Preliminary Evaluation of Graphite in the FMIT Test Cell

W. C. Morgan

March 1980

**Prepared for
Hanford Engineering Development
Laboratory**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

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PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
Under Contract DE-AC06-76RLO 1830

Printed in the United States of America
Available from
National Technical Information Service
United States Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151

Price: Printed Copy \$_____*: Microfiche \$3.00

*Pages	NTIS
	Selling Price
001-025	\$4.00
026-050	\$4.50
051-075	\$5.25
076-100	\$6.00
101-125	\$6.50
126-150	\$7.25
151-175	\$8.00
176-200	\$9.00
201-225	\$9.25
226-250	\$9.50
251-275	\$10.75
276-300	\$11.00

3 3679 00054 4553

PRELIMINARY EVALUATION OF GRAPHITE
IN THE FMIT TEST CELL

W. C. Morgan

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Prepared for
Hanford Engineering Development Laboratory
under a related services agreement with
the U.S. Department of Energy under Contract
DE-AC06-76RLO 1830, Work Order No. A-32099

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Richland, Washington 99352

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ABSTRACT

A design for the Test Cell of the Fusion Materials Irradiation Test facility includes composite steel-graphite cooling/shielding panels. A graphite damage fluence of about $3.2 \times 10^{20}/\text{cm}^2$ at about 80°C can be expected over 20 years of operation. This review defines the potential problem areas (stored energy and dimensional changes) and recommends approaches to avoid or alleviate those problems. It is concluded that the design is viable and that no insurmountable problems are likely to be encountered.

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INTRODUCTION

Staff of the Ralph M. Parsons Co. have evaluated⁽¹⁾ possible combinations of materials for use as cooling/shielding panels in the test cell of the Fusion Materials Irradiation Test (FMIT) facility, and have recommended (with certain reservations) that the panels be composed of alternate layers of steel and graphite. The primary reservation, in regard to the use of this composite, is a concern over the possible consequences of a build-up of stored energy in the graphite panels.

The purpose of this report is to:

- 1) define the extent of the potential problem with stored energy,
- 2) define other potential problem areas associated with this use of graphite, and
- 3) recommend means of avoiding potential problems.

Most of the data on the behavior of graphite at low irradiation temperatures was accumulated during the 1940's and 1950's. Various irradiation exposure units were used in reporting the data, and many of these units have only a tenuous relationship to the actual damage (changes in properties) produced by the irradiation; therefore, it is necessary to reduce the reported exposure units to a common and meaningful scale.

NEUTRON FLUENCE SCALES

The internationally accepted neutron exposure unit, for comparing the damage produced in graphite irradiated in different neutron flux spectra,⁽²⁻⁴⁾ is the "Equivalent Fission Fluence for Damage in Graphite (Φ_G), which can be defined as follows:

$$\phi_G = \frac{\int_{t_1}^{t_2} \int_0^{\infty} \Sigma_d(E) \phi(E,t) dE dt}{\int_0^{\infty} \Sigma_d(E) \chi(E) dE / \int_0^{\infty} \chi(E) dE} \quad (1)$$

where

$\phi(E,t)$ = absolute flux density spectrum at time t

$\chi(E)$ = ^{235}U fission spectrum

$\Sigma_d(E) = \sigma_s(E) p(E)$.

where

$\sigma_s(E)$ = scattering cross section of carbon

$p(E)$ = (Thompson and Wright⁽⁵⁾ form of) the weighing function.

It is also useful to define another quantity, $\bar{\Sigma}_f$, the spectrum averaged atomic displacement cross section in a ^{235}U fission spectrum:

$$\bar{\Sigma}_f = \int_0^{\infty} \Sigma_d(E) \chi(E) dE / \int_0^{\infty} \chi(E) dE$$

Although more recent forms of the damage cross section, $\Sigma_d(E)$, have been developed to obtain better comparisons between irradiations in fission-reactor and fusion-reactor spectra,^(6,7) the Thompson and Wright (T-W) form of the weighing function will be used herein. Since only a few percent of the neutrons in the shield have energies greater than 1.0 MeV, the results which would be obtained using the other functions should be essentially identical to those obtained using the T-W form.

Table 1 lists the normalized neutron flux in different energy groups, at the front and rear faces of the steel/graphite shield composite, and the damage cross sections⁽³⁾ used to calculate ϕ_G .

TABLE 1. Relative Damage per Neutron

Energy Range	$\Sigma_d(a)$	Front Face		Rear Face	
		Fraction(b) per Group	Relative(c) Damage	Fraction(b) Per Group	Relative(c) Damage
Below 1.0 KeV	-0-	0.281	-0-	0.611	-0-
1.0 KeV to 20 KeV	75	0.214	16	0.219	16
20 KeV to 0.1 MeV	210	0.185	39	0.076	16
0.1 MeV to 1.0 MeV	730	0.292	213	0.064	47
Above 1.0 MeV	725	<u>0.028</u>	<u>20</u>	<u>0.030</u>	<u>22</u>
Sum:		1.0	288	1.0	101

$\bar{\Sigma}_f$ is about $720 \times 10^{-24} \text{ cm}^{2(3,8-10)(a)}$ using the T-W weighting function; therefore:

$$\Phi_G \simeq 0.400 \Phi_f, \text{ and}$$

$$\Phi_G \simeq 0.140 \Phi_r$$

where Φ_f = total neutron fluence at the front of the composite region, and

Φ_r = total neutron fluence at the rear face of the composite shield region.

Using a total neutron flux^(b) of $1.85 \times 10^{12}/\text{cm}^2\text{sec}$ at the front and $8.07 \times 10^{10}/\text{cm}^2\text{sec}$ at the rear face of the shield, and 80% operation during 20 years, we obtain:

(a) In barns (10^{-24} cm^2), see Appendix A.

(b) Personal communication from L. L. Carter, Hanford Engineering Development Lab., February 1980.

(c) (displacements per neutron) $\times 10^{-24}$.

$$\begin{aligned}\phi_G (\text{front}) &= 3.73 \times 10^{20}/\text{cm}^2, \text{ and} \\ \phi_G (\text{rear}) &= 5.70 \times 10^{18}/\text{cm}^2\end{aligned}$$

Hanford MWD/AT

Most of the low-temperature irradiation data from the U.S. has been reported in terms of megawatt days of power generation per ton of uranium adjacent to the test facility (MWD/AT). Nightingale⁽¹¹⁾ reported that one MWD/AT in the water-cooled test hole facilities was equivalent to a fluence of $3.9 \times 10^{17}/\text{cm}^2$, thermal neutrons; using this ratio, the thermal-to-fast ($E > 0.5$ MeV) ratio of 10.9^(a) and the ratio: $\phi(E > 0.183 \text{ MeV}) / \phi(E > 0.498 \text{ MeV}) = 1.276$,^(a) we calculate that $\phi(E > 0.183)$ is about $4.56 \times 10^{16}/\text{cm}^2$ per MWD/AT.

deHalas⁽¹³⁾ reported that $\phi(E > 1.0 \text{ MeV})$ per MWD/AT was measured to be about 3.5×10^{16} ; using this number and the ratio: $\phi(E > 0.183 \text{ MeV}) / \phi(E > 1.05 \text{ MeV}) = 1.481$,^(a) we calculate that $\phi(E > 0.183)$ is about $5.18 \times 10^{16}/\text{cm}^2$ per MWD/AT. Using the average of the two values, derived above, and the relationship: $\phi_G = 1.18 \times \phi(E > 0.183)$,⁽³⁾ we conclude that, for the Hanford water-cooled test facilities, $\phi_G = \text{MWD/AT} \times 5.75 \times 10^{16}/\text{cm}^2$.

BEPO nvt

Most of the low-temperature irradiation data from the UK has been reported in terms of BEPO Equivalent Neutron Dose, BEPO nvt, (i.e., the total neutron fluence in the standard test facility in the BEPO Reactor). Hering⁽¹⁴⁾ compared the damage produced (stored energy accumulation rate and the crystallite lattice change rates) per BEPO nvt and the damage produced in the Hanford water-cooled test facilities. Woods, et al.,⁽¹⁵⁾ had reported that 1 MWD/AT is equal to $6.46 \times 10^{17}/\text{cm}^2$ (total neutron fluence); using this number

(a) Calculated for a similar water-cooled facility in the K-East Reactor at Hanford.⁽¹²⁾

and the average value from Hering's comparison: 1 BEPO nvt produces the same damage as 1.08 nvt (total, Hanford), we obtain the result that BEPO nvt/ 5.98×10^{17} = MWD/AT. Although other comparisons^(13,16) yield numbers between 5.5×10^{17} and 6.6×10^{17} , we shall use the relationship derived above; thus, $\Phi_G = 0.0962 \times \text{BEPO nvt}$.

In all figures used in this report, the neutron fluences are quoted in terms of Φ_G as per the above relationships. However, for simple comparisons (For example: the relative damage rate as a function of position within the shield composite, or for comparing shield designs with different thickness of steel and graphite.), one can use the relationships,

$$\begin{aligned} \Phi_G &= 0.74 \times \Phi(E > 15\text{KeV}) \text{ or} \\ \Phi_G &= 0.80 \times \Phi(E > 20\text{KeV}) \end{aligned}$$

which are derived in Appendix B.

STORED ENERGY

The term "stored energy" has been used to denote the increase in internal energy which is associated with the defects produced during irradiation. When these defects are removed (for example, by annealing at a temperature above the irradiation temperature), this excess internal energy is released as sensible heat. Methods for measuring the released stored energy have been reviewed by Bell and Simmons,⁽¹⁶⁾ and by Nightingale.⁽¹⁷⁾

Total stored energy can be measured by the difference between the heat of combustion of an irradiated and that of an unirradiated sample of graphite. Figure 1 shows the total amount of stored energy accumulated⁽¹⁸⁾ at irradiation temperatures of 25°C, 80-100°C and 180°C.^(a) However, for the present application, we are primarily concerned with the amount of stored energy released as a function of temperature. Various "linear temperature

(a) The highest fluence shown in Figure 1 is equivalent to less than 2 years exposure at the highest flux position in the shield, but more than 100 years operation at the rear face of the shield.

FIGURE 1. Total Stored Energy as a Function of Exposure and Irradiation Temperature, Degrees Celsius.

rise" (LTR), methods have been devised to determine the amount of stored energy release as a function of annealing temperature. Typical release curves⁽¹⁹⁾ are shown in Figure 2 for the samples described in Table 2; only "relative neutron doses" are listed because the samples had been annealed and re-irradiated.

FIGURE 2. Linear-Rise Curves of Stored Energy Release

TABLE 2. Details of Samples in Figure 2

<u>Number of Sample</u>	<u>Relative Neutron Dose</u>	<u>Estimated Temperature of Irradiation, °C</u>
1	0.39	30
2	0.81	50
3	1.00	80

Figure 3 shows a release curve, obtained from a linear rise calorimeter for a graphite sample irradiated at 80°C to the equivalent of slightly less than one year of exposure at the front face. One might easily conclude that external heat would be required to increase the sample temperature to Temperature A,

FIGURE 3. Energy Release from Irradiated Graphite.
Irradiation: $1.6 \times 10^{19}/\text{cm}^2$ at 80°C ,
● specimen 1; ○ Specimen 2

FIGURE 4. Calculated Rate of Release of Energy in a Linear Rise Experiment at $2^\circ\text{C per minute}$ and 2°C per hour

and that a spontaneous increase would take place until Temperature B was reached. However, the shape of the release curve is dependent on the rate of temperature rise, as is shown in Figure 4.

It has been shown, by Primak⁽²⁰⁻²²⁾ and others,⁽²³⁻²⁶⁾ that the annealing processes are characterized by a broad spectrum of activation energies; this result implies that the amount of stored energy released will depend on irradiation duration and the time-temperature history during storage,^(21,27) as well as the fluence and irradiation temperature. This is an area where much additional experimental and analytical work needs to be done. However, the uncertainties introduced are, for our present purposes, probably not more significant than the uncertainties introduced by lack of solid information on the irradiation conditions and accuracy of the measurement techniques.

Where the stored energy release exceeds the specific heat, as in Figure 3, adiabatic calorimetry can provide much more useful information than can linear-rise calorimetry. The technique used for adiabatic calorimetry is to initiate a stored energy release in the sample and then to follow the time-temperature history without allowing either a heat-gain-from or a heat-loss-to, the external surroundings. Two such adiabatic release curves⁽²⁸⁾ are shown in Figure 5. The "starting temperature (the temperature at which a self-sustained rise in temperature is initiated) is really a range of temperatures; at the lower "starting temperatures" the rate of stored energy release is quite slow. The rate-of-release of stored energy changes by about an order of magnitude for each 20°C change in "starting temperature",⁽¹⁹⁾ so it becomes progressively more difficult to obtain adiabatic conditions as the "starting temperature" is reduced; this effect is illustrated in Figure 6.

Nevertheless, it is useful to define the "starting temperature" as the temperature at which the adiabatic temperature rise reaches some defined rate. Figure 7 shows a correlation between "starting temperature" and irradiation temperature, where the starting temperature is defined as the temperature at which the rate-of-rise is 12°C/min. As can be seen from Figure 7, the starting temperature is generally 30 to 50°C above the irradiation temperature.⁽²⁸⁾

FIGURE 5. Adiabatic Rise Curves. Irradiation: (a) 35°C, $1.8 \times 10^{19}/\text{cm}^2$, (b) 70°C, $3.1 \times 10^{19}/\text{cm}^2$

FIGURE 6. The Effect of Starting Temperature on Adiabatic Energy Release. Irradiation: 25°C, 1.3×10^{19} total dose, 0.6×10^{19} since last anneal.

FIGURE 7. Effect of Irradiation Temperature on the Starting Temperature to Give Standard Adiabatic Release Rate

Figure 8 shows the amount of stored energy released adiabatically,⁽¹⁸⁾ as a function of irradiation temperature and neutron fluence.^(a) The amount of available data is very small; but, it can be seen that the amount releasable from a graphite irradiated at 80°C, to the equivalent of 2 years at the position of maximum fluence in the test cell shield, should not exceed about 65 calories per gram. Figure 9 shows the heat capacity of graphite⁽²⁹⁾ as a function of temperature. Assuming a starting temperature of $80 + 40^{\circ}\text{C} = 120^{\circ}\text{C}$ and 65 calories per gram released, the maximum temperature which the graphite could reach (under adiabatic conditions) would be 345°C. Under actual conditions the release would not occur adiabatically, the full 65 cal/gm would not be released, and the maximum temperature would be less than 345°C.

Moreover, it can be seen that the amount of stored energy which is adiabatically releaseable is rapidly approaching a saturation value; the total stored energy continues to increase, but not the amount released adiabatically.

Only a very small amount of information on annealing of stored energy is available at exposures greater than shown in Figure 8. Neubert and Lees⁽²³⁾ reported that 40 cal/gm was released below 250°C (during a 9-10°C/min. LTR) from a sample irradiated to $5.8 \times 10^{19}/\text{cm}^2$ at 55°C in a Hanford CTH. Powell, et. al., reported⁽³⁰⁾ about the same stored energy was released below 300°C after one-half of that exposure at 55°C in the Oak Ridge Reactor. Morgan^(b) annealed graphite samples, which had been irradiated⁽²⁶⁾ in the Hanford CTH to $2.85 \times 10^{20}/\text{cm}^2$ and $6.05 \times 10^{20}/\text{cm}^2$ (equivalent to about 15 and 32 years, respectively at the highest fluence position in the FMIT test cell shield). These anneals (5 1/2 hr at 350°C) reduced the total stored energy by 72.0 and 86.5 cal/gm respectively. Although precise irradiation temperatures are not known, they were certainly less than 80°C. Thus, the evidence indicates that the maximum stored energy released adiabatically from graphite irradiated at 80°C

-
- (a) Note: Although the curves are drawn as continuous from zero fluence, one should not interpret this as an indication that an adiabatic release could occur in any reasonable time-frame at the lower fluences. At the lower levels of releasable stored energy, an adiabatic release might require months or years (rather than the minutes shown in Figure 4) and, considering the time-scale involved, the release would certainly not be adiabatic.
- (b) Unpublished data, W. C. Morgan, Hanford Atomic Products Operation, May and August 1962.

FIGURE 8. Adiabatically Released Stored Energy

FIGURE 9. Heat Capacity of Graphite

will not exceed 80 cal/gm; an adiabatic release of this magnitude would produce a temperature of about 390°C.

The above conclusion is also compatible with observations reported on annealing of nuclear reactor moderators;^(28,30) maximum temperatures on the order of 300°C have been observed during a kinetic release of stored energy.

During a typical stored energy annealing operation at BEPO^(19,28) the graphite is slowly heated by slowly increasing the temperature of the flowing air; graphite temperatures first lag behind the air temperatures, then slowly increase and, finally, increase rapidly (about 10°C/min) at some position within the moderator. Following this rapid rise, the release phenomenon proceeds to spread like a thermal wave, moving with a velocity of about 2 cm/min as predicted by Foreman.⁽³¹⁾

If the heating of the air was discontinued, or the air temperature was reduced, appreciable annealing could still occur without initiating the rapid temperature rise and the resultant thermal wave (see Figure 6).

Figure 10 illustrates⁽¹⁹⁾ the effect of isothermal annealing, below the critical temperature, on the spontaneous temperature rise. The four runs are on pieces of a sample irradiated at about 40°C in Windscale. All LTR runs were made at 2°C/min; samples B, C & D were first heated to 110°C at 2°C/hr (see Figure 4), prior to the LTR runs; samples C and D were then held at 110°C for 48 hours and 240 hours, respectively, before the LTR run.

OTHER EFFECTS

- Commercial (non-nuclear) grade graphites may contain a few hundred parts per million (each) of such impurities as iron, calcium, sulfur, and silicon; however, except for boron⁽³²⁾, none of the impurities are known to affect the changes in physical properties during irradiation. Low boron content is desirable for commercial electrode graphites, and can be specified on the purchase specifications.

FIGURE 10. Effect of Pretreatment at 110°C on Linear-Rise Curves

- Dimensional changes are highly dependent⁽³³⁻³⁵⁾ on the raw materials and processing conditions used to manufacture the graphite, as well as on the irradiation temperature. Manufacturing processes and sources of raw material are generally considered proprietary by the graphite manufacturers, so it is advisable to work with the manufacturer in selecting material for a particular application.

In general, it can be said that small crystallite size, low thermal expansion and minimum graphitization temperatures are desirable properties if one is seeking minimum dimensional changes at low temperatures. TSGBF^(a) (Texas-Lockport coke, Standard coal-tar-pitch binder, Gas Bake graphitization process, "F" purification process), which was used in the K Reactors, meets these criteria and is the most dimensionally stable production-grade nuclear graphite which has been irradiated in the Hanford CTH. After a CTH exposure equivalent to the maximum in the FMIT shield, the TSGBF samples had contracted⁽³⁶⁾

(a) TSGBF was a "limited production" grade, produced using coke from a single source during a brief span of time; coke from a single source varies with time as the feed-stock changes. Even the feed-stock derived from a single oil well changes with time; thus, a "TSGBF", produced today, would not be identical to the TSGBF which was produced and tested more than 25 years ago.

about 0.3% in the direction parallel to, and expanded about 1.8% in the direction transverse to, the axis of extrusion of the bar. An irradiation temperature of 80°C would decrease the transverse expansion,^(25,37) but could increase the parallel contraction.

Simmons has postulated⁽³⁷⁾ that a graphite with a mean thermal expansion coefficient of $0.75 \times 10^{-6}/^{\circ}\text{C}$, measured over the temperature range -130°C to -20°C, should be dimensionally stable when irradiated at 80°C; however, although there is a general agreement between the theory and experimental data, there are significant discrepancies^(37,38) which preclude its literal use, especially over a wide exposure range, such as for the FMIT test cell shield.

- Reductions in thermal conductivity by a factor of 10 to 20^(15,33-35) may be expected, depending on the grade of graphite. This change will be beneficial in that it will increase the interior temperatures of the graphite slabs and, thus, reduce the stored energy.
- Carbon-14 will be produced by the $^{13}\text{C} (n,\gamma)^{14}\text{C}$ activation reaction, with the naturally occurring ^{13}C being augmented by the $^{12}\text{C}(n,\gamma)^{13}\text{C}$ reaction; however, the $^{14}\text{N} (n,p)^{14}\text{C}$ activation reaction may be of more importance, especially as this ^{14}C will be produced in the circulating coolant.

CONCLUSIONS AND RECOMMENDATIONS

1. There do not appear to be any major problems associated with use of graphite in the composite shield structure.
2. The grade of graphite should be selected, primarily, on the basis of cost and estimated dimensional stability.
3. The purchasing agent should work with the manufacturer's "product-oriented" staff in selecting the graphite "grade", or the materials and process used to produce the final product.
4. Serious consideration should be given to periodic (yearly) low temperature annealing of the stored energy and dimensional changes.

5. After 20 years operation without annealing, an adiabatic release of stored energy would increase the temperature of the graphite in the front shield panel to about 400°C and the temperature of the rear panel to about 250°C . Heat transfer to the steel and coolant during the release would decrease a) the graphite temperature as a function of time, b) the rate of stored energy release, c) the amount of stored energy released, and d) the maximum temperature which could be attained at the end of the release.

APPENDIX A
DAMAGE CROSS SECTIONS

The energy ranges of the group structure, reported by Carter,^(b) are rather broad for radiation damage calculations; however, they are adequate, considering the present uncertainties in shield design and the antiquity of the radiation damage data. The group-averaged cross-sections used to calculate the Relative Damage (see Table 1) were obtained in the following way:

1. Zijp⁽⁹⁾ used the Thompson and Wright form of the damage function,⁽⁵⁾ with the total scattering cross-section for carbon from the Winfrith (Aldermaston) Nuclear Data Library,⁽³⁹⁾ to calculate the group-averaged damage cross-sections shown in Table 3. The PEGGI⁽⁹⁾ form of the fission neutron spectrum was used to average^(a) the cross-sections for the two groups above 0.1 MeV:

$$\int_0^{2.25} \sigma(U) \phi(U) dU \bigg/ \int_0^{2.25} \phi(U) dU = 726 \text{ barns}$$

$$\int_{2.25}^{4.50} \sigma(U) \phi(U) dU \bigg/ \int_{2.25}^{4.50} \phi(U) dU = 732 \text{ barns}$$

Where U is lethargy,

$$U = \ln E_0/E,$$

with $E_0 = 10 \text{ MeV}$.

At the lower energies, it is not possible to average the cross-sections in this manner; therefore, a smoothed decrease in cross-section was assumed for the regions below Lethargy 4.50 ($E = 111.1 \text{ KeV}$). Table 4 lists the smoothed cross-sections; a simple average was then obtained (i.e., $\phi(U) = \text{a constant}$) for each of the two other groups above 1.0 KeV:

-
- (a) Although the averaging process is denoted here as an integral, in actuality the averages were obtained by summation of the group-averaged quantities listed in Table 3.
- (b) Personal communication from L. L. Carter, Hanford Engineering Development Lab., February 1980.

TABLE 3. ^{235}U Fission Spectrum-Averaged Cross Sections

Group	Lethargy Range	Lower Energy	$\Sigma_d(a)$ (barns)	Relative Flux(b)
1	0 - 0.25	7.79 MeV	610	0.0248
2	0.25 - 0.50	6.07 MeV	632	0.0681
3	0.50 - 0.75	4.72 MeV	548	0.1580
4	0.75 - 1.00	3.68 MeV	827	0.2753
5	1.00 - 1.25	2.86 MeV	882	0.3855
6	1.25 - 1.50	2.26 MeV	669	0.4533
7	1.50 - 1.75	1.74 MeV	679	0.4761
8	1.75 - 2.00	1.35 MeV	701	0.4502
9	2.00 - 2.25	1.05 MeV	747	0.3949
10	2.25 - 2.50	0.821 MeV	777	0.3275
11	2.50 - 2.75	0.639 MeV	792	0.2600
12	2.75 - 3.00	0.498 MeV	772	0.1996
13	3.00 - 3.25	0.387 MeV	736	0.1754
14	3.25 - 3.50	0.302 MeV	694	0.1098
15	3.50 - 3.75	0.235 MeV	639	0.0794
16	3.75 - 4.00	0.183 MeV	570	0.0569
17	4.00 - 4.25	0.143 MeV	490	0.0402
18	4.25 - 4.50	0.111 MeV	410	0.0284
19	4.50 - 4.75	86.5 keV	338	0.0199
20	4.75 - 5.00	67.4 keV	277	0.0138
21	5.00 - 7.00	9.12 keV	105	0.0037
22	7.00 - 9.00	1.23 keV		0.00019
23	9.00 - 11.00	0.167 keV		0.000013
24	11.00 - 13.00	22.6 eV		0
25	13.00 - 15.00	3.06 eV		0
26	15.00 - 16.5884	0.625 eV		0
27	>16.5884	(thermal)		

(a) See References 3 and 9.

(b) Flux per unit lethargy, $\phi(U)$, in this group (Reference 9).

$$\int_0^{5.0} \phi(U) = 1.000$$

$$\int_0^{\infty} \phi(U) = 1.0079$$

$$\int_{4.50}^{6.25} \sigma(U) \phi(U) dU \bigg/ \int_{4.50}^{6.25} \phi(U) dU = 207 \text{ barns}$$

$$\int_{6.25}^{9.25} \sigma(U) \phi(U) dU \bigg/ \int_{6.25}^{9.25} \phi(U) dU = 74 \text{ barns}$$

TABLE 4. Extrapolated Cross Sections

<u>Lower Lethargy</u>	<u>Lower Energy (keV)</u>	<u>Σ_d (barns)</u>
4.50	111.1	410
4.75	86.5	338
5.00	67.4	277
5.25	52.5	227
5.50	40.9	187
5.75	31.8	157
6.00	24.8	137
6.25	19.3	127
6.50	15.0	118
6.75	11.7	110
7.00	9.12	102
7.25	7.10	94
7.50	5.53	86
7.75	4.31	78
8.00	3.35	70
8.25	2.61	62
8.50	2.03	54
8.75	1.58	46
9.00	1.23	38
9.25	0.96	30
9.50	0.75	22
9.75	0.58	14
10.00	0.45	6

APPENDIX B
A USEFUL EXPOSURE UNIT

In 1963, Dahl and Yosikawa⁽⁴⁰⁾ showed that a unique energy, E_1 , could be identified such that the ratio:

$$\bar{\Sigma}_{d(E>E_1)} = \frac{\int_0^\infty \bar{\Sigma}_d(E) \phi(E) dE}{\int_{E_1}^\infty \phi(E) dE}$$

remained a constant for a wide variety of neutron spectra. Thus, Equation 1 can be written in the form:

$$\Phi_G = \frac{\bar{\Sigma}_d(E>E_1) \int_{t_1}^{t_2} \int_{E_1}^\infty \phi(E,t) dE dt}{\bar{\Sigma}_f}$$

However, $\phi(E>E_1) \equiv \int_{t_1}^{t_2} \int_{E_1}^\infty \phi(E,t) dE dt$, and

Equation 1 can be rewritten as:

$$\Phi_G = \frac{\bar{\Sigma}_d(E>E_1)}{\bar{\Sigma}_f} \phi(E>E_1)$$

where the ratio $\bar{\Sigma}_d(E>E_1) / \bar{\Sigma}_f$ is approximately constant over a wide range of neutron spectra. Moreover, they reported that the result $E_1 = 0.065$ MeV was obtained (for various fission reactor spectra), even when quite different models were used for $\Sigma_d(E)$. This result has been confirmed by several other investigators,⁽⁴¹⁻⁴³⁾ and provides a useful tool for estimating the change in damage production per neutron as the neutron flux spectrum is altered (for example, by deeper penetration into the shield).

Table 5 lists the ratio of the damage fluence, ϕ_G , to the total fluence of neutrons with energy greater than E_1 , $\phi(E > E_1)$, for the neutron flux spectrum at the front and rear faces of the shield composite. The ratios are plotted in Figure 11, and it can be seen that:

$$\phi_G \simeq 0.74 \times \phi(E > 15 \text{ KeV}) \text{ or } 0.80 \times \phi(E > 20 \text{ KeV})$$

For simple projections of the damage production in the graphite laminations within different steel/graphite composites these relationships should be reasonably accurate.

TABLE 5. Calculation of Fluence Ratios

Energy, E_1	Front Face		Rear Face	
	$\phi(E > E_1)$	$\frac{\phi_G}{\phi(E > E_1)}$	$\phi(E > E_1)$	$\frac{\phi_G}{\phi(E > E_1)}$
1.0 MeV	0.028	14.29	0.030	4.67
0.1 MeV	0.320	1.25	0.094	1.49
20 keV	0.505	0.79	0.170	0.82
1.0 keV	0.719	0.56	0.389	0.36

FIGURE 11. Ratio of Damage Fluence to Fluence of Neutrons with Energy Greater Than E_1

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