 2 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the results of the Capsule U analysis. See Section 6 of the revised D. C. Cook Unit 2 Capsule U analysis report, WCAP-13515 Rev. 1[1].

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

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Intermediate Shell Plate 10-1	45°	1.625 x 10^{19} n/cm^2	2.457 x 10^{19} n/cm^2
Intermediate Shell Plate 10-2	45°	1.625 x 10^{19} n/cm^2	2.457 x 10^{19} n/cm^2
Lower Shell Plate 9-1	45°	1.625 x 10^{19} n/cm^2	2.457 x 10^{19} n/cm^2
Lower Shell Plate 9-2	45°	1.625 x 10^{19} n/cm^2	2.457 x 10^{19} n/cm^2
Beltline Welds	45°	1.625 x 10^{19} n/cm^2	2.457 x 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 2 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 2 surveillance program data has been evaluated and shown to be credible in WCAP-15047 Rev. 2[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 2 based on average copper and nickel weight percent values were calculated using Tables 1 and 2 from 10 CFR 50.61[2]. Additionally, chemistry factor values based on credible surveillance capsule data are calculated in Table 4 for D. C. Cook. 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	Cu wt. %	Ni wt. %	Chemistry Factor, °F
Inter. Shell Axial Welds	0.056	0.956	76.4°F
Inter. Shell Plate 10-1 (C5556-2)	0.150	0.570	108.4°F
Inter. Shell Plate 10-2 (C5521-2)	0.130	0.580	90.4°F
Int./Lower Shell Circ. Weld	0.056	0.956	76.4°F
Lower Shell Axial Welds	0.056	0.956	76.4°F
Lower Shell Plate 9-1 (C5540-2)	0.110	0.640	74.6°F
Lower Shell Plate 9-2 (5592- 1)	0.140	0.590	99.5°F
Surveillance Weld Metal	0.055	0.970	75°F

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*ΔRT_{NDT}	FF^2
Intermediate Shell Plate C-5521-2 (Longitudinal)	T	0.238	0.612	55	33.66	0.375
	Y	0.664	0.885	90	79.65	0.783
	X	1.019	1.005	95	95.48	1.010
	U	1.583	1.127	95	107.07	1.270
Intermediate Shell Plate C-5521-2 (Transverse)	T	0.238	0.612	80	48.96	0.375
	Y	0.664	0.885	100	88.50	0.783
	X	1.019	1.005	103	103.52	1.010
	U	1.583	1.127	130[e]	146.51	1.270
				SUM:	703.35	6.876
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (703.35) \div (6.876) =$ **102.3 °F**						
Surveillance Weld Metal	T	0.238	0.612	40.76 (40)[d]	24.95	0.375
	Y	0.664	0.885	50.95 (50)[d]	45.09	0.783
	X	1.019	1.005	71.33 (70)[d]	71.68	1.010
	U	1.583	1.127	76.43 (75)[d]	86.14	1.270
				SUM:	227.86	3.438
$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (227.86) \div (3.438) =$ **66.3°F**						

Notes:

(a) f = Calculated fluence from the D. C. Cook Unit 2 capsule U dosimetry analysis results, (x 10^{19} n/cm^2, E > 1.0 MeV).

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

(c) Data obtained from WCAP-13515 Capsule U Analysis.

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 1.019. Original ΔRT_{NDT} values are in parenthesis.

(e) WCAP-13517 Rev. 0 originally reported this value as 138. This was an error. Per WCAP-13515 the value is 130 ft-lbs.

TABLE 5

RT_{PTS} Calculation for D. C. Cook Unit 2 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence ($x10^{19}$ 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 10-1	1.625	1.134	108.4	122.9	34	58	215
Intermediate Shell Plate 10-2	1.625	1.134	90.4	102.5	34	38	175
Intermediate Shell Plate 10-2 → using S/C Data	1.625	1.134	102.3	116.0	17	38	171
Lower Shell Plate 9-1	1.625	1.134	74.6	84.6	34	-20	99
Lower Shell Plate 9-2	1.625	1.134	99.5	112.8	34	-20	127
Beltline Welds	1.625	1.134	76.4	86.6	56	-35	108
Beltline Welds → using S/C Data	1.625	1.134	66.3	75.2	28	-35	68

Notes:

(a) Initial RT_{NDT} values are measured values

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

(c) ΔRT_{PTS} = CF * FF

TABLE 6
RT_{PTS} Calculation for D. C. Cook Unit 2 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence ($x10^{19}$ 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 10-1	2.457	1.242	108.4	134.6	34	58	227
Intermediate Shell Plate 10-2	2.457	1.242	90.4	112.3	34	38	184
Intermediate Shell Plate 10-2 → using S/C Data	2.457	1.242	102.3	127.1	17	38	182
Lower Shell Plate 9-1	2.457	1.242	74.6	92.7	34	-20	107
Lower Shell Plate 9-2	2.457	1.242	99.5	123.6	34	-20	138
Beltline Welds	2.457	1.242	76.4	94.9	56	-35	116
Beltline Welds → using S/C Data	2.457	1.242	66.3	82.3	28	-35	75

Notes:
(a) Initial RT_{NDT} values are measured values
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(c) $\Delta RT_{PTS} = CF * FF$

7 CONCLUSIONS

As shown in Tables 5 and 6, All beltline materials in the D. C. Cook Unit 2 reactor vessel remain below the PTS screening criteria through the end of current license life (32 EFPY) and license renewal (48 EFPY).

8 REFERENCES

1 WCAP-13515, Revision 1, "Analysis of Capsule U from the Indiana Michigan Power Company D.C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., dated December 2001.

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 AEP Design Information Transmittal (DIT), 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.

5 WCAP-15047, Revision 2, D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," C. Brown, et al., December 2001.

APPENDIX A

PROJECTED UPPER SHELF ENERGY

TABLE A-1
Predicted 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[(a)] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate 10-1	0.15	.975	90	24	68
Intermediate Shell Plate 10-2	0.13	.975	86	22	67
Lower Shell Plate 9-1	0.11	.975	110	19	89
Lower Shell Plate 9-2	0.14	.975	103	23	79
Beltline Welds	0.056	.975	77	10	69

Notes:

(a) Matches RVID2.

(b) Determined using Figure 2 of Reg. Guide 1.99, Rev. 2 with the % Cu and 1/4T Fluence.

Hence, all beltline material USE values remain above 50 ft-lb. through EOL (32 EFPY).

TABLE A-2
Predicted 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[(a)] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate 10-1	0.15	1.475	90	26	67
Intermediate Shell Plate 10-2	0.13	1.475	86	23	66
Lower Shell Plate 9-1	0.11	1.475	110	22	86
Lower Shell Plate 9-2	0.14	1.475	103	25	77
Beltline Welds	0.056	1.475	77	12	68

Notes:
(a) Matches RVID2.
(b) Determined using Figure 2 of Reg. Guide 1.99, Rev. 2 with the % Cu and 1/4T Fluence.

Hence, all beltline material USE values remain above 50 ft-lb. through EOL (48 EFPY).

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**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 2. 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 2 Technical Specifications, and the LTOP system enable temperatures 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.

- <u>The requested exemption is authorized by law</u>: 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.

- <u>The requested exemption does not present an undue risk to the public health and safety:</u> 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 (RPV), 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.

- <u>The requested exemption will not endanger the common defense and security</u>: 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, 2002

The acceptability of Code Case N-641 is also recognized in the proposed Revision 13 of R.G. 1.147 (Draft RG 1091, Reference 6 in Enclosure 2 to this letter).