ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

by

E. B. Norris

FINAL REPORT

SwRI Project 02-4884-001

to

Tennessee Valley Authority
505 Edney Building
Chattanooga, Tennessee 37402

August 1978

~~8310280149~~

SOUTHWEST RESEARCH INSTITUTE

SAN ANTONIO CORPUS CHRISTI HOUSTON

SOUTHWEST RESEARCH INSTITUTE
Post Office Drawer 28510, 6220 Culebra Road
San Antonio, Texas 78284

ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

by
E. B. Norris

FINAL REPORT
SwRI Project 02-4884-001

to

Tennessee Valley Authority
505 Edney Building
Chattanooga, Tennessee 37402

August 1978

Approved:

U. S. Lindholm, Director
Department of Materials Sciences

ABSTRACT

The vessel wall neutron dosimeter capsule from Browns Ferry Unit 1 has been analyzed. The results indicate that the peak value of fast neutron flux incident on the reactor vessel wall is 1.24×10^9 $cm^{-2} \cdot sec^{-1}$, $E > 1$ MeV. Although this results in a lifetime neutron fluence of 1.56×10^{18} cm^{-2}, nearly four times that predicted in the FSAR, it is less than the design limit of 1.0×10^{19} cm^{-2} for 40 years of operation.

Based on a conservative estimate of the neutron embrittlement response of the core beltline materials, the increase in the reference nil ductility temperature may exceed 100 F by the end of the design life of the Browns Ferry Unit 1 vessel. The bases for selecting a capsule removal schedule in accordance with Appendix H of 10CFR50 are discussed.

TABLE OF CONTENTS

I. SUMMARY OF RESULTS AND CONCLUSIONS

The results of the analysis of the Browns Ferry Unit 1 vessel wall dosimeter indicate that the peak fast neutron flux ($E > 1$ MeV) at full power during core cycle 1 was 1.24×10^{9} $cm^{-2} \cdot sec^{-1}$. As a result, a 40-year design life fast neutron fluence of 1.56×10^{18} cm^{-2} is predicted, nearly four times the calculated design life fluence given in the Final Safety Analysis Report (FSAR), but considerably less than the FSAR design limit of 1.0×10^{19} cm^{-2}. Utilizing the radiation damage trend curve in the FSAR, the increase in the minimum reactor pressurization temperature over the design life is projected to be approximately 50 F.

However, it is possible that variations in the chemistries, particularly the copper content, of the Browns Ferry Unit 1 pressure vessel beltline materials may result in sensitivities to neutron radiation embrittlement different from the response curve given in the FSAR. Using the 0.3% Cu RT_{NDT} adjustment curve in Regulatory Guide 1.99(1)*, the total shift might reach 105 F at the vessel wall I.D. and 88 F at the vessel wall 1/4t by the end of the 40-year design life of Browns Ferry Unit 1 pressure vessel.

The capsule removal schedule necessary to meet the requirements of 10CFR50(2), Appendix H, depends on the value of the adjusted RT_{NDT} at the end of the design life of the reactor vessel. If the initial RT_{NDT} of the Browns Ferry Unit 1 pressure vessel beltline material was higher than 12 F, the first specimen material surveillance capsule should be removed after six full power years of operation.

---

* Superscript numbers refer to the list of references at the end of the text.

II. INTRODUCTION

The Browns Ferry Nuclear Plant operated by the Tennessee Valley Authority (TVA), consists of three 1065 Mwe (3293 Mwt) Boiling Water Reactor (BWR) units built by General Electric Company (GE). GE provided each unit with a pressure vessel steel surveillance program which consists of baseline Charpy V-notch specimens (base metal, weld metal and heat-affected zone), baseline tensile specimens (base metal, weld metal and heat-affected zone), a vessel wall dosimeter capsule, and three surveillance capsule baskets containing Charpy V-notch and tensile specimens. The latter two items were installed in the three Browns Ferry vessels prior to startup.

The surveillance program is described in detail in NEDO-10115.(3)* Because of the low level of fast neutron flux density at the vessel wall predicted by design calculations, the first surveillance capsules containing mechanical test specimens are not scheduled for removal until four years of operation have accrued. However, the vessel wall dosimeter capsules are scheduled for removal at the first refuelling to provide a check on the design flux calculations. This report describes the results obtained from the testing and analysis of the contents of the vessel wall neutron dosimeter capsule from Unit 1.

III. EVALUATION OF VESSEL WALL NEUTRON DOSIMETER CAPSULE

The vessel wall neutron dosimeter capsule was removed from the Browns Ferry Nuclear Plant Unit 1 vessel during a refuelling outage which began on September 13, 1977, at the end of core cycle 1. This capsule, shown in Figure 1, contained three each pure copper and pure iron dosimeter wires. The nuclear reactions of interest for these wires are:

$$^{63}Cu(n,\alpha)^{60}Co$$

$$^{54}Fe(n,p)^{54}Mn$$

The capsule was shipped to the Southwest Research Institute (SwRI) laboratories the week of October 24, 1977, in a cask supplied by SwRI. The capsule was opened in one of the hot cells at the SwRI Radiation Laboratory with a hand hacksaw. This could be done because of the low level of activity exhibited by the capsule. The contents were examined and visually identified as either iron or copper.

The dosimeter wires were prepared for analysis by weighing on a precision laboratory balance. The number of target atoms per mg, N_o, was computed for each wire as follows:

$$N_o = \frac{N \cdot c}{A} \times 10^{-3} \qquad (1)$$

where: N = 6.02×10^{23} nuclei per gm atom;

c = weight fraction of detector isotope in detector specimen;

A = atomic weight of detector element, gm.

The absolute activities of the dosimeter wires were measured with a NaI(Th) scintillation detector and an NDC 2200 multichannel analyzer. The experimental efficiency, Eff(E), of the system was determined on the day of

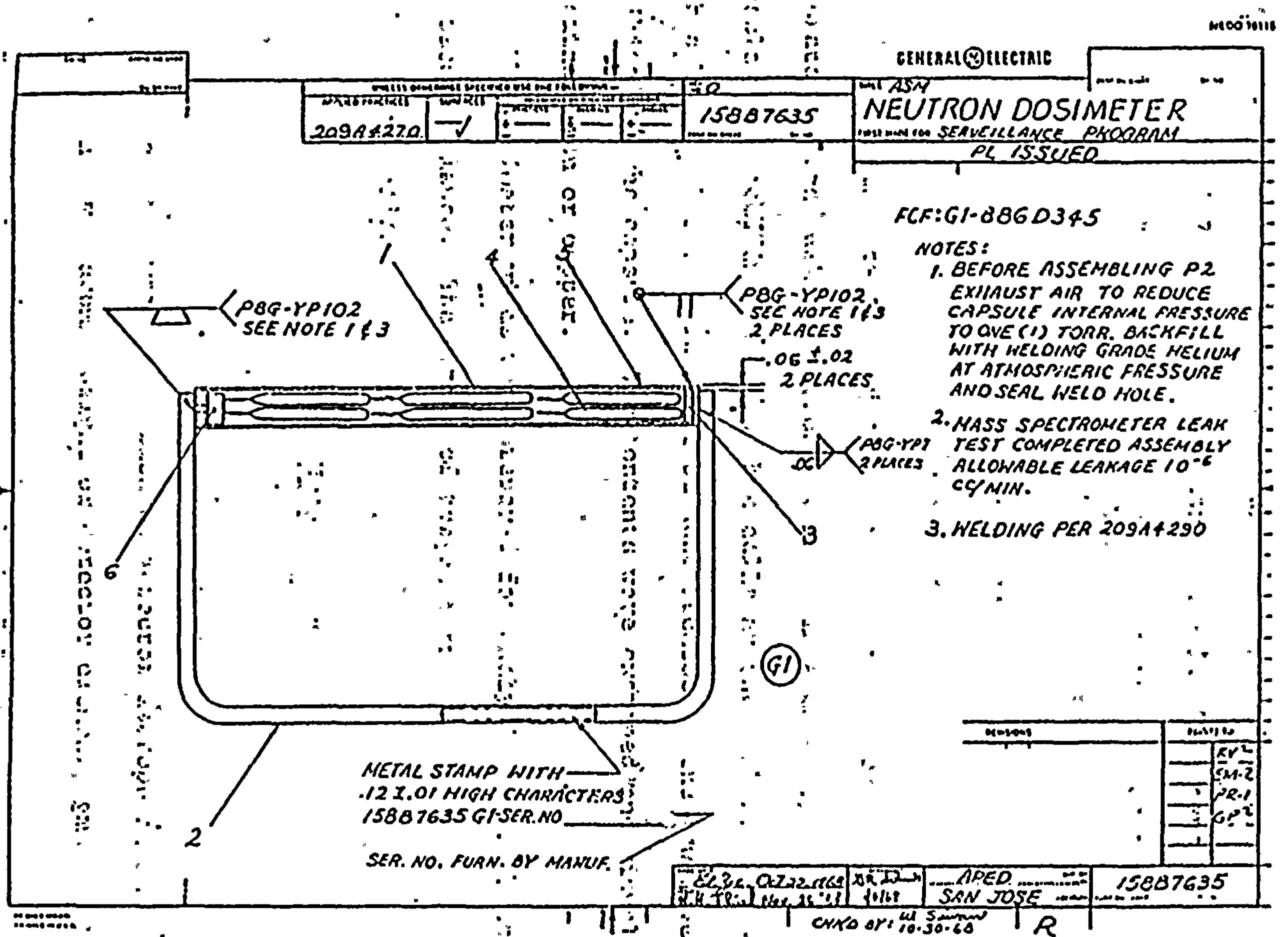


FIGURE 1. VESSEL WALL NEUTRON DOSIMETER

counting for each photopeak of interest, 842 KeV for ^{54}Mn and 1173 KeV for ^{60}Co, with ^{54}Mn and ^{60}Co isotopic standards traceable to the U.S. Bureau of Standards. The counting system and techniques have been previously checked against two other laboratories, see Table I.

The specific activity (dps/mg) of each dosimeter wire at time of reactor shutdown, A(TOR), was computed as follows:

$$A(TOR) = \frac{\text{Total counts under photopeak of energy E less "background"}}{Eff(E) \cdot \tau \cdot w \cdot P \cdot \exp - \lambda t_1} \times \frac{T(E)_s}{T(E)_u} \quad (2)$$

where:

τ	=	counting time, sec;
w	=	weight of wire, mg;
P	=	peak-to-total ratio;
λ	=	disintegration rate, day^{-1};
t_1	=	elapsed time between TOR and counting date, days.
$T(E)_s$	=	intrinsic efficiency factor for the standard source counting geometry;
$T(E)_u$	=	intrinsic efficiency factor for the unknown source counting geometry.

In this program, $T(E)_s/T(E)_u$ was equal to unity because the standard and unknown sources were counted using the same geometry.

The weights, counting rates, and specific activities determined for each dosimeter wire are summarized in Table II. The last column in Table II lists the saturated activities, A_s, of the dosimeter wires computed for the full power level of 3293 Mwt as follows:

$$\frac{A(TOR)}{A_s} = \sum_{m=1}^{m=n} (1 - \exp - \lambda T_m)(\exp - \lambda t_m)$$

where:

m	=	operation period;
T_m	=	equivalent operating time at selected power level for the m^{th} period, days;
t_m	=	elapsed time from the end of the mth period to TOR, days.

The values of T_m and t_m were determined by dividing the Unit 1 plant operation into 19 operating periods, as summarized in Table III.

TABLE I

RESULTS OF INTERLABORATORY GAMMA-COUNTING PROGRAM

Sample Identification	Isotope	Assumed Half-Life	Activity at TOR (dps/mg) SwRI	Other
Top (Co-Cd)	^{60}Co	1913 d	2.69×10^7	2.68×10^7 (a)
Bot (Co-Cd)	^{60}Co	1913 d	2.67×10^7	2.48×10^7 (a)
Top (Co)	^{60}Co	1913 d	6.03×10^7	5.83×10^7 (a)
Bot (Co)	^{60}Co	1913 d	6.23×10^7	5.93×10^7 (a)
2056	^{60}Co	1913 d	1.41×10^7	1.37×10^7 (b)
2062	^{60}Co	1913 d	5.57×10^6	5.20×10^6 (b)
R9	^{54}Mn	312 d	1.30×10^4	1.32×10^4 (a)
R13	^{54}Mn	312 d	1.24×10^4	1.29×10^4 (a)
R16	^{54}Mn	312 d	1.21×10^4	1.23×10^4 (a)

(a) BNWL
(b) WNES

TABLE II

RESULTS OF ACTIVATION ANALYSES OF DOSIMETER WIRES EXPOSED IN BROWNS FERRY UNIT 1 VESSEL FROM 10/1/73 THROUGH 9/13/77

Isotope	Foil	Weight (mg)	Count Rate (dpm)	A(TOR)[(a)] (dps/mg)	A_s[(b)] (dps/mg)
^{60}Co	Cu-1	470.9	1.044×10^5	3.697	24.33
^{60}Co	Cu-2	496.0	1.115×10^5	3.745	24.65
^{60}Co	Cu-3	475.5	1.031×10^5	3.614	23.78
				Average	24.25
^{54}Mn	Fe-1	158.3	5.790×10^5	60.96	132.9
^{54}Mn	Fe-2	158.7	5.669×10^5	59.53	129.8
^{54}Mn	Fe-3	158.5	5.231×10^5	55.00	119.9
				Average	127.5

(a) Specific activity at time of reactor shutdown, 9/13/77. Disintegration rates are subject to a ±3% (1 S%) measurement uncertainty.[(4,5)]

(b) Saturated activity at the 3293 Mwt power level.

TABLE III

OPERATIONS SUMMARY - BROWNS FERRY NUCLEAR PLANT, UNIT 1

Operating Period (m)	Dates Start	Dates Stop	Operating Days	Shutdown Days	Reactor Power (MWDth)	Equivalent[(a)] Operating Days (T_m)	Decay Time in Days (t_m)
1	10-01-73	10-08-73	8	-	977	0.30	1436
	10-09-73	10-11-73	-	3	-	-	-
2	10-12-73	11-09-73	29	-	18,450	5.60	1404
	11-10-73	11-13-73	-	4	-	-	-
3	11-13-73	12-11-73	28	-	30,793	9.35	1372
	12-12-73	12-23-73	-	12	-	-	-
4	12-24-73	01-18-74	26	-	29,789	9.05	1334
	01-19-74	01-22-74	-	4	-	-	-
5	01-23-74	02-12-74	21	-	37,843	11.49	1309
	02-13-74	02-17-74	-	5	-	-	-
6	02-18-74	02-28-74	11	-	21,343	6.48	1293
	03-01-74	03-07-74	-	7	-	-	-
7	03-08-74	04-03-74	27	-	50,891	15.45	1259
	04-04-74	04-11-74	-	8	-	-	-
8	04-12-74	05-07-74	26	-	40,578	12.32	1225
	05-08-74	05-25-74	-	18	-	-	-
9	05-26-74	06-21-74	27	-	57,746	17.54	1180
	06-22-74	06-28-74	-	7	-	-	-
10	06-29-74	09-19-74	83	-	220,525	66.97	1090
	09-20-74	10-03-74	-	14	-	-	-
11	10-04-74	11-18-74	46	-	128,491	35.98	1030
	11-19-74	11-22-74	-	4	-	-	-
12	11-23-74	02-02-75	72	-	205,316	62.35	954
	02-03-75	02-08-75	-	6	-	-	-
13	02-09-75	02-26-75	18	-	40,664	12.35	930
	02-27-75	03-05-75	-	7	-	-	-
14	03-06-75	03-22-75	17	-	44,554	13.53	906
	03-23-75	09-13-76	-	541	-	-	-
15	09-14-76	10-13-76	30	-	14,215	4.32	335
	10-14-76	10-15-76	-	2	-	-	-
16	10-16-76	11-17-76	33	-	53,790	16.33	300
	11-18-76	11-18-76	-	1	-	-	-
17	11-19-76	01-04-77	47	-	110,042	33.42	252
	01-05-77	01-09-77	-	5	-	-	-
18	01-10-77	04-21-77	102	-	263,890	80.14	145
	04-22-77	04-22-77	-	1	-	-	-
19	04-23-77	09-13-77	144	-	373,769	113.50	0
				Totals	1,733,666	526.47[(b)]	

(a) At 3293 Mwt
(b) 526.47 days = 4.5487×10^7 seconds

IV. CALCULATION OF NEUTRON FLUX DENSITY AND FLUENCE

The energy-dependent neutron flux density, ϕ ($cm^{-2} \cdot sec^{-1}$), the spectrum-averaged activation cross-section, $\bar{\sigma}$ (cm^2), and the saturated activity A_S, of each dosimeter wire are related as follows:

$$\phi = \frac{A_S}{N_O \bar{\sigma}} . \tag{4}$$

In the early days of nuclear pressure vessel surveillance activity, the value of $\bar{\sigma}$ was based on the assumption of a fission spectrum energy distribution for the neutron flux at the surveillance capsule location. It was recognized that this assumption was probably in error, but since correlations between neutron exposure and vessel steel mechanical properties were empirical, the fission spectrum assumption was useful. However, as methods of analysis were improved, the use of calculated neutron spectra has increased and is now permitted by NRC Regulatory Guide 1.99(2) for application to reactor pressure vessel wall locations.

The neutron flux energy and spatial distribution were calculated for the Browns Ferry Unit 1 pressure vessel with the DOT 3.5 two-dimensional discrete ordinates transport code, a 22-group neutron cross section library, a P_1 expansion of the scattering matrix and an S_8 order of angular quadrature. An R-θ calculation was made for a horizontal plane perpendicular to the vertical axis of the core, and an R-Z calculation was made for a vertical plane through the axis of the core and the location of the vessel wall dosimeter. A one-eighth segment, shown in Figure 2, was taken to be representative of the R-θ geometry because of the symmetry involved. The boundaries of the core, core shroud, jet pumps, and vessel wall were described in R-θ coordinates. The

LOCATION	AZIMUTH	REMOVAL PERIOD YEARS
Wall	30°	4
Wall	120°	12
Wall	300°	32
Wall	30°	1

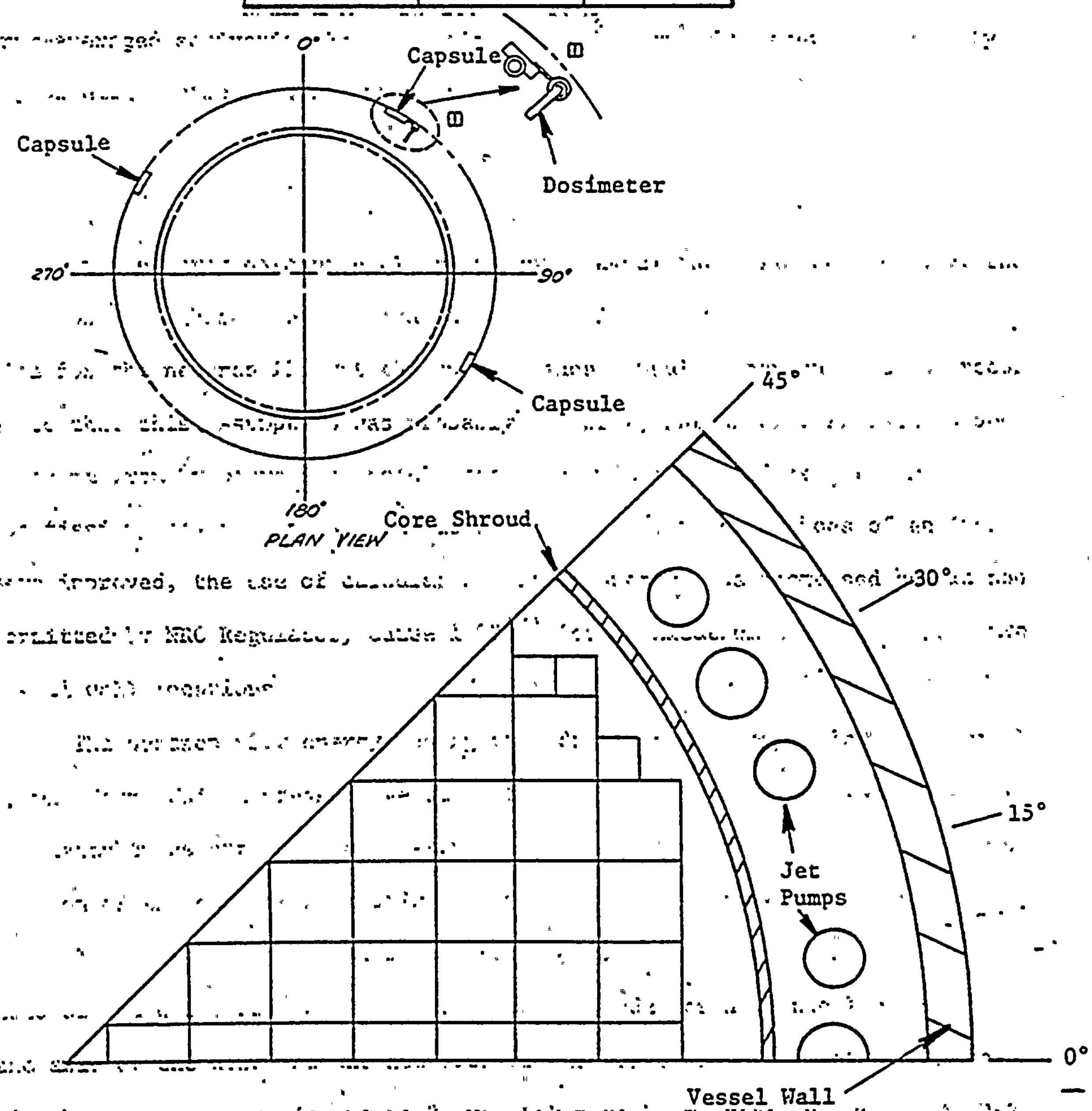


FIGURE 2. ONE-EIGHTH SEGMENT FOR FLUX CALCULATIONS AND LOCATION OF SURVEILLANCE CAPSULES

core was subdivided into two regions, an inner region with one-sixth of the control rods inserted and having a 0.4368 void fraction, and an outer region with all control rods withdrawn and having a 0.4543 void fraction. The core materials within each region were homogenized over their respective areas. Stainless steel was assumed to be 18% Cr, 8% Ni, and 74% Fe, and the pressure vessel was assumed to be 98% Fe. The coolant outside of the core was assumed to have no voids. An average power distribution in the core was derived from data sheets supplied by TVA. The same assumptions were used in modeling the R-Z geometry.

Both of these calculations provide information on the neutron energy spectrum at the vessel wall neutron dosimeter capsule location. In addition, the R-θ calculation provides information on the radial and azimuthal variation in neutron flux, and the R-Z calculation predicts the radial and vertical distribution of the neutron flux. By combining these factors, the relationship between neutron flux at the surveillance capsule locations and that at the point of maximum neutron flux incident on the vessel (I.D. lead factors) can be derived.

The neutron spectrum at the vessel wall dosimeter location, as determined with the R-θ model, and the group-averaged cross-sections for the dosimeter reactions of interest are given in Table IV. The spectrum-averaged cross sections, $\bar{\sigma}$, were determined from the relationship:

$$\bar{\sigma}\ (E > 1\ \text{MeV}) = \frac{\sum_{1.11}^{10.0} \sigma(E)\ \phi(E)\ dE}{\sum_{1.0}^{10.0} \phi(E)\ dE} \quad . \tag{5}$$

TABLE IV

DETERMINATION OF REACTION CROSS SECTION FOR IRON DOSIMETERS

Energy Range (MeV)	Normalized Neutron Flux, $\phi(E)$	$^{54}Fe(n,p)^{54}Mn$ Cross Section, $\sigma_{Fe}(E)$ (barns)	$^{63}Cu(n,\alpha)^{60}Co$ Cross Section, $\sigma_{Cu}(E)$ (barns)
8.18 – 10.0	.0337	.581	.0380
6.36 – 8.18	.0844	.577	.0144
4.96 – 6.36	.1230	.491	.0023
4.06 – 4.96	.0985	.354	.00025
3.01 – 4.06	.1117	.185	.00010
2.35 – 3.01	.1439	.078	.00006
1.83 – 2.35	.1224	.023	.00002
1.11 – 1.83	.2084	.0014	.00002
1.00 – 1.11	.0740	—	.00001
		$\bar{\sigma}_{Fe}$ = .198[(a)]	$\bar{\sigma}_{Cu}$ = .00283[(a)]

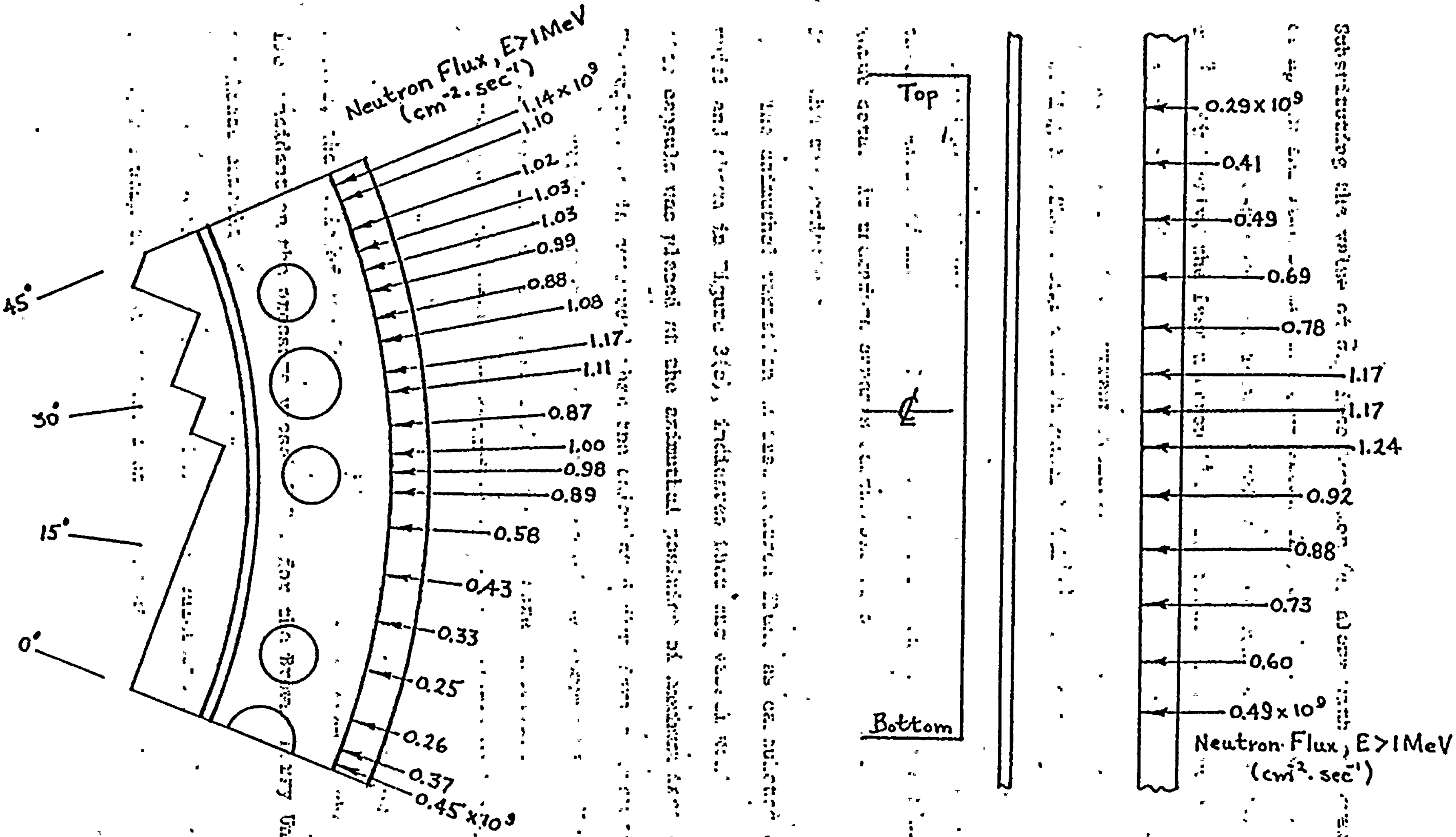
(a) Spectrum-averaged cross sections are subject to a ±15% (1 S%) uncertainty.[(6)]

Substituting the value of $\bar{\sigma}_{Fe}$ into Equation (4) along with the average value of A_s for the iron dosimeters (see Table II), the fast neutron flux (E > 1 MeV) at the vessel wall dosimeter location is calculated to be 1.03×10^9 cm^{-2}. Similarly, the fast neutron flux determined from the copper dosimeters is 1.31×10^9 cm^{-2}. The discrepancy between these two values is largely a result of uncertainties in current evaluated energy-dependent cross sections.[4,5] According to ASTM Recommended Practice E 482[6], errors as large as ±7% (1 S%) in the determination of disintegration rates and ±15% (1 S%) in spectrum-weighted group-averaged cross sections can be encountered, which results in a combined error of ±16.5% (1 S%) for the calculation of neutron flux from the input data. It therefore appears reasonable to average the results obtained from the two dosimeters.

The azimuthal variation in fast neutron flux, as calculated with the R-θ model and shown in Figure 3(a), indicates that the vessel wall neutron dosimeter capsule was placed at the azimuthal position of maximum fast neutron flux. Therefore, it is concluded that the calculated flux derived from the analysis of the dosimeter wires is a direct measure of the maximum fast flux incident on the pressure vessel opposite the vertical core centerline.

However, the axial flux distribution, as calculated with the R-Z model and shown in Figure 3(b), indicates that the peak fast neutron flux is 6% higher at a position 20 cm below the capsule location. Therefore, the lead factor, the ratio of the fast flux at the capsule location to the peak fast flux incident on the pressure vessel I.D., for the Browns Ferry Unit 1 surveillance capsules is calculated to be 0.94.

Based on the results of the DOT 3.5 calculations and the dosimetry results, the peak fast flux incident on the Browns Ferry Unit 1 vessel during



(a) Azimuthal Distribution at Core Vertical Centerline

(b) Vertical Distribution at 30° Azimuthal Position

FIGURE 3. CALCULATED DISTRIBUTIONS OF FAST NEUTRON FLUX INCIDENT ON THE BROWNS FERRY UNIT NO. 1 PRESSURE VESSEL I.D.

the first core cycle is calculated to be 1.24×10^9 $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. Therefore, the neutron fluence per effective full power year (EFPY) is 3.91×10^{16} cm^{-2}, E > 1 MeV. Assuming 100% availability over the 40-year design life of the plant, the design life neutron fluence received by the vessel is predicted to be 1.56×10^{18} cm^{-2}, E > 1 MeV.

The neutron flux is moderated as it moves from the core and penetrates the pressure vessel wall. The radial dependence of the fast neutron flux obtained from the DOT 3.5 analyses is shown as the solid curve in Figure 4. The dashed curve through the pressure vessel wall represents a conservative estimate of the fast flux attenuation by steel which is acceptable to the NRC.[7]

Since the pressure-temperature limits for reactor operation and testing are based on requirements of the ASME Boiler and Pressure Vessel Code[8], the fluence at the 1/4t and 3/4t positions within the pressure vessel wall are of specific interest. Utilizing the conservative estimate of the attenuation of fast neutron flux by a pressure vessel wall shown by the dashed curve in Figure 4, the predicted flux and fluence values obtained at 1/4t and 3/4t for the 6-5/16-in. Browns Ferry Unit 1 pressure vessel are summarized in Table V.

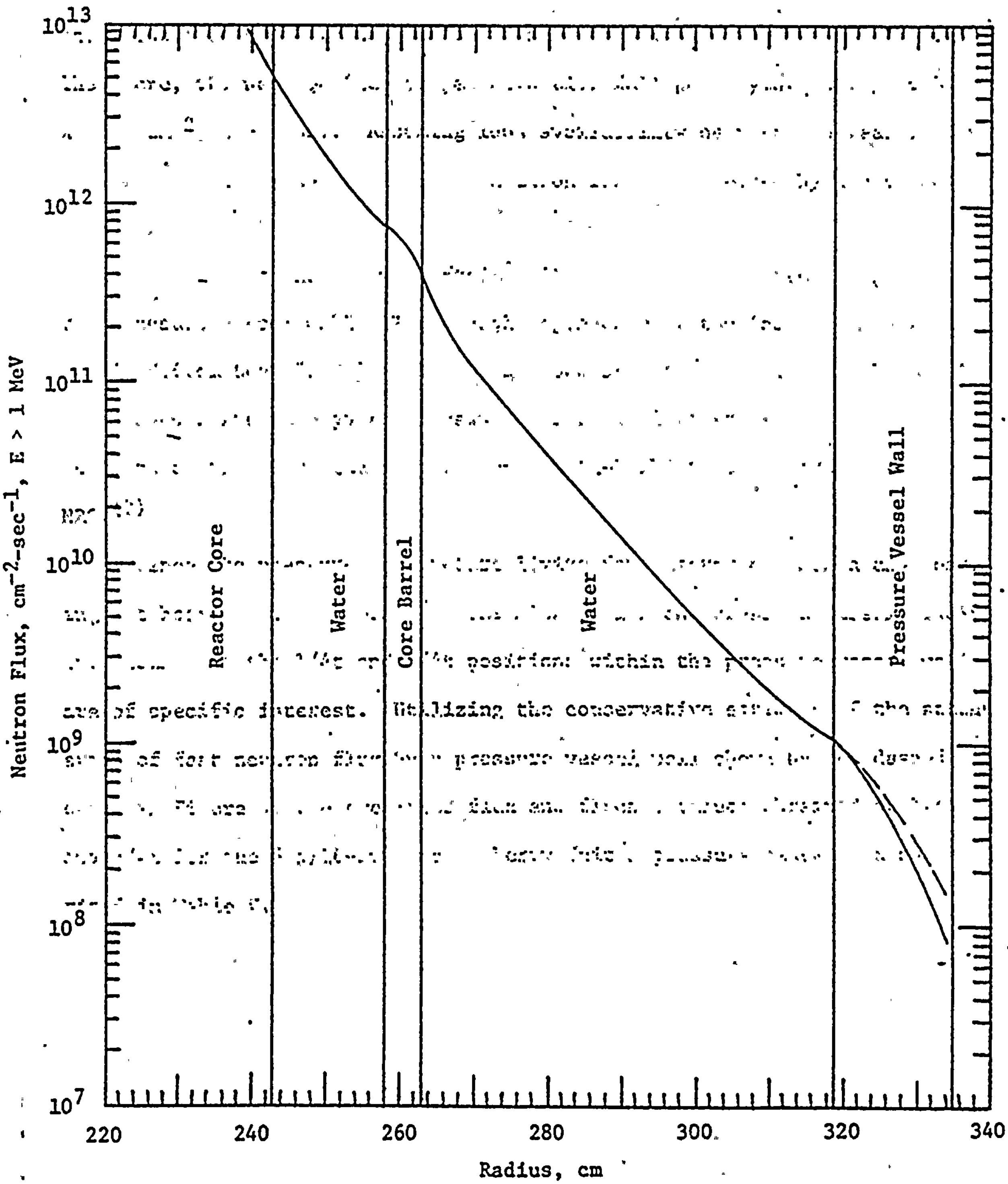


FIGURE 4. CALCULATED NEUTRON FLUX BETWEEN CORE AND PRESSURE VESSEL I.D. NORMALIZED TO THE VESSEL WALL DOSIMETER RESULT

TABLE V

CALCULATED PEAK NEUTRON FLUX[a] AND FLUENCE[a]
FOR BROWNS FERRY UNIT 1 PRESSURE VESSEL WALL

Vessel Wall Location	Relative Fast Neutron Flux	Fast Neutron Flux Density ($cm^{-2} \cdot sec^{-1}$)	Fast Neutron Fluence (cm^{-2})		
			1.442 EFPY[b]	4 EFPY	40 EFPY
I.D. Surface	1.00	1.24×10^{9}	5.64×10^{16}	1.56×10^{17}	1.56×10^{18}
1/4t	0.67	8.3×10^{8}	3.8×10^{16}	1.0×10^{17}	1.0×10^{18}
3/4t	0.24	3.0×10^{8}	1.4×10^{16}	3.7×10^{16}	3.7×10^{17}

(a) E > 1 MeV. Calculated flux and fluence values subject to a ±16.5% uncertainty.[6]
(b) End of core cycle 1.

IV. DISCUSSION

The predicted value of the peak neutron fluence (E > 1 MeV) for the Browns Ferry Unit 1 pressure vessel after 40 EFPY of operation is given in the Final Safety Analysis Report (FSAR) as 3.8×10^{17} cm^{-2} (E > 1 MeV). The analysis of the vessel wall dosimeter capsule projects that the peak neutron fluence will be 1.56×10^{18} cm^{-2}, nearly four times the predicted value, but considerably less than the FSAR design limit of 1.0×10^{19} cm^{-2}. A similar trend has been noted in several other BWR plants with which SwRI has been associated. For example, the neutron dosimetry analyses performed on the first capsules removed from the Elk River, LaCrosse, Millstone Point 1, and Pilgrim reactors indicated that the fast neutron flux densities were higher than the design values by factors ranging from 2 to 6.

The estimation of a 40-year neutron fluence from less than two years of operation is a large extrapolation and will be subject to revision at the time of the next capsule removal, currently scheduled after four years of operation. In the meantime, however, the projected peak fast fluence factor of 3.91×10^{16} cm^{-2} per EFPY can be employed to predict the change in the reference nil ductility temperature (RT_{NDT}) as a function of reactor power generation.

The threshold value of neutron fluence for the 550 F embrittlement of ferritic steels is generally taken to lie between 10^{17} and 10^{18} cm^{-2} (E > 1 MeV). The proposed relationship between fast neutron fluence and the change in the RT_{NDT} of the Browns Ferry Unit 1 reactor vessel, as given in Figure 3.6-2 of the FSAR, is reproduced in Figure 5. Added to this figure are (1) an arrow indicating the fast neutron fluence on the vessel I.D. at the end of core cycle 1, and (2) an additional abscissa relating neutron

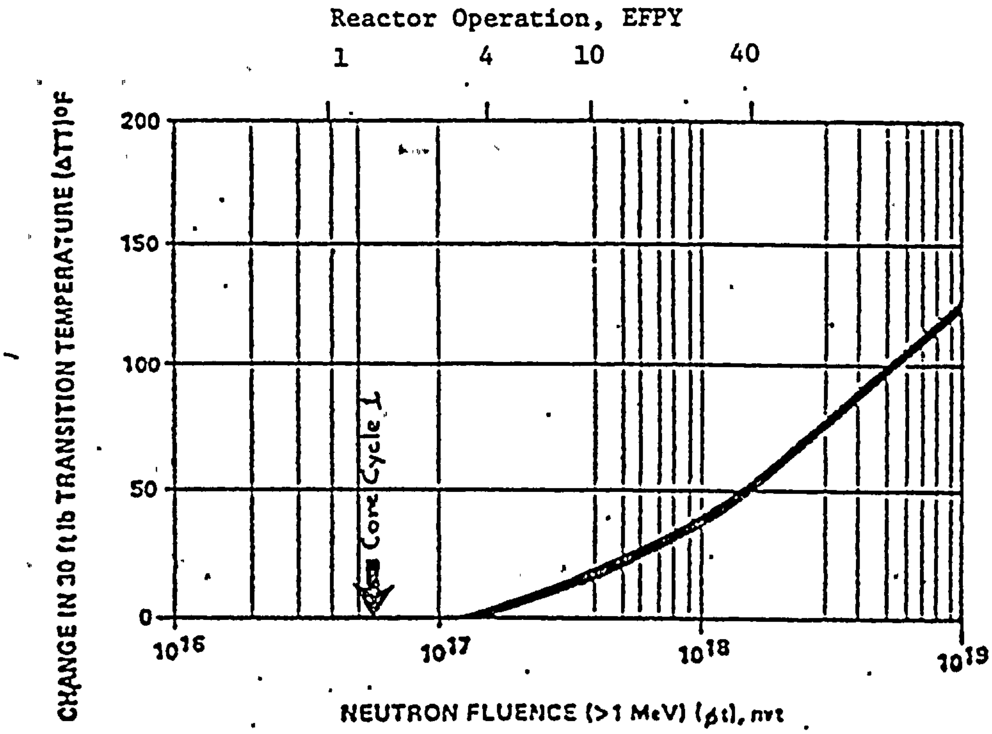


FIGURE 5. VESSEL MATERIAL NEUTRON EMBRITTLEMENT CURVE
FROM BROWNS FERRY UNIT 1 FSAR

fluence and effective full power years as determined from the vessel wall surveillance capsule. Figure 5 indicates that the RT_{NDT} of the Browns Ferry Unit 1 pressure vessel will begin to increase after an exposure of 1.35×10^{17} cm^{-2}, E > 1 MeV. The I.D. surface would reach this fluence in about four EFPY, but it would require over five EFPY of operation to reach this fluence at the 1/4t location in the pressure vessel wall. Also, the predicted shift in RT_{NDT} at the I.D. surface after 40 EFPY is less than 50 F above the baseline (unirradiated) value.

The neutron embrittlement sensitivity curve from the FSAR (Figure 5) corresponds closely with the RT_{NDT} adjustment curve of Regulatory Guide 1.99[2] for 0.15% Cu and 0.012% P, see Figure 6. However, in a recent response to the NRC[9], TVA submitted information from GE indicating that the copper contents of plates might be as high as 0.2% and those of welds might be as high as 0.3%. Utilizing the 0.3% Cu response curve in Figure 6, the predicted shift in RT_{NDT} of the Browns Ferry Unit 1 vessel at the end of design life would be 110 F at the I.D. surface and 88 F at the 1/4t wall position. Since the capsule lead factors are near unity, one-fourth of the end-of-life fluence should be reached in approximately 10 EFPY.

Section II.3 of Appendix H of 10CFR50[2] describes three cases which govern the surveillance specimen capsule removal schedule. The first case, which applies when the adjusted RT_{NDT} of the reactor vessel steel will not exceed 100 F at the end of the design life, requires that a specimen capsule be removed at one-fourth of the design life. The second and third cases, which apply when the adjusted RT_{NDT} of the reactor vessel steel exceeds 100 F at the end of the design life, requires that the first specimen capsule be removed when the predicted adjustment of the reference temperature is approximately 50 F, or at one-fourth service life, whichever is earlier.

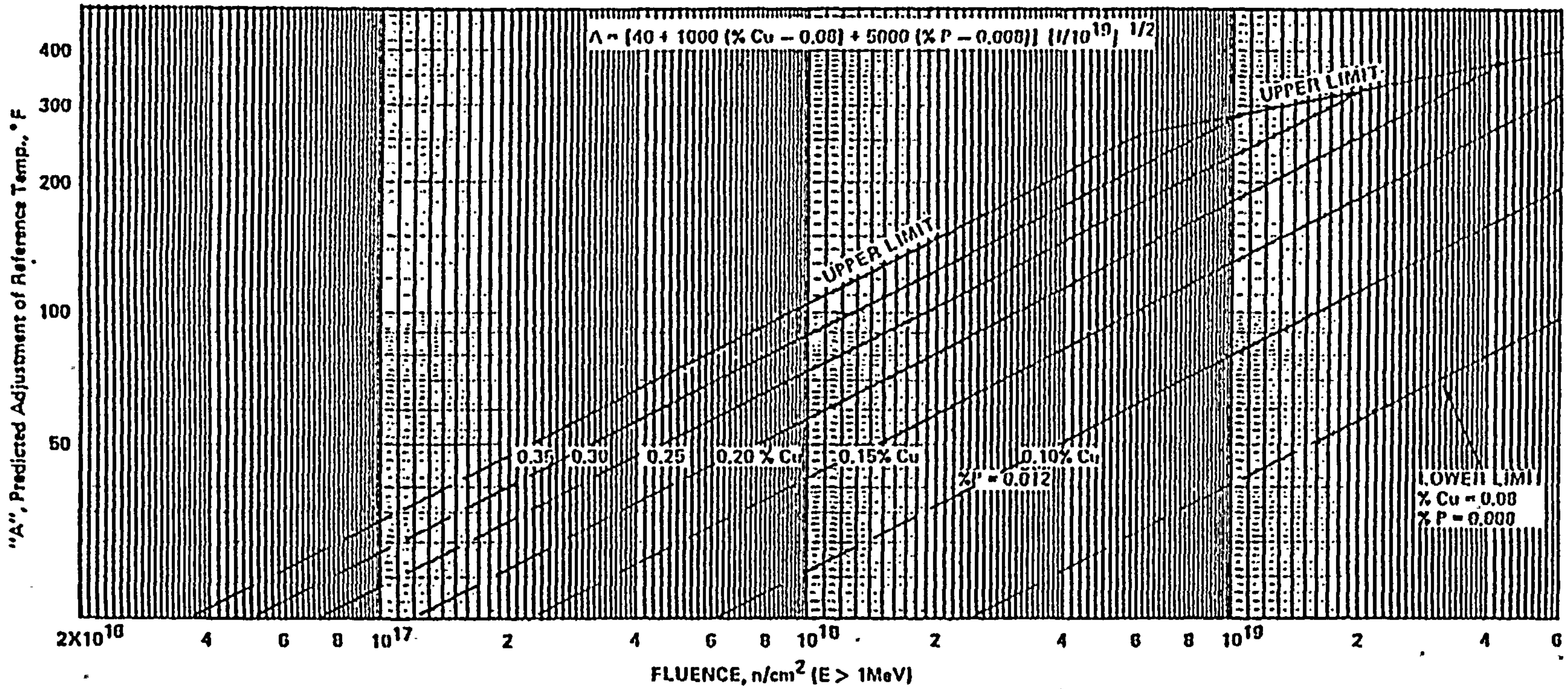


Figure 1 Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure.

(Note: Dashed lines represent GE recommended extrapolations to 20°F for BWR operation.)

FIGURE 6. REFERENCE TEMPERATURE ADJUSTMENT CURVES FROM REGULATORY GUIDE 1.99 (1)

Based on an end-of-design life increase in RT_{NDT} of 88 F determined from the 0.3% Cu response curve in Figure 6 at the 1/4t fluence obtained from the vessel wall dosimeter, the capsule removal schedule necessary to meet the requirements of 10CFR50, Appendix H, are as follows:

Initial RT_{NDT}	End of Design Life RT_{NDT}	Time of First Capsule Removal
≤ (12 F)	≤ 100 F	10 EFPY
> (12 F)	> 100 F	6 EFPY

Since the results of the analysis of the vessel wall dosimeters indicate that the fast neutron flux is higher than that predicted in the FSAR, the current pressure-temperature limits for operation and testing should be reviewed to determine if they are consistent with the projected adjusted values of RT_{NDT} between the first refuelling and the time of removal of the first surveillance capsule. If not, revised pressure-temperature limits should be established in accordance with Section III, Appendix G, of the ASME Boiler and Pressure Vessel Code.(8)

VI. REFERENCES

1. Regulatory Guide 1.99, Revision 1, Office of Standards Development, U.S. Nuclear Regulatory Commission, April 1977.

2. Title 10, Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

3. "Mechanical Property Surveillance of GE BWR Vessels," NEDO-10115, July 1969.

4. ASTM E 523-76, "Standard Method for Measuring Fast-Neutron Flux Density by Radioactivation of Copper," Annual Book of ASTM Standards, Part 45.

5. ASTM E 263-77, "Standard Method for Determining Fast-Neutron Flux by Radioactivation of Iron," Annual Book of ASTM Standards, Part 45.

6. ASTM E 482-76, "Standard Recommended Practice for Neutron Dosimetry for Reactor Pressure Vessel Surveillance," Annual Book of ASTM Standards, Part 45.

7. Telecon, E. B. Norris to Ken Hogue (NRC Staff), January 19, 1977.

8. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-ductile Failure."

9. Letter from J. E. Gilleland, TVA, to A. Schwencer, NRC, regarding Docket Nos. 50-259, 50-260, and 50-296, dated August 23, 1977.