A. B. Holt, Chief
Technical Assistance Branch, SS

AUG 1 1966

S. Pawlicki
Technical Assistance Branch, SS

INDIAN POINT NO. 2 - REACTOR VESSEL EMBRITTLEMENT

This memorandum reviews the "First Supplement to Preliminary Safety Analysis Report" (Ref. 1) for the Consolidated Edison's Indian Point Unit No. 2, with reference to maximum fast neutron exposure of the pressure vessel and the corresponding shift in the NDT temperature. According to my calculations, the reported time-integrated fast (>1Mev) neutron exposure of the vessel of $0.85 \times 10^{19} n/cm^2$ seems realistic, considering core size, power density and the inner diameter of the reactor vessel. Attached to this memorandum is a copy of Westinghouse reference design curves for the prediction of NDT temperature shift, a list of references for the data points, and a fracture analysis diagram used by the applicant to estimate the embrittlement safety margin.

The pertinent Indian Point No. 2 plant design parameters are as follows:

1.	Core thermal power	2758 MW
2.	Plant design life	40 years
3.	Load factor	0.8
4.	Active core height	12 ft.
5.	Effective core diameter	113.7 in.
6.	Pressure vessel I.D.	173.0 in.
7.	Coolant inlet temperature	543°F

Using these parameters, I calculated the average power density to be $83 \frac{watt}{cc}$, and the integrated neutron exposure of the vessel of about 1×10^{19} n/cm^2. This agrees well with the reported value of 0.85×10^{19}, obtained using a corrected P1MG, one-dimensional 55 group diffusion code. The correction factors applied to the P1MG included the following:

OFFICE ▶
SURNAME ▶
DATE ▶
8111230148 660801
ADOCK 05000247
Form AEC-318 (Rev. 9-53)
U. S. GOVERNMENT PRINTING OFFICE 16—62761-3

AUG 1 1966

1. Axial peaking factor of 1.5

2. Corner effect factor of 1.25, which accounts for the irregular shape of the core (azimuthal flux variation).

3. Spatial correction factor of 2.7, obtained by comparison of PINC results with experimental fast neutron data in all water medium.

The applicant expects the actual vessel exposure to be lower by a factor of 1.7, due mainly to the conservatism in the assumed spatial correction factor, and the assumption that all correction factors occur at a given location on the reactor vessel for the entire life of the plant. I consider the calculations to be fairly realistic, and would assign the standard ± 50% accuracy to the calculated nominal exposure of 0.85×10^{19} n/cm^2.

The shift in the NDT temperature at the inner surface of the vessel wall, due to fast neutron exposure, was estimated as 160°F using the 550°F curve (attached). To account for the variation of NDT temperature across plate thickness, the applicant assumed (most likely based on Yankee measurements) that the initial NDT temperature at plate center may be up to 85°F higher than that at quarter thickness. However, while the impact properties are lowest at plate center, the transition temperature shift there was calculated to be only 85°F. Curve A in the attached Fracture Analysis Diagram represents the initial NDT temperature at plate center, and curve B the corresponding shift of 85°F.

The fracture analysis diagram, used by the applicant to estimate his safety margin with respect to neutron embrittlement of the reactor vessel, was obtained from the NRL Report 5920 (Ref. 2). Using this diagram, and assuming a total NDT shift of 170°F (85° due to reduced impact properties at plate center and 85°F due to irradiation), the applicant found a margin of well over 200°F between limiting plant operating conditions (Curve C) and the nearest point on the Crack Arrest Curve, adjusted for the NDT temperature shift (Curve B). It seems significant that the available safety margin was estimated using the CAT curve, rather than the 20% yield stress point at the (NDT + 60) temperature.

cc: Forrest Western, Dir., SS
J. J. DiNunno, Asst. Dir., SS

OFFICE ▶	SS:TAB					
SURNAME ▶	SPawlicki:amp					
DATE ▶	7/29/66					

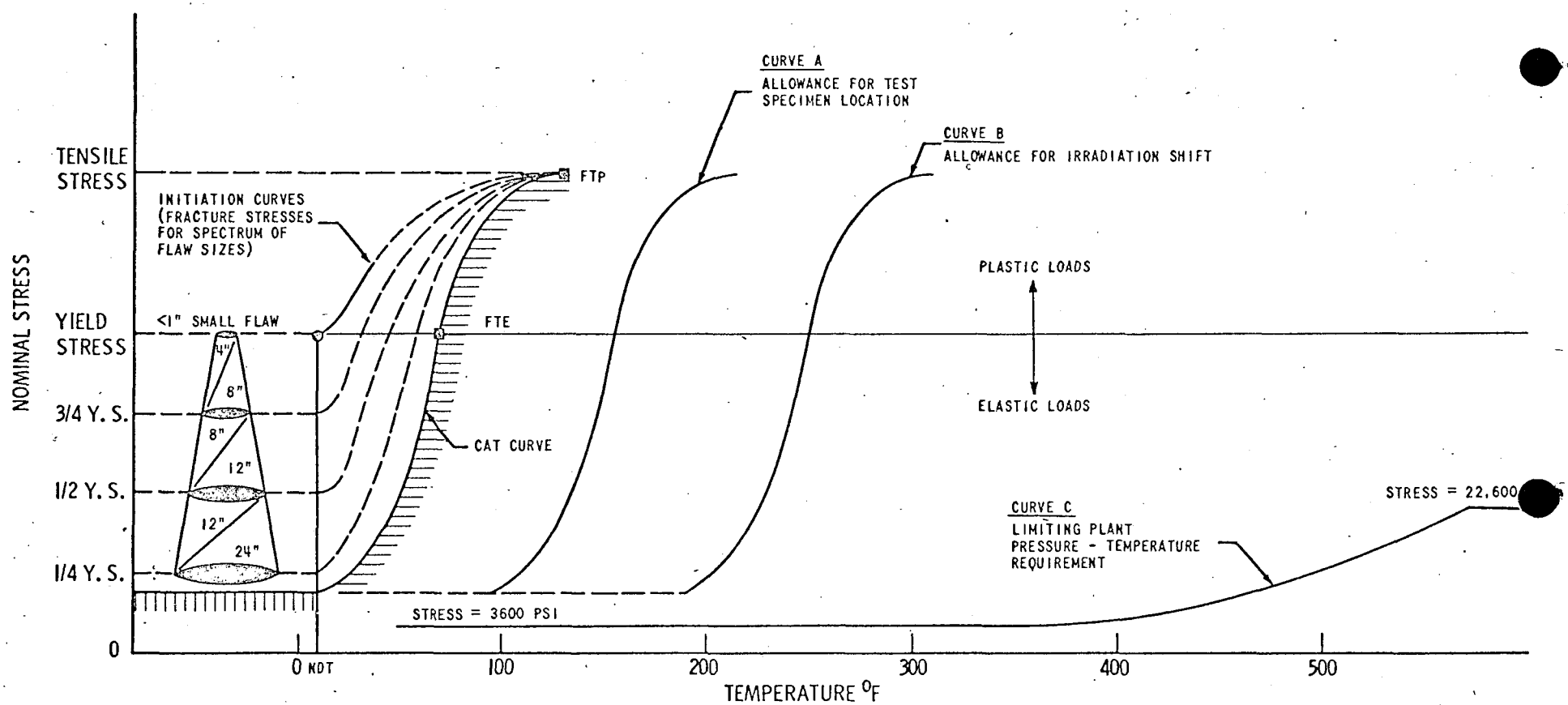
AUG 1 1966

References

1. Consolidated Edison Co. of New York, Inc., Indian Point Nuclear Generating Unit No. 2, "First Supplement to Preliminary Safety Analysis Report," March 1966.

2. W. S. Pellini and P. P. Puzak, "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures", NRL Report 5920, March 1963.

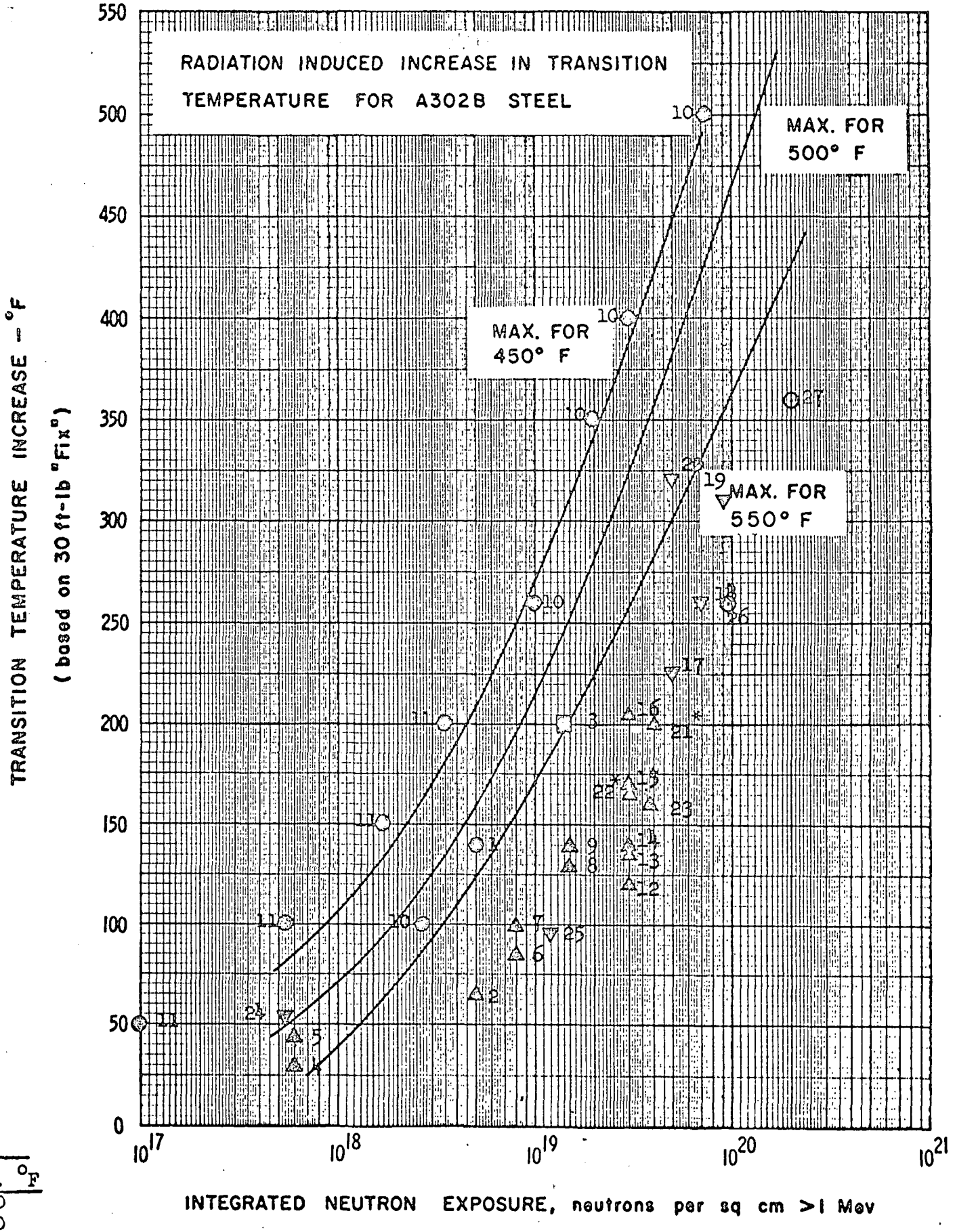
OFFICE ▶						
SURNAME ▶						
DATE ▶						



FRACTURE ANALYSIS DIAGRAM

ATTACHMENT I - QUESTION 3

FIGURE 3-1



Code

Temp. °F
450
490
550
475 to 540
600

Numbers 1 through 27 (see attached sheets)

References for

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
1.	NRL Report 6160 Page 12	SA302B	450	5×10^{18}	140
2.	NRL Report 6160 Page 12	SA302B	550	5×10^{18}	65
3.	NRL Report 6160 Page 13	SA302B	490	1.4×10^{19}	200
4.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	30**
5.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	45
6.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	85**
7.	ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	100
8.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	130**
9.	ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	140
10.	NRL Report 6160 Page 6	All Steels	<450	Various	Various
11.	Nuclear Science & Engineering 19:18-38 (1964)	SA302B	<450	Various	Various
12.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	120
13.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	135

**Transverse Specimens

E.L.

References (Continued)

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
14.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	140
15.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	170
16.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	205
17.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	225
18.	NRL Report 6179 Page 9	SA302B	475-540	7×10^{19}	260
19.	NRL Report 6179 Page 9	SA302B	475-540	9×10^{19}	310
20.	NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	320
21.	NRL Report 6160 Page 15	SA302B	540*	4×10^{19}	200
22.	NRL Report 6160 Page 15	SA302B	540*	3×10^{19}	165
23.	Private Communication with NRL	SA302B	550	3.8×10^{19}	160
24.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	≈525	5.4×10^{18}	54

E.L.

References (Continued)

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (> 1 Mev)	NDT °F
25.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	≈525	1.2×10^{19}	96
26.	"	SA302B	≈600	9.5×10^{19}	260
27.	"	SA302B	≈600	2×10^{20}	360

E.L.