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January 12, 1987

Re: Indian Point Unit No. 2
Docket No. 50-247
RT_{PTS} Rule - 10 CFR 50.61

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Our letter to you dated January 22, 1986 provided the current and projected values of RT_{PTS}, the reference temperature (for nil ductility transition) for pressurized thermal shock evaluation, for Indian Point Unit 2 reactor vessel beltline material. Enclosed is additional information on that submittal as requested by your staff. A figure entitled "Identification and Location of Beltline Region Material for the Indian Point Unit 2 Reactor Vessel" is also provided to supplement our previous submittal. This new figure replaces "figure 1" as enclosed with our January 22, 1986 letter.

Should you or your staff have any questions on this submittal, please do not hesitate to call me.

Very truly yours,

Murray Selman

Encl.

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cc: Senior Resident Inspector
U. S. Nuclear Regulatory Commission
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Supplemental Information

on

January, 1986

Indian Point Unit 2

RT_{PTS} Submittal

Consolidated Edison Co. of N.Y.
January, 1987

Question: What is the basis of standard loading pattern fluence at reactor vessel ID surface = $5.94 \times 10^{17} n/cm^2$ EFPY used in the January 22, 1986 submittal?

Response: The standard loading pattern fluence at the reactor vessel ID surface is based upon the results of analysis of the Indian Point 2 reactor vessel surveillance capsule Z, and a lead factor of 3.52. The capsule Z analysis indicates an average fast neutron flux of $6.64 \times 10^{10} n/cm^2 \cdot sec$ at the capsule location. Total fluence of $20.91 \times 10^{17} n/cm^2$ EFPY derived from this capsule Z flux value was then divided by the lead factor of 3.52 to generate the fluence at the reactor vessel ID surface of $5.94 \times 10^{17} n/cm^2$ EFPY. The Capsule Z report was provided to the NRC in our May 7, 1984(1) submittal. The average cross section used for calculating neutron flux at the capsule location from the measured activity data, and the lead factor of 3.52 used for calculating reactor vessel flux, are based upon Westinghouse transport theory calculations. These calculations use S8 approximation for neutron flux and P_1 anisotopic scattering (see Attachment A).

Question : What is the basis of Low Leakage Loading Pattern fluence at reactor vessel ID surface = $3.33 \times 10^{17} n/cm^2$ EFPY?

Response: The standard loading pattern neutron fluence at reactor vessel ID surface is reduced by 44% to provide the low leakage loading pattern fluence at the reactor vessel ID surface. As summarized in Appendix B use of the low leakage loading pattern effects a 44% reduction in peak neutron flux at the reactor vesel inner radius.

Question: What is the impact of P_3 anisotropic scattering approximation instead of P_1 approximation on RT_{PTS} values given in Table 3 of January 22, 1986 submittal?

Response: A Westinghouse analysis performed in support of the PTS submittal for a four loop reactor having geometry identical to that of Indian Point 2 demonstrates that the P_3 anisotropic scattering approximation (instead of the P_1 approximation) would result in approximately an 8% higher neutron flux/fluence at the reactor vessel ID surface. This is because the P_3 approximation decreases the lead factor which increases the neutron flux by 17%. At the same time, the P_3 approximation increases the average fast neutron cross-section which decreases the neutron flux by 9%. The net effect is an overall increase of 8% in the neutron flux at the reactor vessel ID surface.

For conservatism, we have used a 10% increase in neutron flux to calculate RT_{PTS} values with P_3 approximation. As shown in the attached table, the P_3 approximation increases RT_{PTS} by only 2 to 4^oF for weld and plate materials. This analysis shows that a large margin currently exists between IP2 vessel RT_{PTS} and the screening criterion (270^oF for plates, forging and axial weld materials, or 300^oF for circumferencial weld materials), and that the projected value of RT_{PTS} will not reach the screening criterion for approximately 78 Effective Full Power Years beyond the end of the Cycle 7.

Reference

(1) Letter from John O'Toole to Steven Varga, dated May 7, 1984

Attachments

(A) Analysis of Neutron Flux Levels and Surveillance Capsule Lead Factors for the Indian Point Unit 2 Reactor, S. L. Anderson, Westinghouse Electric Corp., July, 1979

(B) Analysis of Neutron Flux Levels and Surveillance Capsule Lead Factors for The Indian Point Unit 2 Reactor Using a Low Leakage Core, S. L. Anderson, Westinghouse Electric Corp., July, 1982

TABLE -3
CALCULATED VALUVES OF RT_{PTS} for P_1 & P_3 ANISOTROPIC
SCATTERING APPROXIMATIONS

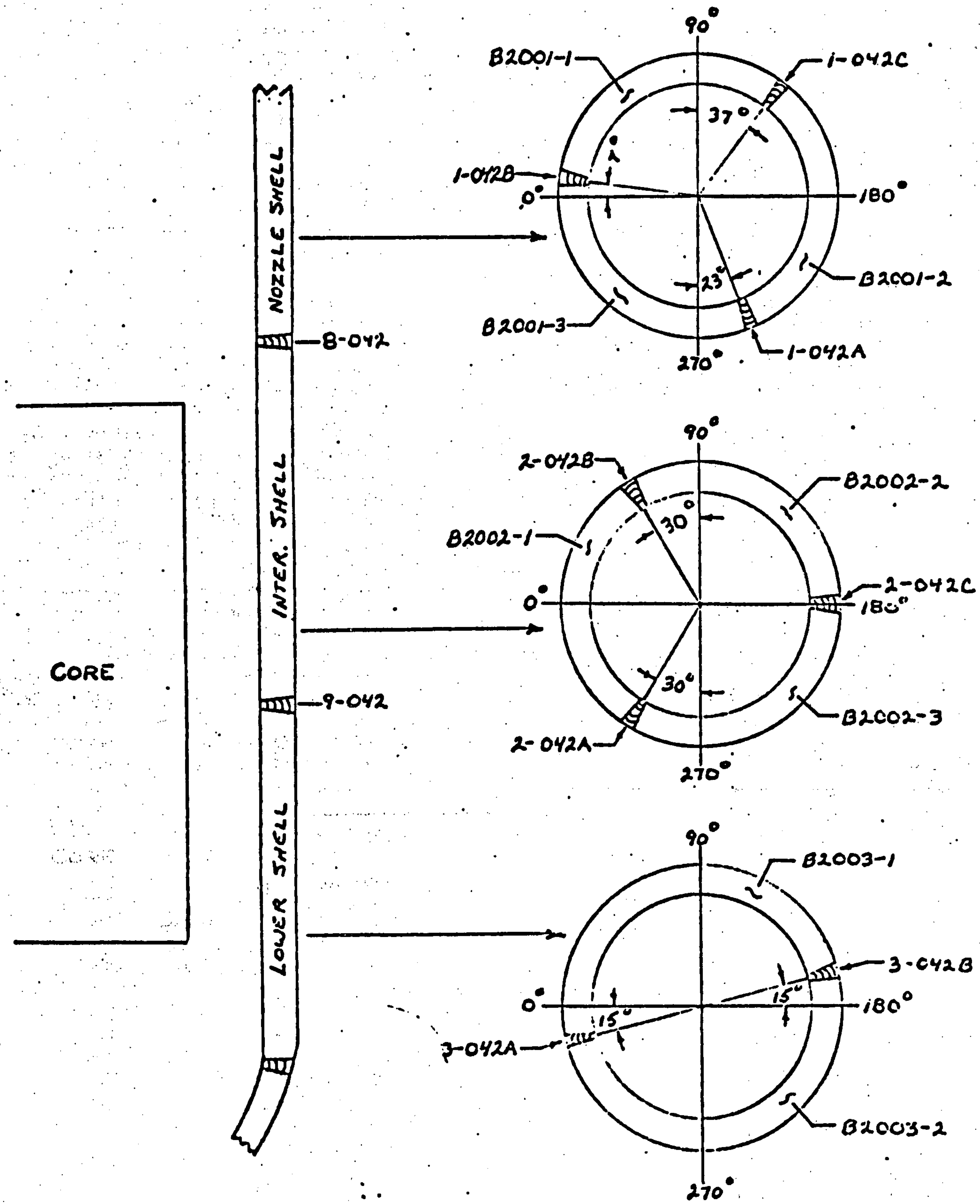
Material	Cu	Data Source	Ni	Data Source	Initial RT_{NDT}	Data Source	Current(a)(f) RT_{PTS} P_1	Current(a)(f) RT_{PTS} P_3		EOL(b)(f) RT_{PTS} P_1	EOL(b)(f) RT_{PTS} P_3	
Plates												
B2001-1	.20	Supplier	.50	Supplier	24°	(c)	164°	166°		187°	190°	
B2001-2	.14	Supplier	.43	Supplier	18°	(c)	125°	127°		141°	142°	
B2001-3	.19	Supplier	.50	Supplier	25°	(c)	160°	162°		182°	185°	
B2002-1	.21	Surveillance Capsules	.65	Surveillance Capsules	34°	(c)	187°	190°		214°	218°	
B2002-2	.17	Surveillance Capsules	.49	Surveillance Capsules	21°	(c)	145°	147°		165°	167°	
B2002-3	.20	Surveillance Capsules	.60	Surveillance Capsules	21°	(c)	166°	169°	R1	191°	194°	R1
B2003-1	.20	Supplier	.66	Supplier	20°	(c)	168°	171°		194°	197°	
B2003-2	.19	Supplier	.48	Supplier	-20°	(c)	114°	116°		136°	139°	
Weld Metal												
W5214	.20	Surveillance Capsules	1.0	Estimate(e)	-56°	generic mean value from 10 CFR 50.61	122°	125°		152°	156°	
34B009	.20	(d)	1.0	Estimate(e)	-56°	generic mean value from 10 CF 50.61	122°	125°		152°	156°	

(a) Fluence as of January, 1986 (end of Cycle 7) is $3.82 \times 10^{18} n/cm^2$ (P_1 appx) and $4.20 \times 10^{18} n/cm^2$ (P_3 appx) R1
(b) Fluence as of October 2006 (EOL) is $8.90 \times 10^{18} n/cm^2$ (P_1 appx) and $9.79 \times 10^{18} n/cm^2$ (P_3 appx).
(c) Estimated from longitudinal data per NRC Standard Review Plan Section 5.3.2 using supplier data
(d) Con Edison letter (John D. O'Toole) to NRC (Steven A Varga) dated 7/5/85
NRC letter (Joseph D. Neighbors) to Con Edison (John D. O'Toole) dated 7/22/85
(e) See page 3 of report
(f) Using the same method of calculation, the projected value of RT_{PTS} for the limiting material will not reach the 10 CFR 50.61 screening criterion for 87 Effective Full Power Years (P_1 app) and 78 Effective Full Power Years (P_3 Appx) beyond the end of Cycle 7 R1

Revised Figure 1

FIGURE 1

IDENTIFICATION AND LOCATION OF BELTLINE REGION
MATERIAL FOR THE INDIAN POINT UNIT NO. 2 REACTOR VESSEL



Attachment A

<u>Indian Point Unit 2</u>

<u>Reactor Vessel Lead Factor Analysis</u>

<u>Report</u>