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EXPERIMENTAL DETERMINATION OF RADIATION DAMAGE
FUNCTION FOR GRAPHITE

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SUMMARY

Highly oriented pyrolytic-graphite samples were irradiated in three different neutron sources with average energies of 1.5, 5.5, and 14.8 MeV, respectively. The resulting changes in the basal-plane elastic shear moduli of these samples show that the relative damage rate is substantially higher at the higher neutron energies than is predicted by existing models. The results of these experiments have been used to generate a new relative damage function which will allow better projections of graphite radiation behavior in fusion reactors based on data obtained by irradiations in fission reactors.

CONTENTS

	<u>Page</u>
SUMMARY	iii
INTRODUCTION	1
EXPERIMENTAL	5
SAMPLE PREPARATION	5
MEASUREMENT OF ELASTIC SHEAR MODULUS	5
IRRADIATIONS	7
RESULTS	11
MODULUS CHANGES	11
GENERATION OF DAMAGE FUNCTION	14
DISCUSSION	19
CONCLUSIONS	21
FUTURE WORK	21
ACKNOWLEDGEMENTS	21
REFERENCES	23
APPENDIX A	25

INTRODUCTION

Graphite has potential for various uses in fusion reactors, e.g., as a neutron moderator, as limiters, as first-wall armor, and as a low-Z liner between the plasma and the first structural wall.⁽¹⁾ In many of these applications, most of the radiation damage will be due to neutrons with energies in excess of 10 MeV (for a D,T plasma). Very few neutrons in fission reactors have such high energies; so, the extensive irradiation data that have been generated in support of fission reactor programs may not be entirely applicable to fusion reactors unless the damage potential of the different spectra can be correlated.

Additional graphite irradiation data that will be required for fusion reactor applications (e.g., new materials, and irradiation temperatures up to 2000°C) must also be generated via fission reactor irradiations (at least until a large-volume high-flux source of 14 MeV neutrons becomes available.) Techniques such as simulation of neutron radiation damage via ion bombardment, which are used with metals, are not generally applicable to graphite.⁽²⁾

Methods of correlating the relative radiation damage produced per unit neutron fluence in graphite by different fission reactor spectra have been developed from theoretical considerations. It is currently recommended⁽³⁾ that graphite irradiation data in fission reactors be reported in terms of the "Equivalent Fission Fluence for Damage in Graphite," Φ_G , which is defined as follows:

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

where

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

$\sigma_s(E)$ = scattering cross section of carbon as a function of neutron energy,

$p(E)$ = atomic displacement weighting function,

$\chi(E)$ = flux spectrum for primary fission neutrons, and

$t_2 - t_1$ = duration of the irradiation test.

The denominator in equation 1 is a constant; therefore, equation 1 can be reduced to the more convenient form:

$$\Phi_G = \int_0^\infty \int_{t_1}^{t_2} \sigma_d(E) \phi(E,t) dt dE \quad (2)$$

where $\sigma_d(E)$ which can be termed either a "normalized atomic displacement cross section" or a "displacement function," is given by:

$$\sigma_d(E) = \frac{\sigma_s(E) p(E)}{\int_0^\infty \sigma_s(E) p(E) \chi(E) dE / \int_0^\infty \chi(E) dE} \quad (3)$$

The weighting function, $p(E)$, is the number of atoms displaced by the subsequent series of atom-atom collisions initiated by a primary knock-on atom displaced by a neutron with initial energy E . Calculations of atomic displacement cross-sections based on the Thompson and Wright⁽⁴⁾ model of the weighting function are currently used to correlate relative radiation damage effects between different fission reactor spectra.

A more recent set of atomic displacement cross-sections has been developed by Morgan⁽⁵⁾ from the third version of the Evaluated Nuclear Data File ENDF/B-III of neutron cross sections and the Lindhard theoretical treatment for the weighting function. In this model, Morgan attempted to compensate for n- α and other high energy reactions which are not important in fission reactors but must be taken into account for fusion reactors. Thus, his model should allow better correlation of radiation damage at higher neutron energies. It should be noted that helium produced by the n- α reactions will readily diffuse out of graphite, at least at elevated temperatures,⁽⁶⁾ rather than remain to cause problems such as the helium embrittlement that takes place in metals.

Despite Morgan's attempt to compensate for high energy reactions, enough uncertainties remain to warrant experimental confirmation of the relative damage rate as a function of energy. By measuring the damage (property change) rate in several different known neutron spectra, an experimentally derived "damage function," $D(E)$, can be obtained using suitable computer codes (see, for example, reference 7); $D(E)$ then replaces $\sigma_d(E)$ in equation 2.

Because of the low neutron fluxes at 14 MeV neutron sources, the property change to be measured must be extremely sensitive to neutron damage at low fluences. Seldin and Nezbeda⁽⁸⁾ and Taylor and Kline⁽⁹⁾ have shown that changes in the basal plane elastic shear modulus, C_{44} , can be detected in samples of Highly Oriented Pyrolytic Graphite (HOPG) after neutron fluences as low as $10^{12}/\text{cm}^2$.

Unirradiated values of C_{44} are very dependent on crystal perfection; in general, C_{44} is lower for HOPG than for more perfect natural crystals which, in turn, have a lower modulus than would a theoretically perfect crystal. Fewer dislocations in the more perfect crystals are responsible for this trend. During irradiation, defects are created which pin dislocations and, perhaps in some cases, assume a configuration that directly tends to lock the basal planes more effectively than dislocation pinning does.^(8,9) Thus, the shear modulus is raised at quite low fluences (a few times $10^{18}/\text{cm}^2$) to the value that a perfect crystal would have. At fluences much below this saturation value, it can reasonably be assumed that the damage as measured by the change in C_{44} is proportional to the number of atomic displacements.

By irradiating HOPG in sources with different neutron energies and measuring the relative change in C_{44} , the relative "damage" caused by the different neutron energies can be determined, allowing confirmation or adjustment of the calculated atomic displacement cross sections. This paper describes the results from an experiment in which HOPG samples were irradiated to neutron fluences in the range $\sim 4 \times 10^{14}$ to $\sim 1.5 \times 10^{17}$ /cm² EFF^(a) in three different sources with average neutron energies of about 14.8, 5.5 and 1.5 MeV, respectively.

(a) Fluences used are in terms of Equivalent Fission Fluence (EFF) for Damage in Graphite⁽³⁾ using the damage function derived in this paper rather than the Thompson and Wright function recommended by reference 3.

EXPERIMENTAL

SAMPLE PREPARATION

The highly oriented pyrolytic graphite (HOPG) used in this study was obtained from Union Carbide Corporation. They prepare HOPG by starting with as-deposited pyrolytic graphite and applying a compressive force perpendicular to the predominant basal plane direction at temperatures above 2800°C .⁽¹⁰⁾ The resulting material is characterized by the mosaic spread, or angular dispersion, of c-axes which is defined as the full width at half maximum intensity of the (002) orientation distribution function as determined by x-ray diffraction methods. Mosaic spreads for the three grades of HOPG used in this study were nominally 1° , 4° , and 6° .

Samples, nominally 5 mm square and 1 mm thick in the c-axis direction, were prepared as follows: first, flat plates of HOPG about 25 mm square were mounted with "Quickmount" resin and mechanically polished, first on one side and then the other, to 1 mm thickness. Maximum variation in thickness of the plates at this point was ± 0.02 mm. These plates were then held between two pieces of polycrystalline graphite with a slight compressive stress in the c-axis direction, and cut into 5 mm squares using a circular diamond saw.

MEASUREMENT OF ELASTIC SHEAR MODULUS

A schematic representation of the experimental set-up used to measure C_{44} before and after irradiation is shown in Figure 1. A one MHz shear wave generated by the lead-zirconate-titanate (PZT) transmitter crystal passed through the aluminum buffer rod and then through the HOPG sample in the direction parallel to the c-axis. The time required for the shear wave to pass through the HOPG sample was determined by using an oscilloscope to measure the time delay between the signals received by the two PZT receivers. The shear wave velocity, V , is related⁽¹¹⁾ to C_{44} by

Figure 1. Schematic Diagram of Apparatus Used to Measure Shear Modulus.

$$E \equiv C_{44} = \rho v^2$$

where ρ is the graphite density (2.26 Mg/m^3).^(a) The resin half of Barco Bond type MB-165 epoxy glue was used as the couplant between the PZT crystals, aluminum buffer rod, and HOPG sample.

IRRADIATIONS

Three different neutron sources were used for the irradiations:

- 1) The Rotating Target Neutron Source (RTNS) at Lawrence Livermore Laboratory (LLL) where neutrons are generated over a narrow energy band at $\sim 14.8 \text{ MeV}$;
- 2) The D, Be neutron source at Davis, California, where neutrons are produced over a broad energy band with average energy, in this case, of $\sim 5.5 \text{ MeV}$;
- 3) The Brookhaven (BNL) Medical Research Reactor where neutrons are produced with approximately a fission spectrum and the average energy was $\sim 1.5 \text{ MeV}$.

About 25 HOPG samples were irradiated in each source. Neutrons in the first two sources travel predominantly in one direction. Most samples were oriented so that the neutrons were moving parallel to the c-axes but a few were oriented so that the neutrons were moving perpendicular to the c-axes. Neutron motion in the BNL reactor was random, so sample orientation was unimportant. In all three irradiations, samples and metal dosimetry foils were sealed in a helium atmosphere inside an aluminum capsule. Sample temperatures in the accelerator target rooms at Davis and LLL were not monitored but were probably about 25°C and could not have been above 35°C . Temperatures of the coolant water adjacent to the capsules in the BNL irradiations were measured to be in the range 16 to 18°C except for a half-hour period during the intermediate (three hour) irradiation when water temperature was 11.5°C . Sample temperatures were calculated to be only 1 to 5°C above water temperature.

(a) Density reported by Union Carbide Corporation for HOPG.

Disintegration rate data and the irradiation time-histories were used to calculate the saturated reaction rates of the dosimetry foils. The saturated reaction rates were then used as input to the SAND-II Code⁽⁷⁾ which was used to unfold the neutron spectra for the Davis and BNL facilities. At LLL, the neutrons were essentially monoenergetic at ~ 14.8 MeV; fluences were calculated using an activation cross section of 458 millibarns for the $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$ reaction as recommended by RTNS Staff.⁽¹²⁾ The neutron energy spectra are shown in Figure 2.

Neutron fluxes at LLL and Davis decreased rapidly with distance from the source, ranging at LLL from 1.1×10^{10} to $8.3 \times 10^{11}/\text{cm}^2\text{sec EFF}$ and at Davis from 5.1×10^9 to $9.9 \times 10^{11}/\text{cm}^2\text{sec EFF}$. The fluxes at BNL were constant at $3.4 \times 10^{12}/\text{cm}^2\text{sec EFF}$, and three different irradiation times (20 min, 3 hr, and 7 hr) were used.

Figure 2. Neutron Energy Spectra.

RESULTS

MODULUS CHANGES

Following irradiation, the shear moduli were found to have increased by factors of up to seven times the unirradiated values. It was also found, however, that the moduli slowly decreased with time back toward the unirradiated values over periods up to two years even though the samples had been both irradiated and stored at room temperature. Up to 40% of the irradiation-induced change in the moduli eventually annealed out. Despite expenditure of considerable time and effort, the cause of this phenomenon has not been unequivocally established. We believe that at least part of the change was due to thermal annealing, but mechanical stresses placed on the samples while handling during the moduli measurements may also have had an effect. Details of our findings have been published elsewhere.⁽¹³⁾

This annealing phenomenon caused some uncertainty as to what value of the moduli changes, $\Delta E/E$, should be used for the correlation of irradiation damage at the different neutron energies. Three possibilities were considered: 1) extrapolation of the data back to zero time (i.e., end of the irradiation period); 2) use of the same interval between the end of irradiation and the time of the moduli measurements (e.g., 19 days); and 3) use of $\Delta E/E$ values obtained after very long times. The first two possibilities were rejected for the following reasons: 1) the method of extrapolating back to zero time was uncertain because the moduli seemed to be effected by both time and the measurements themselves; and 2) sample temperatures during shipment back to PNL following the irradiations were unknown and could have made a difference in the annealing if they were substantially different from room temperature. The third possibility was chosen when, after careful study, it appeared that the modulus had approached a value, E_∞ , which would not be changed by additional time or additional measurements. It is worth noting, however, that the relative slopes shown in Figures 3 and 4 and in Tables 1-3 (pages 15 & 16) would change by no more than about 20% if one of the other two values of $\Delta E/E$ had been used instead.

Figure 3. Comparison of Modulus Changes for 1° Graphites.

Figure 4. Comparison of Modulus Changes for All Three Types of Graphites.

Figure 3 shows $\Delta E_{\infty}/E$ values obtained after very long times plotted versus neutron fluence for the 1⁰ grade of graphite. There is reason to believe that the relationship between modulus change and fluence should be linear, at least up to the point where the defects become so numerous and close together that there is some interaction between them. Our data obey a linear relationship quite well up to $\Delta E_{\infty}/E = 1.6$ as shown in Figure 3. Figure 4 shows data for all three grades of graphite up to $\Delta E_{\infty}/E = 1.6$. Clearly, there is a poor correlation between damage rates for the three different sources when the comparison is based on fluence of neutrons with energies greater than 50 keV.

It was stated earlier that samples with their c-axes oriented both parallel and perpendicular to the neutron beam were included in the Davis and LLL irradiations. All samples whose c-axes were oriented perpendicular to the beam were smaller and, therefore, more delicate than the other samples. As a result, none of the 1⁰ samples survived the modulus measurements. Results for the 4⁰ and 6⁰ samples with different orientations were not readily comparable because fluences for samples oriented perpendicular to the beam were much higher than for all but one (6⁰ sample at LLL) of those oriented parallel to the beam. Also, fluences for the perpendicular oriented samples were high enough so that data were in the non-linear region. Although the data were insufficient for accurate comparisons, it appeared that there were no major differences in moduli changes for the different sample orientations.

GENERATION OF DAMAGE FUNCTION

The lines shown in Figure 4 represent the least squares fit to the data points with the lines constrained to pass through the origin. The slopes of these lines are listed in Table 1.

In Table 2, the slopes are listed on a relative basis for each graphite, averaged for each facility, and normalized with the Davis average as unity. SAND-II requires input ratios (slopes) in terms of fluences above 10^{-10} MeV, so the last column shows the normalized slopes in terms of $\Phi(E > 10^{-4} \text{ eV})$.

TABLE 1. Rates of Property Change $\Delta E_{\infty}/E$ Versus Fluence ($E > 50$ keV)

Facility	Average Neutron Energy (MeV)	Slope ^(a)		
		1 ⁰	4 ⁰	6 ⁰
Davis	5.5	0.760	0.675	0.462
LLL	14.8	0.621	0.565	0.379
BNL	1.5	0.295	0.262	0.204
Average		0.559	0.501	0.349

(a) The "slope" is the relative change in $\Delta E_{\infty}/E$ produced by a fluence of $10^{16}/\text{cm}^2$ ($E > 50$ keV).

TABLE 2. Relative Rates of Property Change Versus Fluence

Facility	Relative Slope ^(a)			Average ± 1 Std Deviation	Normalized	
	1 ⁰	4 ⁰	6 ⁰		$\phi(E > 50 \text{ keV})$	$\phi(E > 10^{-4} \text{ eV})$
Davis	1.360	1.347	1.324	1.344 + 0.018	1.000	1.000
LLL	1.111	1.128	1.086	1.108 + 0.021	0.825	0.825
BNL	0.528	0.523	0.585	0.545 + 0.028	0.406	0.0751

(a) The relative slopes were obtained by dividing the individual slopes in Table 1 by the average slope.

With unfolding codes, such as SAND-II, the best results are obtained when a reasonable approximation of the desired function is input along with property change rates and neutron spectra. For these calculations, we chose to input the atomic displacement cross sections developed by Jenkins⁽¹⁴⁾ for one set of calculations, and the cross sections developed by Morgan⁽⁵⁾ for a second set of calculations. The output damage functions from the two calculations agreed quite well, except in the energy regions where these three spectra did not provide adequate response; the two output functions were therefore averaged to provide a new input function for the third set of calculations. In Figure 5, the third (final) output damage function is compared with the two input atomic displacement functions. Table 3 summarizes the measured and predicted relative damage (or number of displaced atoms) in the three spectra.

TABLE 3. Comparison of Damage Rates

Facility	Relative Damage Rates ^(a)			
	Measured	Calculated		
		Jenkins ⁽¹⁴⁾	Morgan ⁽⁵⁾	SAND-II
LLL	2.031	0.470	1.347	2.038
Davis	2.462	0.755	1.218	2.538
BNL	1.000	1.000	1.000	1.000

(a) Relative property change per unit fluence, $E > 50$ keV.

Data shown in Figure 6 are the same as those in Figure 3 except they have been replotted with fluences based on the new damage function.

Appendix A contains more extensive tabulations of the neutron spectra, displacement functions, and new damage function.

FIGURE 5. Relative Radiation Damage Normalized to Unity Over the ^{235}U Fission Spectrum.

Figure 6. Comparison of Modulus Changes for 1° Graphite Based on New Damage Function.

DISCUSSION

The validity and general applicability of the derived damage function rests on three primary assumptions:

- The change in shear modulus, ΔE_∞ , is a measure of the number of irradiation-induced defects that remain at the end of an irradiation.
- In a given neutron irradiation facility, damage (modulus change) is proportional to neutron fluence.
- The character of the defects which produced the moduli changes is independent of neutron energy.

The first and second assumptions are generally accepted from previous studies^(8,9) and are further supported by our data which show that the moduli changed linearly with fluence up to $\Delta E_\infty/E = 1.6$.

The validity of the third assumption has not been definitely verified. In fact, there were some differences in the annealing behavior between samples irradiated in the different facilities. As noted earlier, however, this caused only minor uncertainties in the relative modulus-change rates that were used to generate the new damage function.

In an effort to directly verify our third assumption, samples from each of the three irradiations were examined using high resolution electron microscopy. The defects were found to be barely resolvable and differences between the three sets of samples, if any, were quite subtle. Results of these studies will be published in more detail elsewhere,⁽¹⁵⁾ but they neither prove nor disprove the validity of this assumption with any degree of certainty.

CONCLUSIONS

Results presented here show that neutron radiation damage rates in graphite are much higher at high neutron energies. Based on total fluences of neutrons with energies greater than 50 keV, the damage rate at 14.8 MeV is about two times higher than in the Medical Research Reactor at Brookhaven National Laboratory. The latter is a typical, well-thermalized, fission reactor. Damage rates in fusion reactors will depend upon the neutron energy spectrum for the particular application being considered, but they clearly will be higher than expected for most applications, particularly those near the plasma.

FUTURE WORK

Our work needs to be extended to include the measurement of other property changes at much higher fluences. In particular, a comparison of dimensional changes at different neutron energies is needed at fluences greater than $1 \times 10^{21}/\text{cm}^2$. The first 14 MeV neutron source with high enough flux and sufficient sample space to conduct such an experiment will probably be the FMIT facility⁽¹⁶⁾ being built at Hanford.

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APPENDIX A

TABULATIONS OF RESULTS

APPENDIX A

TABULATION OF RESULTS

Table A-1 lists the 100-group GAM-II group structure which is used in the following tables:

Tables A-2 to A-4 list the two input functions and the new damage function.

Tables A-5 to A-7 list the relative displacement (or damage) rates calculated for irradiations in the Davis facility, and Tables A-8 to A-10 list the same quantities for the BNL facility. Because the neutron spectrum at the LLL facility is essentially mono-energetic at 14.8 MeV, the relative damage (or displacement) rate is numerically equal to the value of the damage (or displacement) function for lethargy group 1.

TABLE A-1. Neutron Energy and Lethargy Limits for 100 Group Structure
(Energy limits are given in electron volts)

Group Number	Upper Energy	Lower Energy	Lower Lethargy	Group Number	Upper Energy	Lower Energy	Lower Lethargy
1	1.4918+07*	1.3499+07	0.10	51	8.6517+04	6.7379+04	5.40
2	1.3499+07	1.2214+07	0.20	52	6.7379+04	5.2475+04	5.65
3	1.2214+07	1.1052+07	0.30	53	5.2475+04	4.0868+04	5.90
4	1.1052+07	1.0000+07	0.40	54	4.0868+04	3.1828+04	6.15
5	1.0000+07	9.0484+06	0.50	55	3.1828+04	2.4784+04	6.40
6	9.0484+06	8.1873+06	0.60	56	2.4788+04	1.9305+04	6.65
7	8.1873+06	7.4082+06	0.70	57	1.9305+04	1.5034+04	6.90
8	7.4082+06	6.7032+06	0.80	58	1.5034+04	1.1709+04	7.15
9	6.7032+06	6.0653+06	0.90	59	1.1709+04	9.1188+03	7.40
10	6.0653+06	5.4881+06	1.00	60	9.1188+03	7.1017+03	7.65
11	5.4881+06	4.9659+06	1.10	61	7.1017+03	5.5308+03	7.90
12	4.9659+06	4.4933+06	1.20	62	5.5308+03	4.3074+03	8.15
13	4.4933+06	4.0657+06	1.30	63	4.3074+03	3.3546+03	8.40
14	4.0657+06	3.6788+06	1.40	64	3.3546+03	2.6128+03	8.65
15	3.6788+06	3.3287+06	1.50	65	2.6126+03	2.0347+03	8.90
16	3.3287+06	3.0119+06	1.60	66	2.0347+03	1.5846+03	9.15
17	3.0119+06	2.7253+06	1.70	67	1.5846+03	1.2341+03	9.40
18	2.7253+06	2.4660+06	1.80	68	1.2341+03	9.6112+02	9.65
19	2.4660+06	2.2313+06	1.90	69	9.6112+02	7.4852+02	9.90
20	2.2313+06	2.0190+06	2.00	70	7.4852+02	5.8295+02	10.15
21	2.0190+06	1.8268+06	2.10	71	5.8295+02	4.5400+02	10.40
22	1.8268+06	1.6530+06	2.20	72	4.5400+02	3.5358+02	10.65
23	1.6530+06	1.4957+06	2.30	73	3.5358+02	2.7536+02	10.90
24	1.4957+06	1.3534+06	2.40	74	2.7536+02	2.1445+02	11.15
25	1.3534+06	1.2246+06	2.50	75	2.1445+02	1.6702+02	11.40
26	1.2246+06	1.1080+06	2.60	76	1.6702+02	1.3007+02	11.65
27	1.1080+06	1.0026+06	2.70	77	1.3007+02	1.0130+02	11.90
28	1.0026+06	9.0718+05	2.80	78	1.0130+02	7.8893+01	12.15
29	9.0718+05	8.2085+05	2.90	79	7.8893+01	6.1442+01	12.40
30	8.2085+05	7.4274+05	3.00	80	6.1442+01	4.7651+01	12.65
31	7.4274+05	6.7206+05	3.10	81	4.7651+01	3.7267+01	12.90
32	6.7206+05	6.0810+05	3.20	82	3.7267+01	2.9023+01	13.15
33	6.0810+05	5.5023+05	3.30	83	2.9023+01	2.2603+01	13.40
34	5.5023+05	4.9787+05	3.40	84	2.2603+01	1.7603+01	13.65
35	4.9787+05	4.5049+05	3.50	85	1.7603+01	1.3710+01	13.90
36	4.5049+05	4.0762+05	3.60	86	1.3710+01	1.0677+01	14.15
37	4.0762+05	3.6883+05	3.70	87	1.0677+01	8.3153+00	14.40
38	3.6883+05	3.3373+05	3.80	88	8.3153+00	6.4760+00	14.65
39	3.3373+05	3.0197+05	3.90	89	6.4760+00	5.0435+00	14.90
40	3.0197+05	2.7324+05	4.00	90	5.0435+00	3.9279+00	15.15
41	2.7324+05	2.4724+05	4.10	91	3.9279+00	3.0590+00	15.40
42	2.4724+05	2.2371+05	4.20	92	3.0590+00	2.3824+00	15.65
43	2.2371+05	2.0242+05	4.30	93	2.3824+00	1.8554+00	15.90
44	2.0242+05	1.8316+05	4.40	94	1.8554+00	1.4450+00	16.15
45	1.8316+05	1.6573+05	4.50	95	1.4450+00	1.1254+00	16.40
46	1.6573+05	1.4966+05	4.60	96	1.1254+00	8.7643+01	16.65
47	1.4966+05	1.3569+05	4.70	97	8.7643+01	6.8756+01	16.90
48	1.3569+05	1.2277+05	4.80	98	6.8756+01	5.3158+01	17.15
49	1.2277+05	1.1109+05	4.90	99	5.3158+01	4.1399+01	17.40
50	1.1109+05	8.6517+04	5.15	100	4.1399+01	(THERMAL)	*****

*±N represents 10^{±N}

TABLE A-2. Normalization Factor for Equivalent Fission Fluence
Based on Jenkins' Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	U-235 Fission Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
1	.4614	4.122-005*	4.122-005	1.902-005	1.902-005
2	.5076	1.133-004	1.545-004	5.752-005	7.654-005
3	.5352	2.770-004	4.316-004	1.482-004	2.248-004
4	.4543	6.091-004	1.041-003	2.767-004	5.015-004
5	.4224	1.217-003	2.258-003	5.140-004	1.016-003
6	.4348	2.233-003	4.491-003	9.709-004	1.956-003
7	.8300	3.791-003	8.282-003	3.147-003	5.133-003
8	.4875	6.003-003	1.429-002	2.926-003	5.059-003
9	.5951	8.930-003	2.322-002	5.314-003	1.337-002
10	.5899	1.256-002	3.577-002	7.409-003	2.073-002
11	.7135	1.679-002	5.257-002	1.198-002	3.276-002
12	.8595	2.145-002	7.402-002	1.844-002	5.120-002
13	1.0983	2.632-002	1.003-001	2.891-002	8.011-002
14	1.2732	3.113-002	1.315-001	3.964-002	1.197-001
15	1.4417	3.564-002	1.671-001	5.138-002	1.711-001
16	.9672	3.961-002	2.067-001	3.831-002	2.094-001
17	1.2094	4.289-002	2.496-001	5.187-002	2.613-001
18	.8963	4.536-002	2.950-001	4.066-002	3.020-001
19	.8565	4.696-002	3.419-001	4.022-002	3.422-001
20	.9815	4.771-002	3.896-001	4.683-002	3.390-001
21	.8913	4.766-002	4.373-001	4.248-002	4.315-001
22	.9188	4.689-002	4.842-001	4.308-002	4.746-001
23	.9456	4.551-002	5.297-001	4.321-002	5.178-001
24	.9804	4.363-002	5.733-001	4.278-002	5.606-001
25	1.0095	4.138-002	6.147-001	4.177-002	6.023-001
26	1.0361	3.886-002	6.536-001	4.026-002	6.426-001
27	1.0597	3.618-002	6.898-001	3.834-002	6.809-001
28	1.0798	3.342-002	7.232-001	3.609-002	7.170-001
29	1.0961	3.065-002	7.538-001	3.359-002	7.506-001
30	1.1084	2.793-002	7.816-001	3.096-002	7.816-001
31	1.1163	2.531-002	8.071-001	2.826-002	8.098-001
32	1.1204	2.282-002	8.299-001	2.557-002	8.354-001
33	1.1201	2.048-002	8.504-001	2.294-002	8.583-001
34	1.1160	1.830-002	8.687-001	2.042-002	8.788-001
35	1.1078	1.630-002	8.850-001	1.806-002	8.958-001
36	1.0954	1.446-002	8.994-001	1.584-002	9.127-001
37	1.0789	1.280-002	9.122-001	1.381-002	9.265-001
38	1.0583	1.129-002	9.235-001	1.195-002	9.384-001
39	1.0304	9.941-003	9.334-001	1.024-002	9.487-001
40	.9987	8.732-003	9.422-001	8.721-003	9.574-001

*±N represents $10^{\pm N}$

TABLE A-2. (continued)

Lethargy Group Number	Relative Displ. Function	U-235 Fission Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.9646	7.654-003	9.498-001	7.383-003	9.648-001
42	.9284	6.698-003	9.565-001	6.219-003	9.710-001
43	.8905	5.852-003	9.624-001	5.211-003	9.762-001
44	.8513	5.105-003	9.675-001	4.346-003	9.805-001
45	.8113	4.448-003	9.719-001	3.609-003	9.842-001
46	.7707	3.871-003	9.758-001	2.984-003	9.871-001
47	.7301	3.365-003	9.792-001	2.457-003	9.896-001
48	.6898	2.923-003	9.821-001	2.016-003	9.916-001
49	.6499	2.536-003	9.846-001	1.648-003	9.933-001
50	.5820	4.959-003	9.896-001	2.886-003	9.961-001
51	.4907	3.458-003	9.931-001	1.697-003	9.978-001
52	.4076	2.403-003	9.955-001	9.794-004	9.988-001
53	.3336	1.666-003	9.971-001	5.558-004	9.994-001
54	.2689	1.153-003	9.983-001	3.101-004	9.997-001
55	.2144	7.967-004	9.991-001	1.708-004	9.999-001
56	.1705	5.498-004	9.996-001	9.373-005	9.999-001
57	.1354	3.791-004	1.000+000	5.135-005	1.000+000
58	.1075	.000	1.000+000	.000	1.000+000
59	.0852	.000	1.000+000	.000	1.000+000
60	.0677	.000	1.000+000	.000	1.000+000
61	.0536	.000	1.000+000	.000	1.000+000
62	.0425	.000	1.000+000	.000	1.000+000
63	.0337	.000	1.000+000	.000	1.000+000
64	.0267	.000	1.000+000	.000	1.000+000
65	.0211	.000	1.000+000	.000	1.000+000
66	.0167	.000	1.000+000	.000	1.000+000
67	.0132	.000	1.000+000	.000	1.000+000
68	.0104	.000	1.000+000	.000	1.000+000
69	.0082	.000	1.000+000	.000	1.000+000
70	.0065	.000	1.000+000	.000	1.000+000
71	.0051	.000	1.000+000	.000	1.000+000
72	.0039	.000	1.000+000	.000	1.000+000
73	.0027	.000	1.000+000	.000	1.000+000
74	.0011	.000	1.000+000	.000	1.000+000
75	.0000	.000	1.000+000	.000	1.000+000
76	.0000	.000	1.000+000	.000	1.000+000
77	.0000	.000	1.000+000	.000	1.000+000
78	.0000	.000	1.000+000	.000	1.000+000

- (1) The U-235 fission neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral, and the displacement rate has been normalized to unity over the integral.
- (2) Neutrons per cm^2 per second in this lethargy group.
- (3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.
- (4) Relative displacement rate in this lethargy group.
- (5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-3. Normalization Factor for Equivalent Fission Fluence
Based on Morgan's Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	U-235 Fission Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ Rate(5)
1	1.2368	4.122-005*	4.122-005	5.107-005	5.107-005
2	1.3612	1.133-004	1.545-004	1.542-004	2.053-004
3	1.3476	2.770-004	4.316-004	3.733-004	5.756-004
4	1.0024	6.091-004	1.041-003	6.105-004	1.189-003
5	1.0936	1.217-003	2.258-003	1.331-003	2.520-003
6	1.0253	2.233-003	4.491-003	2.289-003	4.610-003
7	1.6829	3.791-003	8.282-003	6.380-003	1.119-002
8	.7792	6.003-003	1.429-002	4.678-003	1.587-002
9	.9843	8.930-003	2.322-002	8.790-003	2.466-002
10	.8195	1.256-002	3.577-002	1.029-002	3.495-002
11	.9035	1.679-002	5.257-002	1.517-002	5.012-002
12	1.0344	2.145-002	7.402-002	2.219-002	7.231-002
13	1.4938	2.632-002	1.003-001	3.932-002	1.116-001
14	1.5000	3.113-002	1.315-001	4.670-002	1.583-001
15	1.8325	3.564-002	1.671-001	6.531-002	2.236-001
16	1.0873	3.961-002	2.067-001	4.307-002	2.667-001
17	1.4100	4.289-002	2.496-001	6.048-002	3.272-001
18	.8342	4.536-002	2.950-001	3.784-002	3.650-001
19	.7989	4.696-002	3.419-001	3.752-002	4.025-001
20	.9675	4.771-002	3.896-001	4.616-002	4.487-001
21	.8310	4.766-002	4.373-001	3.961-002	4.883-001
22	.8716	4.689-002	4.842-001	4.087-002	5.292-001
23	.9047	4.551-002	5.297-001	4.117-002	5.703-001
24	.9331	4.363-002	5.733-001	4.071-002	6.110-001
25	.9555	4.138-002	6.147-001	3.954-002	6.506-001
26	.9731	3.886-002	6.536-001	3.782-002	6.384-001
27	.9898	3.618-002	6.898-001	3.581-002	7.242-001
28	.9978	3.342-002	7.232-001	3.335-002	7.576-001
29	1.0030	3.065-002	7.538-001	3.074-002	7.883-001
30	1.0013	2.793-002	7.818-001	2.797-002	8.163-001
31	.9972	2.531-002	8.071-001	2.524-002	8.415-001
32	.9889	2.282-002	8.299-001	2.257-002	8.641-001
33	.9718	2.048-002	8.504-001	1.990-002	8.840-001
34	.9528	1.830-002	8.687-001	1.744-002	9.014-001
35	.9313	1.630-002	8.850-001	1.518-002	9.166-001
36	.9066	1.446-002	8.994-001	1.311-002	9.297-001
37	.8793	1.280-002	9.122-001	1.125-002	9.410-001
38	.8500	1.129-002	9.235-001	9.597-003	9.506-001
39	.8184	9.941-003	9.334-001	8.136-003	9.537-001
40	.7862	8.732-003	9.422-001	6.865-003	9.656-001

*±N represents $10^{\pm N}$

TABLE A-3. (continued)

Lethargy Group Number	Relative Displ. Function	U-235 Fission Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.7512	7.654-003	9.498-001	5.749-003	9.713-001
42	.7170	6.698-003	9.565-001	4.802-003	9.761-001
43	.6818	5.852-003	9.624-001	3.990-003	9.801-001
44	.6464	5.105-003	9.675-001	3.300-003	9.834-001
45	.6116	4.448-003	9.719-001	2.721-003	9.861-001
46	.5771	3.871-003	9.758-001	2.234-003	9.884-001
47	.5558	3.365-003	9.792-001	1.870-003	9.902-001
48	.5299	2.923-003	9.821-001	1.549-003	9.918-001
49	.5180	2.536-003	9.846-001	1.314-003	9.931-001
50	.5921	4.959-003	9.896-001	2.936-003	9.960-001
51	.4981	3.458-003	9.931-001	1.722-003	9.977-001
52	.4205	2.403-003	9.955-001	1.010-003	9.988-001
53	.3467	1.666-003	9.971-001	5.776-004	9.993-001
54	.2837	1.153-003	9.983-001	3.271-004	9.997-001
55	.2303	7.967-004	9.991-001	1.835-004	9.998-001
56	.1859	5.498-004	9.996-001	1.022-004	9.999-001
57	.1494	3.791-004	1.000+000	5.665-005	1.000+000
58	.1195	.000	1.000+000	.000	1.000+000
59	.0953	.000	1.000+000	.000	1.000+000
60	.0757	.000	1.000+000	.000	1.000+000
61	.0601	.000	1.000+000	.000	1.000+000
62	.0476	.000	1.000+000	.000	1.000+000
63	.0377	.000	1.000+000	.000	1.000+000
64	.0298	.000	1.000+000	.000	1.000+000
65	.0235	.000	1.000+000	.000	1.000+000
66	.0185	.000	1.000+000	.000	1.000+000
67	.0146	.000	1.000+000	.000	1.000+000
68	.0115	.000	1.000+000	.000	1.000+000
69	.0090	.000	1.000+000	.000	1.000+000
70	.0071	.000	1.000+000	.000	1.000+000
71	.0056	.000	1.000+000	.000	1.000+000
72	.0044	.000	1.000+000	.000	1.000+000
73	.0034	.000	1.000+000	.000	1.000+000
74	.0027	.000	1.000+000	.000	1.000+000
75	.0021	.000	1.000+000	.000	1.000+000
76	.0016	.000	1.000+000	.000	1.000+000
77	.0013	.000	1.000+000	.000	1.000+000
78	.0010	.000	1.000+000	.000	1.000+000

(1) The U-235 fission neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral, and the displacement rate has been normalized to unity over the integral.

(2) Neutrons per cm^2 per second in this lethargy group.

(3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.

(4) Relative displacement rate in this lethargy group.

(5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-4. Normalization Factor for Equivalent Fission Fluence
Based on New Damage Function

Lethargy Group Number	Relative Damage Function	U-235 Fission Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Damage Rate Per Group(4)	Integral Damage Rate(5)
1	1.4363	4.122-005 *	4.122-005	5.920-005	5.920-005
2	1.7114	1.133-004	1.545-004	1.939-004	2.531-004
3	1.9363	2.770-004	4.316-004	5.364-004	7.895-004
4	1.7616	6.092-004	1.041-003	1.073-003	1.863-003
5	1.8884	1.218-003	2.258-003	2.300-003	4.163-003
6	1.8858	2.233-003	4.491-003	4.211-003	8.374-003
7	1.8788	3.791-003	8.282-003	7.123-003	1.550-002
8	1.8719	6.003-003	1.429-002	1.124-002	2.673-002
9	1.8684	8.930-003	2.322-002	1.668-002	4.342-002
10	1.5865	1.256-002	3.577-002	1.993-002	6.355-002
11	1.7272	1.679-002	5.257-002	2.900-002	9.234-002
12	2.0085	2.145-002	7.402-002	4.308-002	1.354-001
13	2.2898	2.632-002	1.003-001	6.027-002	1.957-001
14	2.4304	3.113-002	1.315-001	7.566-002	2.714-001
15	2.1225	3.564-002	1.671-001	7.565-002	3.470-001
16	1.9032	3.961-002	2.067-001	7.539-002	4.224-001
17	1.4646	4.289-002	2.496-001	6.282-002	4.352-001
18	1.0260	4.536-002	2.950-001	4.654-002	5.317-001
19	.8067	4.696-002	3.419-001	3.738-002	5.696-001
20	.8086	4.771-002	3.896-001	3.858-002	6.082-001
21	.7860	4.766-002	4.373-001	3.746-002	6.457-001
22	.7410	4.689-002	4.842-001	3.475-002	6.804-001
23	.6960	4.551-002	5.297-001	3.168-002	7.121-001
24	.6734	4.363-002	5.733-001	2.938-002	7.415-001
25	.7813	4.138-002	6.147-001	3.233-002	7.738-001
26	.6796	3.886-002	6.536-001	2.641-002	8.002-001
27	.6690	3.618-002	6.898-001	2.420-002	8.244-001
28	.6584	3.342-002	7.232-001	2.200-002	8.464-001
29	.6531	3.065-002	7.538-001	2.002-002	8.664-001
30	.6478	2.793-002	7.818-001	1.809-002	8.845-001
31	.6408	2.531-002	8.071-001	1.622-002	9.007-001
32	.6267	2.282-002	8.299-001	1.430-002	9.130-001
33	.6126	2.048-002	8.504-001	1.255-002	9.276-001
34	.6056	1.830-002	8.687-001	1.108-002	9.387-001
35	.5864	1.630-002	8.850-001	9.553-003	9.482-001
36	.5749	1.446-002	8.994-001	8.313-003	9.565-001
37	.5520	1.280-002	9.122-001	7.065-003	9.636-001
38	.5290	1.129-002	9.235-001	5.973-003	9.696-001
39	.5176	9.941-003	9.334-001	5.145-003	9.747-001
40	.4925	8.732-003	9.422-001	4.301-003	9.790-001

* ±N represents $10^{\pm N}$

TABLE A-4 (Continued)

Lethargy Group Number	Relative Damage Function	U-235 Fission Neutron Flux Per Group ⁽²⁾	Integral Neutron Flux ⁽³⁾	Relative Damage Rate Per Group ⁽⁴⁾	Integral Damage Rate ⁽⁵⁾
41	.4792	7.654-003	9.498-001	3.668-003	9.827-001
42	.4527	6.698-003	9.565-001	3.032-003	9.857-001
43	.4261	5.852-003	9.624-001	2.493-003	9.882-001
44	.4128	5.105-003	9.675-001	2.107-003	9.903-001
45	.3846	4.448-003	9.719-001	1.711-003	9.920-001
46	.3735	3.871-003	9.758-001	1.446-003	9.935-001
47	.3513	3.365-003	9.792-001	1.182-003	9.947-001
48	.3291	2.923-003	9.821-001	9.620-004	9.956-001
49	.3180	2.536-003	9.846-001	8.065-004	9.964-001
50	.2877	4.959-003	9.896-001	1.426-003	9.979-001
51	.2589	3.458-003	9.931-001	3.954-004	9.988-001
52	.2262	2.403-003	9.955-001	5.434-004	9.993-001
53	.1935	1.666-003	9.971-001	3.223-004	9.996-001
54	.1607	1.153-003	9.983-001	1.853-004	9.998-001
55	.1272	7.967-004	9.991-001	1.013-004	9.999-001
56	.1012	5.498-004	9.996-001	5.565-005	1.000+000
57	.0800	3.791-004	1.000+000	3.033-005	1.000+000
58	.0636	.000	1.000+000	.000	1.000+000
59	.0506	.000	1.000+000	.000	1.000+000
60	.0401	.000	1.000+000	.000	1.000+000
61	.0317	.000	1.000+000	.000	1.000+000
62	.0252	.000	1.000+000	.000	1.000+000
63	.0199	.000	1.000+000	.000	1.000+000
64	.0157	.000	1.000+000	.000	1.000+000
65	.0121	.000	1.000+000	.000	1.000+000
66	.0098	.000	1.000+000	.000	1.000+000
67	.0077	.000	1.000+000	.000	1.000+000
68	.0061	.000	1.000+000	.000	1.000+000
69	.0048	.000	1.000+000	.000	1.000+000
70	.0038	.000	1.000+000	.000	1.000+000
71	.0030	.000	1.000+000	.000	1.000+000
72	.0023	.000	1.000+000	.000	1.000+000
73	.0018	.000	1.000+000	.000	1.000+000
74	.0014	.000	1.000+000	.000	1.000+000
75	.0012	.000	1.000+000	.000	1.000+000
76	.0009	.000	1.000+000	.000	1.000+000
77	.0007	.000	1.000+000	.000	1.000+000
78	.0006	.000	1.000+000	.000	1.000+000

- (1) The U-235 fission neutron flux has been normalized to 1.0 neutrons per cm² per second over the integral, and the damage rate has been normalized to unity over the integral.
- (2) Neutrons per cm² per second in this lethargy group.
- (3) Neutrons per cm² per second with energies greater than the lower energy for this group.
- (4) Relative damage rate in this lethargy group.
- (5) Relative damage rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-5. Relative Displacement Rate at UC Davis Based on
Jenkins' Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate, Per Group(4)	Integral Displ. Rate(5)
1	.4614	9.874-003*	9.874-003	4.556-003	4.556-003
2	.5076	2.313-002	3.301-002	1.174-002	1.630-002
3	.5352	3.771-002	7.072-002	2.018-002	3.648-002
4	.4543	4.770-002	1.184-001	2.167-002	5.815-002
5	.4224	5.703-002	1.754-001	2.409-002	8.224-002
6	.4348	6.560-002	2.410-001	2.852-002	1.108-001
7	.8300	7.300-002	3.140-001	6.059-002	1.714-001
8	.4875	8.099-002	3.950-001	3.948-002	2.108-001
9	.5951	8.605-002	4.811-001	5.121-002	2.620-001
10	.5899	9.056-002	5.716-001	5.342-002	3.155-001
11	.7135	8.510-002	6.567-001	6.072-002	3.762-001
12	.8595	5.978-002	7.265-001	5.998-002	4.362-001
13	1.0983	5.579-002	7.823-001	6.127-002	4.974-001
14	1.2732	4.454-002	8.269-001	5.671-002	5.541-001
15	1.4417	3.529-002	8.622-001	5.088-002	6.050-001
16	.9672	2.807-002	8.902-001	2.715-002	6.322-001
17	1.2094	2.184-002	9.121-001	2.642-002	6.536-001
18	.8963	1.699-002	9.291-001	1.523-002	6.738-001
19	.8565	1.327-002	9.423-001	1.137-002	6.852-001
20	.9815	1.037-002	9.527-001	1.018-002	6.954-001
21	.8913	8.422-003	9.611-001	7.506-003	7.029-001
22	.9188	7.123-003	9.682-001	6.545-003	7.094-001
23	.9496	5.932-003	9.742-001	5.633-003	7.151-001
24	.9804	4.860-003	9.790-001	4.764-003	7.198-001
25	1.0095	3.896-003	9.829-001	3.933-003	7.238-001
26	1.0361	3.055-003	9.860-001	3.165-003	7.269-001
27	1.0597	2.538-003	9.885-001	2.689-003	7.296-001
28	1.0798	2.126-003	9.907-001	2.295-003	7.319-001
29	1.0961	1.784-003	9.924-001	1.955-003	7.339-001
30	1.1084	1.460-003	9.939-001	1.618-003	7.355-001
31	1.1165	1.146-003	9.950-001	1.279-003	7.368-001
32	1.1204	9.046-004	9.959-001	1.014-003	7.378-001
33	1.1201	7.569-004	9.967-001	8.478-004	7.386-001
34	1.1160	6.219-004	9.973-001	6.941-004	7.393-001
35	1.1078	5.114-004	9.978-001	5.666-004	7.399-001
36	1.0954	4.117-004	9.982-001	4.510-004	7.403-001
37	1.0789	3.239-004	9.986-001	3.405-004	7.407-001
38	1.0587	2.723-004	9.988-001	2.882-004	7.410-001
39	1.0304	2.296-004	9.991-001	2.366-004	7.412-001
40	.9927	1.818-004	9.993-001	1.816-004	7.414-001

* ±N represents $10^{\pm N}$

TABLE A-5. (continued)

Lethargy Group Number	Relative Displ. Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.9646	1.474-004	9.994-001	1.422-004	7.415-001
42	.9284	1.170-004	9.995-001	1.086-004	7.416-001
43	.8905	1.005-004	9.996-001	8.950-005	7.417-001
44	.8513	7.956-005	9.997-001	6.773-005	7.418-001
45	.8113	7.332-005	9.998-001	5.949-005	7.418-001
46	.7707	5.740-005	9.998-001	4.424-005	7.419-001
47	.7301	4.547-005	9.999-001	3.320-005	7.419-001
48	.6898	3.558-005	9.999-001	2.455-005	7.420-001
49	.6499	3.440-005	9.999-001	2.236-005	7.420-001
50	.5820	5.366-005	1.000+000	3.123-005	7.420-001
51	.4907	.000	1.000+000	.000	7.420-001
52	.4076	.000	1.000+000	.000	7.420-001
53	.3336	.000	1.000+000	.000	7.420-001
54	.2689	.000	1.000+000	.000	7.420-001
55	.2144	.000	1.000+000	.000	7.420-001
56	.1705	.000	1.000+000	.000	7.420-001
57	.1354	.000	1.000+000	.000	7.420-001
58	.1075	.000	1.000+000	.000	7.420-001
59	.0853	.000	1.000+000	.000	7.420-001
60	.0677	.000	1.000+000	.000	7.420-001
61	.0536	.000	1.000+000	.000	7.420-001
62	.0425	.000	1.000+000	.000	7.420-001
63	.0337	.000	1.000+000	.000	7.420-001
64	.0267	.000	1.000+000	.000	7.420-001
65	.0211	.000	1.000+000	.000	7.420-001
66	.0167	.000	1.000+000	.000	7.420-001
67	.0132	.000	1.000+000	.000	7.420-001
68	.0104	.000	1.000+000	.000	7.420-001
69	.0082	.000	1.000+000	.000	7.420-001
70	.0065	.000	1.000+000	.000	7.420-001
71	.0051	.000	1.000+000	.000	7.420-001
72	.0039	.000	1.000+000	.000	7.420-001
73	.0027	.000	1.000+000	.000	7.420-001
74	.0011	.000	1.000+000	.000	7.420-001
75	.0000	.000	1.000+000	.000	7.420-001
76	.0000	.000	1.000+000	.000	7.420-001
77	.0000	.000	1.000+000	.000	7.420-001
78	.0000	.000	1.000+000	.000	7.420-001

(1) The neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral.

(2) Neutrons per cm^2 per second in this lethargy group.

(3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.

(4) Relative displacement rate in this lethargy group.

(5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-6. Relative Displacement Rate at UC Days Based on Morgan's Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
1	1.2388	9.874-003*	9.874-003	1.223-002	1.223-002
2	1.3612	2.313-002	3.301-002	3.149-002	4.372-002
3	1.3476	3.771-002	7.072-002	5.082-002	9.454-002
4	1.0024	4.770-002	1.184-001	4.781-002	1.424-001
5	1.0936	5.703-002	1.754-001	6.237-002	2.047-001
6	1.0253	6.560-002	2.410-001	6.726-002	2.720-001
7	1.6829	7.300-002	3.140-001	1.229-001	3.948-001
8	.7792	8.099-002	3.950-001	6.311-002	4.579-001
9	.9843	8.605-002	4.811-001	8.470-002	5.426-001
10	.8195	9.056-002	5.716-001	7.421-002	6.169-001
11	.9035	8.510-002	6.567-001	7.689-002	6.937-001
12	1.0344	6.978-002	7.265-001	7.218-002	7.659-001
13	1.4938	5.579-002	7.823-001	8.334-002	8.493-001
14	1.5000	4.454-002	8.269-001	6.681-002	9.161-001
15	1.8325	3.529-002	8.622-001	6.467-002	9.808-001
16	1.0872	2.807-002	8.902-001	3.052-002	1.011+000
17	1.4100	2.184-002	9.121-001	3.080-002	1.042+000
18	.8342	1.699-002	9.291-001	1.417-002	1.056+000
19	.7989	1.327-002	9.423-001	1.060-002	1.067+000
20	.9675	1.037-002	9.527-001	1.003-002	1.077+000
21	.8310	8.422-003	9.611-001	6.998-003	1.084+000
22	.8716	7.123-003	9.682-001	6.208-003	1.090+000
23	.9047	5.932-003	9.742-001	5.366-003	1.095+000
24	.9331	4.860-003	9.790-001	4.535-003	1.100+000
25	.9555	3.896-003	9.829-001	3.722-003	1.104+000
26	.9731	3.055-003	9.860-001	2.973-003	1.107+000
27	.9898	2.538-003	9.885-001	2.512-003	1.109+000
28	.9978	2.126-003	9.907-001	2.121-003	1.111+000
29	1.0030	1.784-003	9.924-001	1.789-003	1.113+000
30	1.0013	1.460-003	9.939-001	1.462-003	1.115+000
31	.9972	1.146-003	9.950-001	1.143-003	1.116+000
32	.9869	9.046-004	9.959-001	8.946-004	1.117+000
33	.9718	7.569-004	9.967-001	7.356-004	1.117+000
34	.9528	6.219-004	9.973-001	5.926-004	1.118+000
35	.9313	5.114-004	9.978-001	4.763-004	1.118+000
36	.9066	4.117-004	9.982-001	3.733-004	1.119+000
37	.8793	3.239-004	9.986-001	2.848-004	1.119+000
38	.8500	2.723-004	9.988-001	2.315-004	1.119+000
39	.8184	2.296-004	9.991-001	1.879-004	1.119+000
40	.7862	1.818-004	9.993-001	1.430-004	1.120+000

*N represents 10^{-N}

TABLE A-6. (continued)

Lethargy Group Number	Relative Displ. Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.7512	1.474-004	9.994-001	1.107-004	1.120+000
42	.7170	1.170-004	9.995-001	8.387-005	1.120+000
43	.6818	1.005-004	9.996-001	6.852-005	1.120+000
44	.6464	7.956-005	9.997-001	5.143-005	1.120+000
45	.6116	7.332-005	9.998-001	4.485-005	1.120+000
46	.5771	5.740-005	9.998-001	3.312-005	1.120+000
47	.5558	4.547-005	9.999-001	2.527-005	1.120+000
48	.5299	3.558-005	9.999-001	1.886-005	1.120+000
49	.5180	3.440-005	9.999-001	1.782-005	1.120+000
50	.5921	5.366-005	1.000+000	3.177-005	1.120+000
51	.4981	.000	1.000+000	.000	1.120+000
52	.4205	.000	1.000+000	.000	1.120+000
53	.3467	.000	1.000+000	.000	1.120+000
54	.2837	.000	1.000+000	.000	1.120+000
55	.2303	.000	1.000+000	.000	1.120+000
56	.1859	.000	1.000+000	.000	1.120+000
57	.1494	.000	1.000+000	.000	1.120+000
58	.1195	.000	1.000+000	.000	1.120+000
59	.0953	.000	1.000+000	.000	1.120+000
60	.0757	.000	1.000+000	.000	1.120+000
61	.0601	.000	1.000+000	.000	1.120+000
62	.0476	.000	1.000+000	.000	1.120+000
63	.0377	.000	1.000+000	.000	1.120+000
64	.0298	.000	1.000+000	.000	1.120+000
65	.0235	.000	1.000+000	.000	1.120+000
66	.0185	.000	1.000+000	.000	1.120+000
67	.0146	.000	1.000+000	.000	1.120+000
68	.0115	.000	1.000+000	.000	1.120+000
69	.0090	.000	1.000+000	.000	1.120+000
70	.0071	.000	1.000+000	.000	1.120+000
71	.0056	.000	1.000+000	.000	1.120+000
72	.0044	.000	1.000+000	.000	1.120+000
73	.0034	.000	1.000+000	.000	1.120+000
74	.0027	.000	1.000+000	.000	1.120+000
75	.0021	.000	1.000+000	.000	1.120+000
76	.0016	.000	1.000+000	.000	1.120+000
77	.0013	.000	1.000+000	.000	1.120+000
78	.0010	.000	1.000+000	.000	1.120+000

- (1) The neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral.
- (2) Neutrons per cm^2 per second in this lethargy group.
- (3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.
- (4) Relative displacement rate in this lethargy group.
- (5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-7. Relative Damage Rate at U. C. Davis Based on
New Damage Function

Lethargy Group Number	Relative Damage Function	Neutron Flux per Group ⁽²⁾	Integral Neutron Flux ⁽³⁾	Relative Damage Rate Per Group ⁽⁴⁾	Integral Damage Rate ⁽⁵⁾
1	1.4363	9.874-003*	9.874-003	1.418-002	1.418-002
2	1.7114	2.314-002	3.301-002	3.960-002	5.378-002
3	1.9363	3.771-002	7.072-002	7.302-002	1.268-001
4	1.7616	4.770-002	1.184-001	8.403-002	2.108-001
5	1.8884	5.703-002	1.754-001	1.077-001	3.185-001
6	1.8858	6.560-002	2.410-001	1.237-001	4.422-001
7	1.8788	7.300-002	3.141-001	1.372-001	5.794-001
8	1.8719	8.099-002	3.950-001	1.516-001	7.310-001
9	1.8684	8.605-002	4.811-001	1.608-001	8.918-001
10	1.5865	9.056-002	5.716-001	1.437-001	1.035+000
11	1.7272	8.510-002	6.567-001	1.470-001	1.182+000
12	2.0085	6.978-002	7.265-001	1.402-001	1.323+000
13	2.2898	5.579-002	7.823-001	1.277-001	1.450+000
14	2.4304	4.454-002	8.269-001	1.083-001	1.559+000
15	2.1225	3.529-002	8.622-001	7.490-002	1.633+000
16	1.9032	2.807-002	8.902-001	5.342-002	1.687+000
17	1.4646	2.184-002	9.121-001	3.199-002	1.719+000
18	1.0260	1.699-002	9.291-001	1.743-002	1.736+000
19	.8067	1.327-002	9.423-001	1.070-002	1.747+000
20	.8086	1.037-002	9.527-001	8.385-003	1.755+000
21	.7860	8.422-003	9.611-001	6.620-003	1.762+000
22	.7410	7.123-003	9.682-001	5.278-003	1.767+000
23	.6950	5.932-003	9.742-001	4.129-003	1.771+000
24	.6734	4.860-003	9.790-001	3.273-003	1.775+000
25	.7813	3.896-003	9.829-001	3.044-003	1.778+000
26	.6796	3.055-003	9.860-001	2.076-003	1.780+000
27	.6690	2.538-003	9.885-001	1.698-003	1.782+000
28	.6584	2.126-003	9.907-001	1.400-003	1.783+000
29	.6531	1.784-003	9.924-001	1.165-003	1.784+000
30	.6478	1.460-003	9.939-001	9.458-004	1.785+000
31	.6408	1.146-003	9.950-001	7.343-004	1.786+000
32	.6267	9.046-004	9.959-001	5.669-004	1.786+000
33	.6126	7.569-004	9.967-001	4.637-004	1.787+000
34	.6056	6.219-004	9.973-001	3.766-004	1.787+000
35	.5864	5.114-004	9.978-001	2.999-004	1.787+000
36	.5749	4.118-004	9.982-001	2.368-004	1.788+000
37	.5520	3.239-004	9.986-001	1.788-004	1.788+000
38	.5290	2.723-004	9.988-001	1.441-004	1.788+000
39	.5176	2.296-004	9.991-001	1.188-004	1.788+000
40	.4925	1.813-004	9.993-001	8.954-005	1.788+000

* ±N represents 10^{-N}

TABLE A-7 (Continued)

Lethargy Group Number	Relative Damage Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Damage Rate Per Group(4)	Integral Damage Rate(5)
41	.4792	1.474-004	9.994-001	7.063-005	1.788+000
42	.4527	1.170-004	9.995-001	5.297-005	1.788+000
43	.4261	1.005-004	9.996-001	4.282-005	1.788+000
44	.4128	7.956-005	9.997-001	3.284-005	1.788+000
45	.3846	7.333-005	9.998-001	2.821-005	1.788+000
46	.3735	5.740-005	9.998-001	2.144-005	1.788+000
47	.3513	4.547-005	9.999-001	1.597-005	1.789+000
48	.3291	3.558-005	9.999-001	1.171-005	1.789+000
49	.3180	3.440-005	9.999-001	1.094-005	1.789+000
50	.2877	5.366-005	1.000+000	1.544-005	1.789+000
51	.2589	.000	1.000+000	.000	1.789+000
52	.2262	.000	1.000+000	.000	1.789+000
53	.1935	.000	1.000+000	.000	1.789+000
54	.1607	.000	1.000+000	.000	1.789+000
55	.1272	.000	1.000+000	.000	1.789+000
56	.1012	.000	1.000+000	.000	1.789+000
57	.0800	.000	1.000+000	.000	1.789+000
58	.0636	.000	1.000+000	.000	1.789+000
59	.0506	.000	1.000+000	.000	1.789+000
60	.0401	.000	1.000+000	.000	1.789+000
61	.0317	.000	1.000+000	.000	1.789+000
62	.0252	.000	1.000+000	.000	1.789+000
63	.0199	.000	1.000+000	.000	1.789+000
64	.0157	.000	1.000+000	.000	1.789+000
65	.0121	.000	1.000+000	.000	1.789+000
66	.0098	.000	1.000+000	.000	1.789+000
67	.0077	.000	1.000+000	.000	1.789+000
68	.0061	.000	1.000+000	.000	1.789+000
69	.0048	.000	1.000+000	.000	1.789+000
70	.0038	.000	1.000+000	.000	1.789+000
71	.0030	.000	1.000+000	.000	1.789+000
72	.0023	.000	1.000+000	.000	1.789+000
73	.0018	.000	1.000+000	.000	1.789+000
74	.0014	.000	1.000+000	.000	1.789+000
75	.0012	.000	1.000+000	.000	1.789+000
76	.0009	.000	1.000+000	.000	1.789+000
77	.0007	.000	1.000+000	.000	1.789+000
78	.0006	.000	1.000+000	.000	1.789+000

(1) The neutron flux has been normalized to 1.0 neutrons per cm² per second over the integral.

(2) Neutrons per cm² per second in this lethargy group.

(3) Neutrons per cm² per second with energies greater than the lower energy for this group.

(4) Relative damage rate in this lethargy group.

(5) Relative damage rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-8. Relative Displacement Rate at BNL Based on Jenkins' Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	Neutron Flux per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
1	.4614	1.113-005*	1.113-005	5.136-006	5.136-006
2	.5076	2.634-005	3.747-005	1.337-005	1.850-005
3	.5352	5.593-005	9.340-005	2.994-005	4.844-005
4	.4543	1.106-004	2.040-004	5.023-005	9.867-005
5	.4224	2.068-004	4.103-004	8.736-005	1.860-004
6	.4348	3.484-004	7.592-004	1.515-004	3.375-004
7	.8300	5.293-004	1.289-003	4.394-004	7.769-004
8	.4875	7.345-004	2.023-003	3.581-004	1.135-003
9	.5951	1.003-003	3.026-003	5.968-004	1.732-003
10	.5899	1.381-003	4.407-003	8.148-004	2.547-003
11	.7135	1.905-003	6.312-003	1.359-003	3.906-003
12	.8595	2.636-003	8.947-003	2.265-003	6.171-003
13	1.0982	3.425-003	1.237-002	3.762-003	9.933-003
14	1.2732	4.178-003	1.655-002	5.319-003	1.525-002
15	1.4417	5.205-003	2.175-002	7.504-003	2.276-002
16	.9672	6.438-003	2.819-002	6.227-003	2.898-002
17	1.2094	7.455-003	3.565-002	9.016-003	3.800-002
18	.8963	8.131-003	4.378-002	7.288-003	4.529-002
19	.8565	8.965-003	5.274-002	7.679-003	5.297-002
20	.9815	9.813-003	6.256-002	9.632-003	6.260-002
21	.8913	1.010-002	7.266-002	9.005-003	7.160-002
22	.9188	1.007-002	8.273-002	9.248-003	8.085-002
23	.9496	1.004-002	9.276-002	9.531-003	9.038-002
24	.9804	1.002-002	1.028-001	9.822-003	1.002-001
25	1.0095	9.991-003	1.128-001	1.009-002	1.103-001
26	1.0361	9.935-003	1.227-001	1.029-002	1.206-001
27	1.0597	9.879-003	1.326-001	1.047-002	1.311-001
28	1.0798	9.579-003	1.422-001	1.034-002	1.414-001
29	1.0961	9.250-003	1.514-001	1.014-002	1.515-001
30	1.1084	8.906-003	1.603-001	9.871-003	1.614-001
31	1.1165	8.465-003	1.688-001	9.452-003	1.709-001
32	1.1204	8.005-003	1.768-001	8.968-003	1.798-001
33	1.1201	7.658-003	1.844-001	8.578-003	1.884-001
34	1.1160	7.378-003	1.918-001	8.233-003	1.966-001
35	1.1078	7.407-003	1.992-001	8.206-003	2.048-001
36	1.0954	7.753-003	2.070-001	8.493-003	2.133-001
37	1.0789	8.097-003	2.151-001	8.736-003	2.221-001
38	1.0582	8.273-003	2.235-001	8.861-003	2.309-001
39	1.0304	8.694-003	2.322-001	8.959-003	2.399-001
40	.9987	9.093-003	2.412-001	9.082-003	2.490-001

* ±N represents $10^{\pm N}$

TABLE A-8. (continued)

Lethargy Group Number	Relative Displ. Function	Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.9646	9.505-003	2.508-001	9.168-003	2.581-001
42	.9284	9.776-003	2.605-001	9.076-003	2.672-001
43	.8905	9.757-003	2.703-001	8.689-003	2.759-001
44	.8513	9.748-003	2.800-001	8.298-003	2.842-001
45	.8113	9.738-003	2.898-001	7.901-003	2.921-001
46	.7707	9.729-003	2.995-001	7.498-003	2.996-001
47	.7301	9.720-003	3.092-001	7.096-003	3.067-001
48	.6898	9.710-003	3.189-001	6.698-003	3.134-001
49	.6499	9.692-003	3.286-001	6.299-003	3.197-001
50	.5820	2.416-002	3.528-001	1.406-002	3.338-001
51	.4907	2.402-002	3.768-001	1.179-002	3.455-001
52	.4076	2.390-002	4.007-001	9.740-003	3.553-001
53	.3336	2.379-002	4.245-001	7.935-003	3.632-001
54	.2689	2.368-002	4.482-001	6.368-003	3.696-001
55	.2144	2.360-002	4.718-001	5.059-003	3.746-001
56	.1705	2.353-002	4.953-001	4.012-003	3.787-001
57	.1354	2.349-002	5.188-001	3.180-003	3.818-001
58	.1075	2.344-002	5.422-001	2.520-003	3.844-001
59	.0853	2.338-002	5.656-001	1.995-003	3.864-001
60	.0677	2.334-002	5.889-001	1.580-003	3.879-001
61	.0536	2.328-002	6.122-001	1.248-003	3.892-001
62	.0425	2.323-002	6.355-001	9.874-004	3.902-001
63	.0337	2.318-002	6.586-001	7.811-004	3.909-001
64	.0267	2.313-002	6.818-001	6.176-004	3.916-001
65	.0211	2.307-002	7.048-001	4.869-004	3.921-001
66	.0157	2.303-002	7.279-001	3.846-004	3.924-001
67	.0132	2.297-002	7.509-001	3.032-004	3.927-001
68	.0104	2.293-002	7.738-001	2.384-004	3.930-001
69	.0082	2.288-002	7.966-001	1.876-004	3.932-001
70	.0065	2.283-002	8.195-001	1.484-004	3.933-001
71	.0051	2.278-002	8.422-001	1.162-004	3.934-001
72	.0039	2.273-002	8.650-001	8.864-005	3.935-001
73	.0027	2.267-002	8.876-001	6.122-005	3.936-001
74	.0011	2.263-002	9.103-001	2.489-005	3.936-001
75	.0000	2.257-002	9.328-001	.000	3.936-001
76	.0000	2.252-002	9.554-001	.000	3.936-001
77	.0000	2.248-002	9.778-001	.000	3.936-001
78	.0000	2.242-002	1.000+000	.000	3.936-001

- (1) The neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral.
 (2) Neutrons per cm^2 per second in this lethargy group.
 (3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.
 (4) Relative displacement rate in this lethargy group.
 (5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-2. Relative Displacement Rate at BNL Based on Morgan's Atomic Displacement Cross Sections(1)

Lethargy Group Number	Relative Displ. Function	Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ Rate(5)
1	1.238E	1.113-005*	1.113-005	1.379-005	1.379-005
2	1.3612	2.634-005	3.747-005	3.585-005	4.964-005
3	1.3476	5.593-005	9.340-005	7.538-005	1.250-004
4	1.0024	1.106-004	2.040-004	1.108-004	2.358-004
5	1.0936	2.068-004	4.108-004	2.262-004	4.620-004
6	1.0253	3.484-004	7.592-004	3.572-004	8.192-004
7	1.6822	5.293-004	1.289-003	3.908-004	1.710-003
8	.7792	7.345-004	2.023-003	5.723-004	2.282-003
9	.9843	1.003-003	3.026-003	9.871-004	3.269-003
10	.8195	1.381-003	4.407-003	1.132-003	4.401-003
11	.9035	1.905-003	6.312-003	1.721-003	6.122-003
12	1.0344	2.636-003	8.947-003	2.726-003	8.848-003
13	1.4938	3.425-003	1.237-002	5.117-003	1.397-002
14	1.5000	4.178-003	1.655-002	6.266-003	2.023-002
15	1.8325	5.205-003	2.175-002	9.538-003	2.577-002
16	1.0873	6.438-003	2.819-002	7.000-003	3.677-002
17	1.4100	7.455-003	3.565-002	1.051-002	4.728-002
18	.8342	8.131-003	4.378-002	6.783-003	5.406-002
19	.7989	8.965-003	5.274-002	7.162-003	6.123-002
20	.9675	9.813-003	6.256-002	9.494-003	7.072-002
21	.8310	1.010-002	7.266-002	8.395-003	7.912-002
22	.8716	1.007-002	8.273-002	8.773-003	8.789-002
23	.9047	1.004-002	9.276-002	9.081-003	9.697-002
24	.9331	1.002-002	1.028-001	9.348-003	1.063-001
25	.9555	9.991-003	1.128-001	9.546-003	1.159-001
26	.9731	9.935-003	1.227-001	9.667-003	1.255-001
27	.9898	9.879-003	1.326-001	9.778-003	1.353-001
28	.9978	9.579-003	1.422-001	9.558-003	1.449-001
29	1.0030	9.250-003	1.514-001	9.278-003	1.541-001
30	1.0013	8.906-003	1.603-001	8.917-003	1.631-001
31	.9972	8.465-003	1.688-001	8.442-003	1.715-001
32	.9889	8.005-003	1.768-001	7.916-003	1.794-001
33	.9718	7.658-003	1.844-001	7.442-003	1.869-001
34	.9528	7.378-003	1.918-001	7.029-003	1.939-001
35	.9313	7.407-003	1.992-001	6.899-003	2.008-001
36	.9066	7.753-003	2.070-001	7.029-003	2.078-001
37	.8792	8.097-003	2.151-001	7.120-003	2.149-001
38	.8500	8.373-003	2.235-001	7.117-003	2.221-001
39	.8184	8.694-003	2.322-001	7.115-003	2.292-001
40	.7862	9.093-003	2.412-001	7.149-003	2.363-001

* ±N represents $10^{\pm N}$

TABLE A-9. (continued)

Lethargy Group Number	Relative Displ. Function	Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Displ. Rate Per Group(4)	Integral Displ. Rate(5)
41	.7512	9.505-003	2.508-001	7.140-003	2.435-001
42	.7170	9.776-003	2.605-001	7.009-003	2.505-001
43	.6818	9.757-003	2.703-001	6.652-003	2.571-001
44	.6464	9.748-003	2.800-001	6.301-003	2.634-001
45	.6116	9.738-003	2.898-001	5.956-003	2.694-001
46	.5771	9.729-003	2.995-001	5.615-003	2.750-001
47	.5558	9.720-003	3.092-001	5.402-003	2.804-001
48	.5299	9.710-003	3.189-001	5.145-003	2.855-001
49	.5180	9.692-003	3.286-001	5.020-003	2.906-001
50	.5921	2.416-002	3.528-001	1.430-002	3.049-001
51	.4981	2.402-002	3.768-001	1.195-002	3.168-001
52	.4205	2.390-002	4.007-001	1.005-002	3.269-001
53	.3467	2.379-002	4.245-001	8.246-003	3.351-001
54	.2837	2.368-002	4.482-001	6.719-003	3.418-001
55	.2303	2.360-002	4.718-001	5.435-003	3.473-001
56	.1859	2.353-002	4.953-001	4.375-003	3.517-001
57	.1494	2.349-002	5.188-001	3.509-003	3.552-001
58	.1195	2.344-002	5.422-001	2.801-003	3.580-001
59	.0953	2.338-002	5.656-001	2.228-003	3.602-001
60	.0757	2.334-002	5.889-001	1.767-003	3.620-001
61	.0601	2.328-002	6.122-001	1.399-003	3.634-001
62	.0476	2.323-002	6.355-001	1.106-003	3.645-001
63	.0377	2.318-002	6.586-001	8.738-004	3.653-001
64	.0298	2.313-002	6.818-001	6.893-004	3.660-001
65	.0235	2.307-002	7.048-001	5.423-004	3.666-001
66	.0185	2.303-002	7.279-001	4.260-004	3.670-001
67	.0146	2.297-002	7.508-001	3.354-004	3.673-001
68	.0115	2.293-002	7.738-001	2.636-004	3.676-001
69	.0090	2.288-002	7.966-001	2.059-004	3.678-001
70	.0071	2.283-002	8.195-001	1.621-004	3.680-001
71	.0056	2.278-002	8.422-001	1.275-004	3.681-001
72	.0044	2.273-002	8.650-001	1.000-004	3.682-001
73	.0034	2.267-002	8.876-001	7.709-005	3.683-001
74	.0027	2.263-002	9.103-001	6.109-005	3.683-001
75	.0021	2.257-002	9.328-001	4.740-005	3.684-001
76	.0016	2.252-002	9.554-001	3.604-005	3.684-001
77	.0013	2.248-002	9.778-001	2.922-005	3.684-001
78	.0010	2.242-002	1.000+000	2.242-005	3.685-001

- (1) The neutron flux has been normalized to 1.0 neutrons per cm^2 per second over the integral.
- (2) Neutrons per cm^2 per second in this lethargy group.
- (3) Neutrons per cm^2 per second with energies greater than the lower energy for this group.
- (4) Relative displacement rate in this lethargy group.
- (5) Relative displacement rate caused by neutrons with energies greater than the lower energy for this group.

TABLE A-10. Relative Damage Rate at BNL Based on New Damage Function

Lethargy Group Number	Relative Damage Function	Neutron Flux Per Group(2)	Integral Neutron Flux(3)	Relative Damage Rate Per Group(4)	Integral Damage Rate(5)
1	1.4363	1.113-005 *	1.113-005	1.599-005	1.599-005
2	1.7114	2.633-005	3.746-005	4.506-005	6.105-005
3	1.9363	5.592-005	9.338-005	1.083-004	1.693-004
4	1.7616	1.105-004	2.039-004	1.947-004	3.640-004
5	1.8884	2.067-004	4.106-004	3.903-004	7.543-004
6	1.8858	3.483-004	7.589-004	6.568-004	1.411-003
7	1.8788	5.293-004	1.288-003	9.945-004	2.406-003
8	1.8719	7.343-004	2.022-003	1.375-003	3.780-003
9	1.8684	1.002-003	3.025-003	1.872-003	5.652-003
10	1.5865	1.381-003	4.406-003	2.191-003	7.843-003
11	1.7272	1.904-003	6.310-003	3.239-003	1.113-002
12	2.0085	2.635-003	8.945-003	5.292-003	1.642-002
13	2.2898	3.425-003	1.237-002	7.842-003	2.427-002
14	2.4304	4.177-003	1.655-002	1.015-002	3.442-002
15	2.1225	5.204-003	2.175-002	1.105-002	4.546-002
16	1.9032	6.437-003	2.819-002	1.225-002	5.772-002
17	1.4646	7.453-003	3.564-002	1.092-002	6.863-002
18	1.0260	8.129-003	4.377-002	8.340-003	7.697-002
19	.8067	8.963-003	5.273-002	7.230-003	8.420-002
20	.8086	9.813-003	6.255-002	7.935-003	9.214-002
21	.7860	1.010-002	7.264-002	7.939-003	1.001-001
22	.7410	1.006-002	8.271-002	7.455-003	1.075-001
23	.6960	1.004-002	9.274-002	6.988-003	1.145-001
24	.6734	1.002-002	1.028-001	6.748-003	1.213-001
25	.7813	9.985-003	1.127-001	7.801-003	1.291-001
26	.6796	9.930-003	1.227-001	6.749-003	1.358-001
27	.6690	9.876-003	1.326-001	6.607-003	1.424-001
28	.6584	9.579-003	1.421-001	6.307-003	1.487-001
29	.6531	9.248-003	1.514-001	6.040-003	1.548-001
30	.6478	8.903-003	1.603-001	5.767-003	1.605-001
31	.6408	8.463-003	1.687-001	5.423-003	1.660-001
32	.6267	8.003-003	1.767-001	5.015-003	1.710-001
33	.6126	7.656-003	1.844-001	4.690-003	1.757-001
34	.6056	7.375-003	1.918-001	4.466-003	1.801-001
35	.5864	7.405-003	1.992-001	4.242-003	1.845-001
36	.5749	7.752-003	2.069-001	4.457-003	1.889-001
37	.5520	8.095-003	2.150-001	4.468-003	1.934-001
38	.5290	8.371-003	2.234-001	4.429-003	1.978-001
39	.5176	8.692-003	2.321-001	4.499-003	2.023-001
40	.4925	9.091-003	2.412-001	4.477-003	2.068-001

*±N represents $10^{\pm N}$

TABLE A-10. (Continued)

Lethargy Group Number	Relative Damage Function	Neutron Flux Per Group ⁽²⁾	Integral Neutron Flux ⁽³⁾	Relative Damage Rate Per Group ⁽⁴⁾	Integral Damage Rate ⁽⁵⁾
41	.4792	9.502-003	2.507-001	4.553-003	2.114-001
42	.4527	9.771-003	2.605-001	4.423-003	2.158-001
43	.4261	9.758-003	2.702-001	4.158-003	2.199-001
44	.4128	9.746-003	2.800-001	4.023-003	2.240-001
45	.3846	9.737-003	2.897-001	3.745-003	2.277-001
46	.3735	9.728-003	2.994-001	3.634-003	2.313-001
47	.3513	9.716-003	3.091-001	3.413-003	2.348-001
48	.3291	9.704-003	3.188-001	3.194-003	2.379-001
49	.3180	9.692-003	3.285-001	3.082-003	2.410-001
50	.2877	2.415-002	3.527-001	6.947-003	2.480-001
51	.2589	2.402-002	3.767-001	6.219-003	2.542-001
52	.2262	2.389-002	4.006-001	5.403-003	2.596-001
53	.1935	2.378-002	4.244-001	4.601-003	2.642-001
54	.1607	2.368-002	4.481-001	3.805-003	2.680-001
55	.1272	2.359-002	4.716-001	2.999-003	2.710-001
56	.1012	2.353-002	4.952-001	2.382-003	2.734-001
57	.0800	2.348-002	5.187-001	1.879-003	2.753-001
58	.0636	2.343-002	5.421-001	1.491-003	2.768-001
59	.0506	2.338-002	5.655-001	1.183-003	2.779-001
60	.0401	2.333-002	5.888-001	9.356-004	2.789-001
61	.0317	2.328-002	6.121-001	7.383-004	2.796-001
62	.0252	2.322-002	6.353-001	5.842-004	2.802-001
63	.0199	2.317-002	6.585-001	4.601-004	2.807-001
64	.0157	2.312-002	6.816-001	3.633-004	2.810-001
65	.0121	2.307-002	7.047-001	2.802-004	2.813-001
66	.0098	2.302-002	7.277-001	2.264-004	2.815-001
67	.0077	2.297-002	7.507-001	1.771-004	2.817-001
68	.0061	2.292-002	7.736-001	1.392-004	2.818-001
69	.0048	2.287-002	7.964-001	1.102-004	2.820-001
70	.0038	2.282-002	8.193-001	8.579-005	2.820-001
71	.0030	2.277-002	8.420-001	6.805-005	2.821-001
72	.0023	2.272-002	8.648-001	5.256-005	2.822-001
73	.0018	2.266-002	8.874-001	4.150-005	2.822-001
74	.0014	2.262-002	9.100-001	3.271-005	2.822-001
75	.0012	2.256-002	9.326-001	2.610-005	2.823-001
76	.0009	2.252-002	9.551-001	1.954-005	2.823-001
77	.0007	2.247-002	9.776-001	1.516-005	2.823-001
78	.0006	2.241-002	1.000+000	1.296-005	2.823-001

- (1) The neutron flux has been normalized to 1.0 neutrons per cm² per second over the integral.
- (2) Neutrons per cm² per second in this lethargy group.
- (3) Neutrons per cm² per second with energies greater than the lower energy for this group.
- (4) Relative damage rate in this lethargy group.
- (5) Relative damage rate caused by neutrons with energies greater than the lower energy for this group.

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