YANKEE ATOMIC ELECTRIC COMPANY

Telephone (508) 779-6711
TWX 710-380-7619

YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

February 20, 199.
BYR 91-019

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Attention: Mr. Patrick Sears
Senior Project Manager
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) Letter, Yankee Atomic Electric Company to NRC, dated November 28, 1990

Subject: Reactor Pressure Vessel Fluence Uncertainty

Dear Sir:

In Reference (b), Yankee Atomic Electric Company (Yankee) submitted the results of an updated reactor pressure vessel fluence assessment. In telephone conversations with Mr. Lambros Lois of the NRC on the assessment, further work was requested with regard to the uncertainty applied to the Yankee fluence calculations. Attachment A provides a response to the uncertainty request.

Included in Attachment A is the comparison of the Westinghouse methodology to the PCA benchmark experiments, comparison of Westinghouse calculations to surveillance capsule and cavity dosimetry measurements, and a radial traverse of fast neutron flux within the reactor internals and pressure vessel. As a result of the comparison of Westinghouse calculation to measured data, Yankee has concluded that a 13% bias should be applied to the Westinghouse fluence calculations.

The application of a 13% bias on the fluence has resulted in an increase in the projected fluence. Attachment B contains the fluence and associated reference temperatures for the beltline materials at the end of Cycle 21 (1992) and the end of Cycle 22 (1994).

Sincerely,

John D Haseltine

John D. Haseltine
Director
Yankee Project

JDH/gjt/WPP77/272
Attachments
cc: B. Elliott (NRC, NRR)
R. Wessman (NRC, NRR)
W. Russell (NRC, NRR)

A001
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ATTACHMENT A

VALIDATION OF NEUTRON TRANSPORT METHODOLOGY

The neutron transport methodology used by Westinghouse in the evaluation of the fast neutron exposure experienced by reactor pressure vessels has been designed to be consistent with ASTM Standards E482, "Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance" and E853, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results". In keeping with the recommendations set forth in those standards, the methodology has been validated by comparison with measurements obtained from a series of facilities exhibiting various degrees of complexity. In particular comparisons of analytical predictions with measurements have been carried for the PCA benchmark experiments carried out at ORNL and for a variety of surveillance capsule and reactor cavity data sets obtained from power reactor irradiations.

PCA COMPARISONS

The PCA benchmark experiments, documented in References 1, 2, and 3, provide an opportunity to test basic transport methodology in a well characterized environment. The PCA replica is a slab configuration designed to model a thermal shield - pressure vessel geometry in a pool type reactor. The core source strength and spatial distribution as well as material compositions and dimensions were well characterized and supplied to the analyst. Therefore, any uncertainties relating to the determination of the core neutron source or the reactor geometry were eliminated and the test of the methodology was limited to basic cross-section input and other code approximations such as mesh size, order of angular quadrature, cross-section expansion, energy group structure, and convergence criteria.

This phase of the benchmarking studies used in the evaluation of the Westinghouse methodology was based on the analysis of the PCA 12/13 experimental configuration. A schematic description of this configuration is provided in Figures 1 and 2. A plan view of the PCA reactor and pressure vessel simulator showing materials characteristic of the core axial midplane is shown in Figure 1; whereas, a section view through the center of the mockup is shown in Figure 2.

The 12/13 configuration was chosen for this evaluation due to the geometric similarity of this particular mockup to the thermal shield - downcomer - pressure vessel designs that are typical of many pressurized water reactors. Of particular note in regard to the areas of similarity are the 12 cm. water gap on the core side of the thermal shield, the 13 cm. water gap between the thermal shield and the pressure vessel, the 6 cm. thick stainless steel thermal shield, the 22.5 cm. thick low alloy steel pressure vessel, and the simulated reactor cavity (void box) positioned behind the pressure vessel mockup.

From the viewpoint of fast neutron attenuation, the 12/13 configuration results in a reduction factor for fast neutron flux (E > 1.0 MeV) of approximately 1000 between the reactor core and the inner wall of the pressure vessel; and a corresponding reduction factor of about 30 from the inner surface to the outer surface of the pressure vessel wall. These similarities in both geometry and the neutron attenuating properties of the configuration provide confidence that judgements made regarding analytical/experimental comparisons in the benchmark mockup can be related to analyses performed for operating pressurized water reactors.

During the PCA experiments, measurements were taken at several locations within the mockup to provide traverse data extending from the reactor core outward through the pressure vessel simulator. The specific measurement locations applicable to the 12/13 configuration are listed in Table 1. It should be noted that all of the measurements were obtained on the lateral centerline of the mockup at the axial midplane elevation.

Data from locations A4, A5, and A6 internal to the pressure vessel simulator establish the means for verification of calculated neutron flux magnitudes and exposure gradients within the pressure vessel wall itself. Since measurements at operating power reactors can at best provide data in the downcomer region interior to the vessel wall or in the reactor cavity exterior to the vessel wall, these PCA data points establish a key set of comparisons to aid in the accurate determination of embrittlement gradients within the pressure vessel of operating power plants; and provide valuable information to be used in extrapolating power reactor surveillance capsule and cavity dosimetry measurements to positions within the pressure vessel wall.

Comparison of analytical predictions with measurements at the A4, A5, and A6 locations of the 12/13 configuration are provided in Table 2. These comparisons are provided on two levels. The first comparison relates predicted reaction rates for the Ni-58 (n,p), U-238 (n,f), and Np-237 (n,f) reactions directly with the individual foil measurements. The second comparison relates the calculated flux (E > 1.0 MeV) with the recommended flux values derived from a least squares adjustment of the measured foil data.

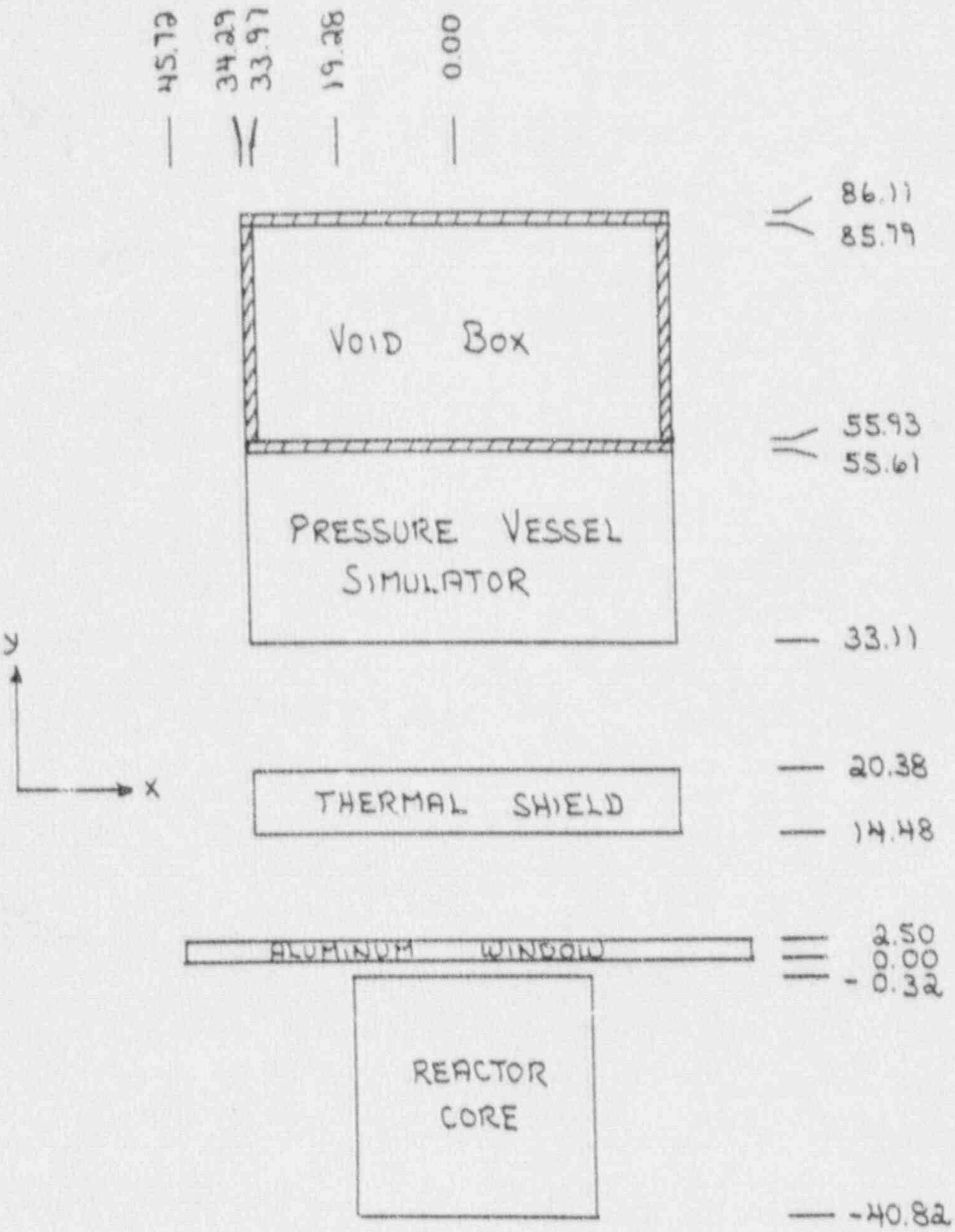
From Table 2 it is noted that agreement among the various predictions and measurements is excellent. The agreement for the individual foil reactions ranged from 2.5 - 10.6 percent, while the comparison with the derived fast neutron flux ranged from 3.4 -7.7 percent. It should be noted that this level of agreement is consistent with the 1 sigma uncertainties associated with the measurements themselves.

From these comparisons it is concluded that, given an accurate representation of the core neutron source and the reactor geometry, the basic transport methodology currently in use at Westinghouse will produce accurate representations of neutron flux magnitude, energy spectrum, and exposure gradients within light water reactor pressure vessels.

ATTACHMENT A

FIGURE 1

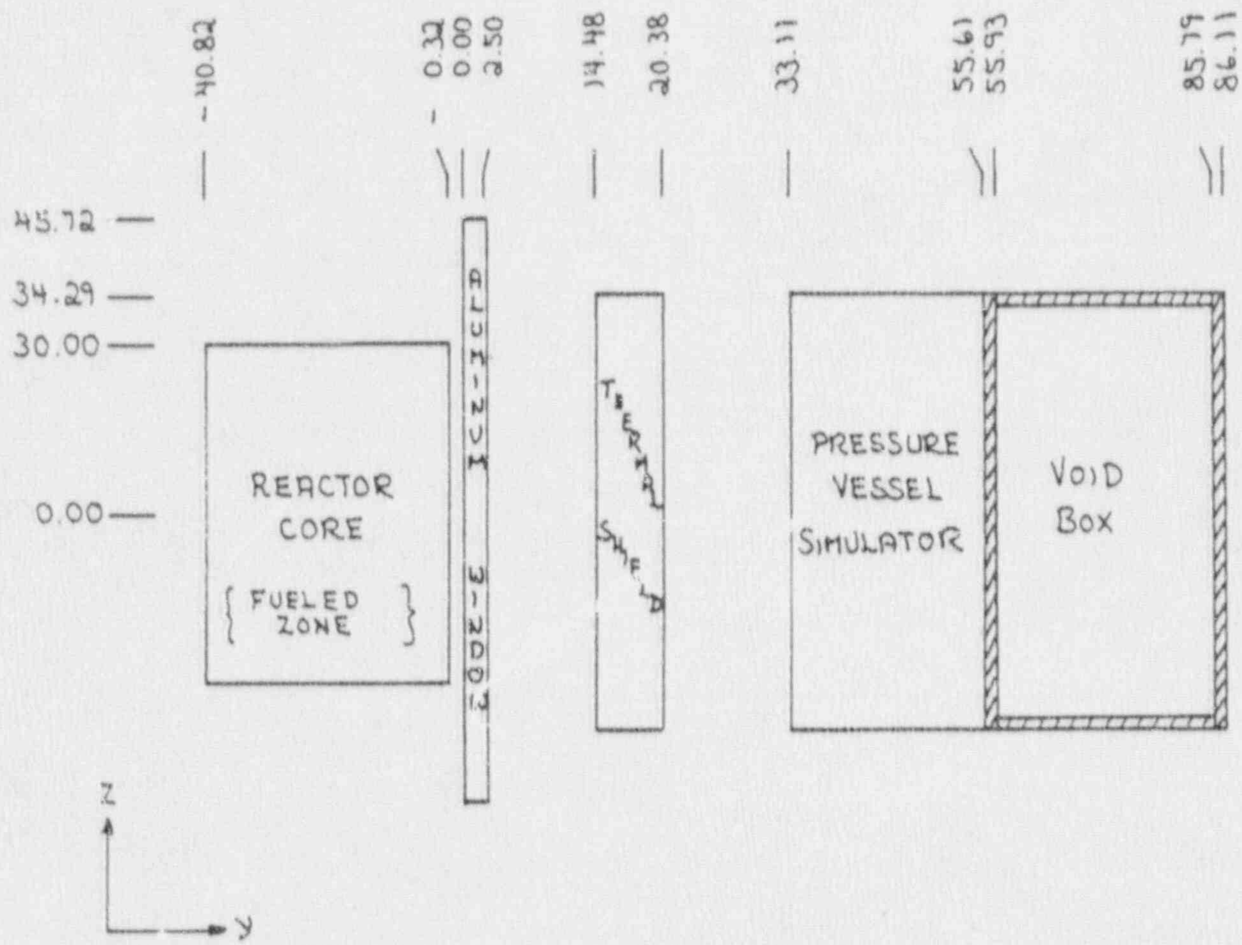
PCA 12/13 CONFIGURATION - X,Y GEOMETRY



ATTACHMENT A

FIGURE 2

PCA 12/13 CONFIGURATION - Y,Z GEOMETRY



ATTACHMENT A

TABLE 1

SUMMARY OF MEASUREMENT LOCATIONS WITHIN
THE PCA 12/13 CONFIGURATION

LOCATION	ID	Y(cm)
Core Center	A0	-20.50
Thermal Shield Front	A1	11.98
Thermal Shield Back	A2	22.80
Pressure Vessel Front	A3	29.71
Pressure Vessel 1/4T	A4	39.51
Pressure Vessel 1/2T	A5	44.67
Pressure Vessel 3/4T	A6	50.13
Void Box	A7	59.13

Note: Y dimensions are referenced to the core side of the aluminum window (see Figure 1)

TABLE 2

COMPARISON OF CALCULATED AND MEASURED SENSOR REACTION RATES
WITHIN THE PCA 12/13 PRESSURE VESSEL SIMULATOR

Ni-58 (n,p) REACTION RATE (rps/nucleus)

LOCATION	CALCULATED	MEASURED	C/M
A4	5.52E-09	5.69E-09	0.970
A5	2.18E-09	2.25E-09	0.969
A6	8.45E-10	7.99E-10	1.058

U-238 (n,f) REACTION RATE (rps/nucleus)

LOCATION	CALCULATED	MEASURED	C/M
A4	1.73E-08	1.69E-08	1.024
A5	7.56E-09	7.36E-09	1.027
A6	3.22E-09	3.11E-09	1.035

Np-237 (n,f) REACTION RATE (rps/nucleus)

LOCATION	CALCULATED	MEASURED	C/M
A4	1.20E-07	1.17E-07	1.026
A5	6.72E-08	6.17E-08	1.089
A6	3.65E-08	3.30E-08	1.106

COMPARISON OF CALCULATED AND MEASURED FAST NEUTRON FLUX (E > 1.0 MeV)
WITHIN THE PCA 12/13 PRESSURE VESSEL SIMULATOR

NEUTRON FLUX (n/cm2-sec)

LOCATION	CALCULATED	MEASURED	C/M
A4	4.51E-08	4.36E-08	1.034
A5	2.18E-08	2.08E-08	1.048
A6	1.01E-08	9.39E-09	1.077

SURVEILLANCE CAPSULE COMPARISONS

Over the course of the operating lifetime of power reactors, surveillance capsules are periodically withdrawn to provide materials data as well as neutron dosimetry applicable to the specific reactor. These dosimetry results afford the opportunity for power reactor benchmarking against for points located interior to the reactor vessel wall. The following is a summary of comparisons of plant specific calculations with capsule measurements for a variety of Westinghouse reactors.

PLANT/CAPSULE	Neutron Flux (n/cm2-sec) CALCULATED	MEASURED	C/M
A1	9.25E+10	1.01E+11	0.916
B1	1.08E+11	1.34E+11	0.806
C1	9.25E+10	1.06E+11	0.873
D1	9.58E+10	1.01E+11	0.949
E1	9.44E+10	1.05E+11	0.899
F1	1.07E+11	1.24E+11	0.863
G1	9.23E+10	9.47E+10	0.975
H1	9.51E+10	1.09E+11	0.872
I1	9.51E+10	1.09E+11	0.872
J1	9.32E+10	1.30E+11	0.717
J2	8.94E+10	1.01E+11	0.885
K1	8.33E+10	1.04E+11	0.801
K2	9.21E+10	1.10E+11	0.837
L1	9.52E+10	1.16E+11	0.821
L2	8.34E+10	9.05E+10	0.922
M1	1.10E+11	1.43E+11	0.769
M2	6.64E+10	8.56E+10	0.776
M3	1.10E+11	1.46E+11	0.753
N1	1.31E+11	1.61E+11	0.814
N2	1.19E+11	1.42E+11	0.838
N3	7.66E+10	8.27E+10	0.926
O1	6.15E+10	6.92E+10	0.889
O2	6.77E+10	7.30E+10	0.927
O3	5.94E+10	6.28E+10	0.946
P1	5.64E+10	6.47E+10	0.872
P2	5.96E+10	6.84E+10	0.871
P3	5.41E+10	4.99E+10	1.084
Q1	6.45E+10	7.47E+10	0.863
Q2	7.04E+10	8.43E+10	0.835
Q3	7.26E+10	7.12E+10	1.020
Q4	6.33E+10	5.78E+10	1.095

AVERAGE C/M RATIO FOR 31 SURVEILLANCE DATA POINTS 0.880

1 SIGMA STANDARD DEVIATION OF THE DATA BASE 0.085

From this surveillance capsule data base, it is seen that at the capsule locations calculated values using the current radiation transport methodology tend to be low relative to measurement by about 12 %.

REACTOR CAVITY DOSIMETRY COMPARISONS

Over the course of the last decade many utilities either because of a need for very accurate fluence evaluations for regulatory concerns or as a mode of data aquisition to establish a life extension data base have installed neutron dosimetry in the annular space between the outer radius of the reactor vessel and the inner radius of the primary biological shield. Programs have been in place since 1983 and comparisons of calculations with measured data from these programs provide an additional means to benchmark analytical capability against data obtained directly from power reactor facilities. The following is a summary of comparisons of plant specific calculations with cavity dosimetry measurements from a variety of Westinghouse reactors.

Neutron Flux (n/cm2-sec)

PLANT/DATA POINT	CALCULATED	MEASURED	C/M
R1	6.86E+08	8.13E+08	0.844
R2	6.14E+08	6.75E+08	0.910
R3	4.01E+08	3.78E+08	1.061
R4	2.75E+08	2.99E+08	0.920
R5	5.65E+08	6.49E+08	0.871
R6	5.25E+08	6.37E+08	0.824
R7	2.97E+08	3.28E+08	0.905
R8	2.40E+08	3.17E+08	0.757
S1	6.62E+08	8.22E+08	0.805
S2	6.44E+08	6.37E+08	1.011
S3	6.60E+08	6.99E+08	0.944
S4	4.92E+08	5.41E+08	0.909
S5	5.07E+08	6.65E+08	0.762
S6	4.64E+08	5.82E+08	0.797
S7	4.23E+08	4.02E+08	1.052
S8	3.39E+08	3.70E+08	0.916
T1	5.36E+08	5.51E+08	0.973
T2	4.44E+08	4.57E+08	0.972
T3	3.33E+08	3.58E+08	0.930
T4	2.04E+08	2.34E+08	0.872
T5	5.25E+08	6.39E+08	0.822
T6	4.44E+08	5.10E+08	0.871
T7	3.83E+08	4.34E+08	0.882
T8	2.56E+08	2.79E+08	0.918
U1	4.76E+08	5.53E+08	0.861
U2	4.16E+08	5.12E+08	0.813
U3	3.70E+08	4.39E+08	0.843
U4	2.40E+08	2.94E+08	0.816
V1	1.73E+09	1.87E+09	0.925
V2	1.45E+09	1.69E+09	0.858
V3	1.12E+09	1.23E+09	0.911
V4	9.28E+08	1.10E+09	0.844

AVERAGE C/M RATIO FOR 32 CAVITY DATA POINTS 0.887

1 SIGMA STANDARD DEVIATION OF THE DATA BASE 0.073

From this cavity dosimetry data base, it is seen that at the cavity sensor locations calculated values using the current radiation transport methodology tend to be low relative to measurement by about 11 %. This observation is fully consistent with the previous comparisons from the surveillance capsule data base.

Combining the 31 sample capsule data base and the 32 sample cavity data base yields an overall average C/M ratio of 0.884 with a 1 sigma standard deviation of 0.079. That is the calculations tend to be biased low by a factor of 0.884 and the uncertainty on that bias factor is approximately 9 percent. This observed bias may be removed from the calculation by multiplying the analytical results by a factor of (1.0)/(0.884) = 1.13.

The implication of the data base evaluation is that, having applied the bias factor to the analytical results, further comparison with plant specific measurements either from surveillance capsules or reactor cavity should result in observed C/M ratios of 1.00 $\pm$ 0.09, where 0.09 represents a 1 sigma standard deviation in the final ratio.

It should be noted that, since the surveillance capsule and cavity dosimetry data bases were obtained from a number of different reactors, the observed variations in C/M ratios include not only the effect of cycle to cycle variations at a given facility, but also the effects of varying reactor geometries and operational characteristics from reactor to reactor. Therefore, the application of this multiple facility bias factor to the Yankee-Rowe fluence calculations is a reasonable approach to yield minimum uncertainty projections for the reactor vessel. The resultant fluence levels after application of the 1.13 bias factor should have an associated 1 sigma uncertainty within the $\pm$ 20% limit specified in our recent telephone conversation.

REFERENCES

1 - McElroy, W. N. et al, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test", NUREG/CR-1861, July 1981.

2 - McElroy, W. N. et al, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments, Blind Test, and Physics-Dosimetry Support for the PSF experiments", NUREG/CR-3318, Sept. 1984.

3 - McElroy, W. N. et al, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: 1986 HEDL Summary Annual Report, NUREG/CR-4307, January 1987.

In addition to the information relative to benchmarking of the transport methodology and the resultant uncertainties in fluence projections, the NRC requested a radial traverse of fast neutron flux within the reactor internals and pressure vessel. In Table 3 a radial traverse from the inner side of the core barrel to the outer radius of the pressure vessel is provided. The data were taken from the 500 degree F forward DOT calculation at the 14.5 degree azimuthal position. Since the traverse was taken from the 500 F forward DOT, it should be noted that the core source in the calculation was the burnup weighted average of 20 cycles of operation and included a burnup weighted axial peaking factor of 1.22. Further no temperature corrections were applied to the traverse. Corrections for water temperature variations over plant life would of course change with radial position through the water zones. The data traverse as presented should, however, provide a good characteristic description of the attenuation afforded by the reactor internals structures.

ATTACHMENT A

TABLE 3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF RADIAL POSITION ALONG A 14.5 DEGREE AZIMUTHAL TRAVERSE [DATA TAKEN FROM THE 500 F FORWARD DOT CALCULATION]

RADIUS (cm)	FLUX (n/cm2-sec)	RADIUS (cm)	FLUX (n/cm2-sec)
117.48	1.02E+12	139.83	3.26E+10
117.79	Barrel IR	139.97	Vessel IR
118.05	9.74E+11	140.14	3.20E+10
118.56	9.14E+11	140.60	3.07E+10
119.07	8.39E+11	141.34	2.84E+10
119.57	7.55E+11	142.22	2.56E+10
120.08	6.63E+11	143.10	2.28E+10
120.33	Barrel OR	143.99	2.02E+10
120.79	5.56E+11	144.87	1.79E+10
121.72	4.64E+11	145.75	1.58E+10
122.64	3.98E+11	146.63	1.39E+10
123.56	3.49E+11	147.51	1.22E+10
124.50	3.11E+11	148.39	1.08E+10
125.45	2.85E+11	149.27	9.45E+09
126.22	2.75E+11	150.15	8.29E+09
126.51	Thermal Sh. IR	151.04	7.26E+09
126.89	2.67E+11	151.92	6.35E+09
127.64	2.52E+11	152.80	5.55E+09
128.39	2.31E+11	153.68	4.85E+09
129.14	2.09E+11	154.56	4.23E+09
129.89	1.88E+11	155.44	3.67E+09
130.64	1.68E+11	156.32	3.19E+09
131.39	1.48E+11	157.21	2.75E+09
132.14	1.31E+11	158.09	2.35E+09
132.89	1.14E+11	158.97	1.99E+09
133.64	9.75E+10	159.85	1.66E+09
134.39	8.11E+10	159.41	Vessel OR
134.76	Thermal Sh. OR		
134.77	7.24E+10		
135.15	6.60E+10		
135.89	5.57E+10		
136.63	4.80E+10		
137.19	4.32E+10		
137.58	4.05E+10		
137.96	3.81E+10		
138.35	3.62E+10		
138.73	3.47E+10		
139.12	3.37E+10		
139.50	3.31E+10		
139.69	Clad IR		

ATTACHMENT B

Table 1

Fluence Distribution for Beltline Materials

Peak Fluence at End of Cycle 21 2.58 e19 n/cm2

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	0.362	0.378	0.387	0.365	0.323	0.272	0.221	0.179	0.159	0.159
	20 to 30	1.242	1.298	1.329	1.252	1.108	0.934	0.757	0.615	0.546	0.546
	30 to 40	1.959	2.047	2.095	1.973	1.747	1.473	1.194	0.970	0.861	0.861
	40 to 50	2.234	2.334	2.389	2.251	1.992	1.680	1.362	1.106	0.982	0.982
% of Height	50 to 60	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
	60 to 70	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
	70 to 80	2.412	2.521	2.580	2.430	2.152	1.814	1.471	1.195	1.060	1.060
	80 to 90	2.369	2.475	2.534	2.387	2.113	1.781	1.444	1.173	1.041	1.041
	90 to 100	2.347	2.453	2.510	2.365	2.094	1.765	1.431	1.162	1.032	1.032
Circ Weld		2.147	2.243	2.2[illegible]6	2.163	1.915	1.614	1.309	1.063	0.944	
Lower Plate											35 to 40
	0 to 10	2.147	2.243	2.296	2.163	1.915	1.614	1.309	1.063	0.944	1.063
% of Height	10 to 20	1.689	1.764	1.806	1.701	1.506	1.270	1.029	0.836	0.742	0.836
	20 to 30	0.975	1.018	1.042	0.982	0.869	0.733	0.594	0.483	0.428	0.483
	30 to 40	0.169	0.176	0.181	0.170	0.151	0.127	0.103	0.084	0.074	0.084

Table 2

Mean Delta RTNDT Distribution for Beltline Materials

Peak Fluence at End of Cycle 21 2.58 e19 n/cm2

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	114	117	118	114	107	97	86	77	72	167
	20 to 30	205	208	210	206	197	185	169	152	143	238
	30 to 40	233	236	238	234	227	216	202	188	179	267
	40 to 50	242	245	246	242	234	224	211	197	189	275
% of Height	50 to 60	245	248	250	246	238	227	214	201	192	278
	60 to 70	245	248	250	246	238	227	214	201	192	278
	70 to 80	247	250	252	248	239	229	216	203	194	280
	80 to 90	246	249	251	246	238	228	215	201	193	279
	90 to 100	245	248	250	246	238	227	214	201	192	278
Circ Weld		324	326	327	324	317	306	293	280	273	
Lower Plate											35 to 40
	0 to 10	319	322	324	320	312	302	289	274	266	280
% of Height	10 to 20	304	307	308	305	298	287	272	256	247	265
	20 to 30	268	271	273	269	259	246	230	214	205	230
	30 to 40	154	156	157	155	150	144	138	133	131	137

Note: To determine the reference temperature, an initial temperature of 30F for plates and 10F for welds must be added to these mean reference temperatures.

Table 3

Fluence Distribution for Beltline Materials

Peak Fluence at End of Cycle 22 2.74 e19 n/cm2

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	0.384	0.402	0.411	0.387	0.343	0.289	0.234	0.190	0.169	0.169
	20 to 30	1.319	1.379	1.411	1.329	1.177	0.992	0.804	0.653	0.580	0.580
	30 to 40	2.080	2.174	2.225	2.096	1.856	1.564	1.268	1.030	0.914	0.914
	40 to 50	2.372	2.479	2.537	2.390	2.116	1.784	1.446	1.175	1.043	1.043
% of Height	50 to 60	2.493	2.605	2.666	2.511	2.223	1.874	1.520	1.234	1.096	1.096
	60 to 70	2.493	2.605	2.666	2.511	2.223	1.874	1.520	1.234	1 096	1.096
	70 to 80	2.562	2.677	2.740	2.581	2.285	1.926	1.562	1.269	1.126	1.126
	80 to 90	2.516	2.629	2.691	2.535	2.244	1.892	1.534	1.246	1.106	1.106
	90 to 100	2.493	2.605	2.666	2.511	2.223	1.874	1.520	1.234	1.096	1.096
Circ Weld		2.280	2.383	2.439	2.297	2.034	1.714	1.340	1.129	1.002	
Lower Plate											35 to 40
	0 to 10	2.280	2.383	2.439	2.297	2.[illegible]4	1.714	1.390	1.129	1.002	1.129
% of Height	10 to 20	1.793	1.874	1.918	1.807	1.600	1.348	1.093	0.888	0.788	0.888
	20 to 30	1.035	1.081	1.107	1.043	0.923	0.778	0.631	0.513	0.455	0.513
	30 to 40	0.179	0.187	0.192	0.181	0.160	0.135	0.109	0.089	0.079	0.089

Table 4

Mean Delta RTNDT Distribution for Beltline Materials

Peak Fluence at End of Cycle 22 2.74 e19 n/cm2

		AZIMUTHAL VARIATION									Axial Welds
		0 to 5	5 to 10	10 to 15	15 to 20	20 to 25	25 to 30	30 to 35	35 to 40	40 to 45	40 to 45
Upper Plate											
	10 to 20	118	121	123	118	111	100	89	79	74	170
	20 to 30	209	212	213	210	201	189	173	157	148	242
	30 to 40	237	240	242	238	230	220	207	192	183	271
	40 to 50	246	249	251	246	238	228	215	201	193	279
% of Height	50 to 60	249	253	254	250	241	231	218	205	196	282
	60 to 70	249	253	254	250	241	231	218	205	196	282
	70 to 80	251	255	257	252	243	232	220	207	198	284
	80 to 90	250	253	255	251	242	231	219	205	197	283
	90 to 100	249	253	254	250	241	231	218	205	196	282
Circ Weld		327	330	331	327	320	310	297	284	277	
Lower Plate											35 to 40
	0 to 10	323	326	328	324	316	305	293	279	270	284
% of Height	10 to 20	308	311	312	309	301	291	276	261	252	269
	20 to 30	272	276	277	273	264	251	234	218	210	234
	30 to 40	157	159	160	157	152	146	140	135	132	139

Note: To determine the reference temperature, an initial temperature of 30F for plates and 10F for welds must be added to these mean reference temperatures.